UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

July 5, 2012

David J. Bannister, Vice President and
Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION - NRC EXAMINATION REPORT 05000285/2012301

Dear Mr. Bannister:

On April 20, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an initial operator license examination at Fort Calhoun Station. The enclosed report documents the examination results and licensing decisions. The preliminary examination results were discussed on April 20, 2012, with Mike Prospero, Plant Manager, and other members of your staff. A telephonic meeting was conducted on May 14, 2012, with Mr. Thomas Giebelhausen, Manager, Operations Training, who was provided the NRC licensing decisions. The final telephonic exit meeting was conducted on June 7, 2012, with Mr. Thomas Giebelhausen.

The examination included the evaluation of three applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses, and two applicants for upgrade senior reactor operator licenses. The license examiners determined that six of the applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued. There were nine post examination comments submitted by your staff. Enclosure 1 contains details of this report and Enclosure 2 summarizes post examination comment resolution.

Additionally, the NRC has identified nineteen issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has determined that violations are associated with all of these issues. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations, consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest the violations or the significance of the non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 1600 E. Lamar Blvd, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the facility. In addition, if you disagree with the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response, if you choose to provide one for cases where a response is not required, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the Public without redaction.

Sincerely,

**/RA/**

Mark S. Haire, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-285
License: DPR-40

Enclosures:
1. NRC Examination Report 05000285/2012301, w/Attachment
2. NRC Review of FCS Written Post-Examination Comments

cc w/enclosure:

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| SUNSI Review Completed: __KDC__ | ADAMS: ☒ Yes ☐ No | | Initials: __KDC__ |
| ☒ Publicly Available | ☐ Non-Publicly Available | ☐ Sensitive | ☒ Non-Sensitive |

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**U.S. NUCLEAR REGULATORY COMMISSION**

**REGION IV**

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| Docket: | 50-285 |
| License: | DPR-40 |
| Report: | 05000285/2012301 |
| Licensee: | Omaha Public Power District |
| Facility: | Fort Calhoun Station |
| Location: | Fort Calhoun Station FC-2-4 Adm.<br>P.O. Box 399, Highway 75 – North of Fort Calhoun<br>Fort Calhoun, Nebraska |
| Dates: | March 12 – June 7, 2012 |
| Inspectors: | K. Clayton, Senior Operations Engineer<br>B. Larson, Senior Operations Engineer<br>D. Strickland, Operations Engineer<br>N. Hernandez, Operations Engineer<br>T. Buchanan, Operations Engineer |
| Approved By: | Mark Haire, Chief<br>Operations Branch<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

ER05000285/2012301; March 12 – June 7, 2012; Fort Calhoun Station; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of three applicants for reactor operator licenses, five applicants for instant senior reactor operator licenses, and two applicants for upgrade senior reactor operator licenses at Fort Calhoun Station.

The NRC developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The written examination was administered by the NRC on April 13, 2012. NRC examiners administered the operating tests during the week of April 16, 2012.

The examiners determined that six of the applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," with seven examples:

    - Example 1: There is no procedure guidance provided for tripping bistables on a trip unit if required for any instrument failure other than nuclear instrumentation. The annunciator response procedures only provide guidance to bypass the trip unit. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

    - Example 2: Abnormal Operating Procedure AOP-15, "Loss of Flux Indication or Flow Streaming" does not provide guidance during a nuclear instrument failure for tripping only those trip units that need to be tripped and bypassing the others. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03141.

    - Example 3: Procedure SO-O-1, "Conduct of Operations," and Procedure OPD-04-09, "Emergency Operating Procedure / Abnormal Operating Procedure Use and Adherence Procedure" each direct the operator to the other procedure for a discussion on the concept of procedure use and adherence in emergency operations procedure usage without addressing procedure use and adherence. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03143.

- Example 4: There is no procedural guidance in OP-2A, “Plant Startup,” on how to plot the 1/M data against reactivity and control element assembly position nor on how to determine the Estimated Critical Position – 1% $\Delta\rho$. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03138.

- Example 5: Operating Instruction OI-SI-1, “Safety Injection – Normal Operation,” Attachment 4, “Filling SI Tank(s) Using HPSI Pumps,” does not contain sufficient guidance for operators to successfully fill the safety injection tank using high pressure safety injection pumps. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03139.

- Example 6: The licensee failed to include directions in Alarm Response Procedure ARP-CB-10,11/A12, to set the 43FW switch to OFF prior to attempting a manual start of the standby condensate pump when the auto-start feature fails to start the standby pump. This switch must be placed in OFF before the standby condensate pump can be started. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

- Example 7: The licensee failed to include direction to start an auxiliary lube oil pump prior to attempting to start the main feedwater pump in Abnormal Operating Procedure AOP-28, “Auxiliary Feedwater Malfunctions.” After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03973.

These failures to prescribe activities affecting quality by procedures or to include appropriate acceptance criteria are performance deficiencies. Each example is more than minor and is therefore a finding because it adversely affects the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. These examples either could have significantly affected, or were shown during examination preparation and performance to have actually affected the operator’s ability to perform the activity affecting quality. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a Phase 1 screening was performed and determined that each example was of very low safety significance (Green) because each example: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. These findings have a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)](Section 4OA5).

- Severity Level IV. The team identified a Severity Level IV non-cited violation of Title 10 CFR Part 50.71(e), for failure to periodically update the final safety analysis report originally submitted as part of the application for the license to assure that the information included in the report contains the latest information developed. Specifically, the licensee failed to update the final safety analysis report to include the required continuous hour rating used for emergency diesel generator testing as referenced by Technical Specification Surveillance Requirement 3.7(1)(a)(ii). After identification, the licensee entered this issue in the corrective action program as Condition Report 2011-06612.

  The failure to update the final safety analysis report to include information that is specifically referenced by the technical specifications is a performance deficiency. The inspectors considered this issue to be within the traditional enforcement process because it has the potential to impede or impact the NRC's ability to perform its regulatory function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. The violation is more than minor because the omitted final safety analysis report update information had a potential impact on safety and licensed activities in that the licensee did not have the required reference for a surveillance test. Consistent with the guidance in Section 2.2.2 and Section 6.1.d.3 of the NRC Enforcement Policy, the inspectors concluded that the violation is a Severity Level IV because the omitted information did not result in an unacceptable change to the facility or procedures (Section 4OA5).

- Green. The team identified a non-cited violation for failing to comply with Technical Specification 2.3(1)(i) in that multiple Safety Injection Tanks were connected together simultaneously for filling operations on at least two occasions, once while sluicing on 01/18/2010 and once where all four tanks were connected together on 03/31/2011. This Limiting Condition for Operation requires that all valves, piping and interlocks associated with the Safety Injection Tanks (that are required to function during accident conditions) are operable to maintain Safety Injection Tank operability. Operability of these fill valves is met when each valve is shut. With multiple fill valves open during normal operations, this technical specification is not met, and there is no remedial action described when more than one Safety Injection Tank is inoperable with the reactor critical, requiring a unit shutdown in accordance with Technical Specification 2.0.1. This action was not performed by the licensee. After identification, the licensee entered this issue into the corrective action program as Condition Reports 2012-01956 and 2012-04815.

  Failure to comply with technical specifications was a performance deficiency. The performance deficiency is more than minor and therefore a finding because it adversely impacted the equipment performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process Phase 1 and 2 Worksheets, the finding was determined to affect the loss of system safety function and required entry into Appendix A of this process for screening. The senior reactor analyst screened the issue based on a less than one-hour exposure time and determined that the finding was of very low safety significance

(Green) because the calculated bounding delta core damage frequency was 1 E-8. The finding has a cross-cutting aspect in the area of work control because the licensee failed to plan work activities to support long-term equipment reliability by not limiting safety system unavailability, specifically the Safety Injection Tanks [H.3(b)](Section 4OA5).

Cornerstone: Initiating Events

- Green. The team identified a finding of very low safety significance involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," with five examples.

    - Example 1: Alarm Response Procedure ARP-CB-10,11/A12 for a main feedwater pump trip does not provide guidance that the auxiliary lube oil pump must be started prior to starting the main feedwater pump. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

    - Example 2: Alarm Response Procedure ARP-CB-1,2,3/A2 provides inadequate instructions for restoration of letdown following a controller or instrument failure that causes letdown isolation. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

    - Example 3: Alarm Response Procedure ARP-AI-66A/A66A does not contain guidance to determine if an auxiliary feedwater actuation is inadvertent nor does it contain guidance to enter AOP-28, "Auxiliary Feedwater System Malfunctions," if the operators determine that the actuation is inadvertent. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

    - Example 4: Alarm Response Procedure ARP-CB-1,2,3/A1 does not contain guidance for entering AOP-35, "Reactor Coolant Pump Malfunctions," when there is a seal cooler leak. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

    - Example 5: Alarm Response Procedure ARP-CB-1,2,3/A2 does not contain any procedural guidance for a failure of the VCT level instrument. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03140.

  These failures to prescribe activities affecting quality by procedures or to include appropriate acceptance criteria are performance deficiencies. Each example is more than minor and therefore a finding because it adversely affects the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. These examples either could have significantly affected, or were shown during examination preparation and administration to have actually affected the operator's ability to perform the activity affecting quality. In accordance with

Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and each example except for Example 1 was determined to be of very low safety significance (Green) because each example does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. For Example 1, a phase 1 screening was performed and the finding was determined to contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available and required a phase 3 analysis. A senior reactor analyst determined that the finding was of very low safety significance because the calculated bounding delta core damage frequency was 1.4 E-7. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)](Section 4OA5).

Cornerstone: Barrier Integrity

- Green. The team identified a finding of very low safety significance involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that, design control measures did not assure that the design basis for safety related systems were correctly translated into procedures. Specifically, the licensee failed to correctly translate the design basis of the containment spray system into Technical Basis Document Procedure TBD-EOP-05, "Uncontrolled Heat Extraction." After identification, the licensee entered this issue in the corrective action program as Condition Report 2011-06802.

  The failure to correctly translate the design basis of the containment spray system into an emergency operating procedure technical basis document is a performance deficiency. It is more than minor and therefore a finding because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The incorrect guidance in the emergency operating procedure basis document could result in a licensed operator taking incorrect action to secure containment spray based on a faulty understanding of the expected system response. Securing containment spray during a main steam line break would challenge the safety function of the containment building, increasing the risk to public health and safety. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding: (1) did not represent only a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool; (2) did not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; (3) did not represent an actual open pathway in the physical integrity of reactor containment system, containment isolation system, and heat removal components; and (4) did not involve an actual reduction in function of hydrogen ignitors in the reactor containment. There was no cross-cutting aspect assigned to this performance deficiency because it was not indicative of current plant performance (Section 4OA5).

- Green. The team identified a finding of very low safety significance involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," with four examples.

    - Example 1. The normal operating instruction for reactor coolant pumps, OI-RC-9, "Reactor Coolant Pump Operation," contains pump trip requirements that conflict with the pump trip requirements provided in the Abnormal Operating Procedure AOP-35, "Reactor Coolant Pump Malfunctions." After identification, the licensee entered this issue in the corrective action program as Condition Report 2012-03145.

    - Example 2: Annunciator Response Procedure ARP-DCS-SCEAPIS incorrectly directs the operators to restore a control element assembly group to within proper overlap using manual group mode, instead of manual individual mode. After identification, the licensee entered the issue into the corrective action program as Condition Report 2011-07172.

    - Example 3: Neither the Annunciator Response Procedure ARP-DCS-SCEAPIS, nor the control element assembly Abnormal Operating Procedure AOP-02, "CEA and Control System Malfunctions," address excessive overlap between control element assembly groups caused by operator error instead of a digital control system failure. After identification, the licensee entered the issue into the corrective action program as Condition Report 2011-09653.

    - Example 4: The licensee's Abnormal Operating Procedure AOP-21, "Reactor Coolant System High Activity," has multiple values for high reactor coolant system activity requirements that conflict on whether or not it is necessary to initiate a plant shutdown. Additionally, this procedure is not current with the most recent action levels contained in SO-O-43 "Fuel Reliability Management Plan." This fuel reliability management plan currently lists four action levels, while the actions in the abnormal operating procedure are based on five action levels. The fifth action level actions would not be performed since no fifth action level is defined in SO-O-43. After identification, the licensee entered the issue into the corrective action program as Condition Report 2012-03143.

  These failures to prescribe activities affecting quality by procedures or to include the appropriate acceptance criteria are performance deficiencies. Each example is more than minor and is therefore a finding because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. These examples either could have significantly affected, or were shown during examination preparation and administration to have actually affected the operator's ability to perform the activity affecting quality. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a phase 1 screening was performed and each example except for Example 1 was determined to be of very low safety significance (Green) because the fuel cladding barrier was affected but did not affect the reactor coolant system or containment barriers. Example 1 was determined to be of very low safety significance (Green) because the finding: (1) did not represent only a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool; (2) did not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; (3) did not represent an actual open pathway in the physical integrity of reactor containment system, containment isolation system, and heat removal components; and (4) did not involve an actual reduction in function of hydrogen ignitors in the reactor containment. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)](Section 4OA5).

B. Licensee-Identified Violations

None.

# REPORT DETAILS

## 4. OTHER ACTIVITIES (OA)

### 4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

NRC examiners reviewed all license applications submitted to ensure each applicant satisfied relevant license eligibility requirements. The examiners also audited three of the license applications in detail to confirm that they accurately reflected the subject applicant's qualifications. This audit focused on the applicant's experience and on-the-job training, including control manipulations that provided significant reactivity changes.

b. Findings

No findings were identified. However, one issue was noted as a performance deficiency and was turned-over to the resident inspector staff to complete formal inspection activities. This issue involved the lack of control room staff logging reactor power changes in the control room as required by the "Conduct of Operations" SO-O-1 procedure. During the application review the examiners attempted to verify the in-plant control manipulation credits documented on the NRC 398 forms were completed as described on the form and could not determine any information from the control room logs because these power changes were not logged. Additionally, there were no log entries that an under-instruction watch-stander was actually in control of the reactor for any of these reactivity credits for every applicant that took this examination and required these credits to apply for a license. The only way that the examiners were able to verify the actions in the control room against the credits documented on the applications was to verify the items within the qualification cards and the signatures that the shift managers provided on the qualification cards for each of the individuals that they observed on shift during the respective evolutions. The NRC determined that this was acceptable for the credits but that control room log-taking was an issue at the station. The licensee wrote a Condition Report 2012-02186 for this issue.

A second issue that was identified was that one individual did not meet the intent in NRC guidance documents for crediting a discrete down-power, because the licensee documented two credits on an individual's application during a power change from approximately 41 percent to 30 percent while he was on the turbine controls as the Balance of Plant Operator under instruction. This was effectively one credit for this individual and therefore required the individual to perform another reactivity credit, which he completed on the simulator and the corresponding NRC form 398 was edited to reflect this change. This has been discussed with industry and is documented in Information Notice IN-1997-67. The licensee wrote a Condition Report 2012-02186 for this issue.

## .2 Examination Development

a. Scope

The NRC developed the written examination and the operating test in accordance with the requirements of NUREG-1021. The NRC examination team conducted an onsite validation of the operating tests.

b. Findings

**Inadequate Procedures with Seven Examples for the Mitigating Systems Cornerstone**

The team identified a finding of very low safety significance (Green) involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," with seven examples.

Example 1: No Instructions for Tripping Bistables on a Trip Unit for Any Instrument Failure Other Than Nuclear Instrumentation

Introduction. The team identified one of seven examples of a Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's alarm response procedure for the reactor protection system trips and pretrips only directs operators to bypass the affected channel for that trip unit in the event of an instrument failure. If another channel is already bypassed, the technical specifications require placing one of the channels in trip.

Description. During examination preparation, the examiners drafted a scenario with an initial condition of the thermal margin / low pressure (TM/LP) trip unit for channel A in bypass. The scenario then has a failure of the channel B reactor coolant system cold leg temperature. The reactor coolant system cold leg temperature is an input into the TM/LP trip unit, and the failure of this instrument causes an annunciator to alarm on control panel CB-4/A20. The annunciator response procedure for this alarm is ARP-CB-4/A20, Revision 43, and includes actions to determine if the alarm is due to an instrument failure, and if that is the case, to bypass the affected channel of the TM/LP trip unit. In the case of the scenario being developed, one channel of the TM/LP trip unit was already in bypass, and per Technical Specification 2.15, channel B for the TM/LP trip unit must be placed in trip within one hour. During the examination pre-validation visit, the instructors did not know how to place this trip unit into trip using ARP-CB-4/A20. The instructors were eventually able to place the B channel of the TM/LP trip unit in trip by using guidance that is contained in Abnormal Operating Procedure AOP-15, "Loss of Flux Indication or Flow Streaming," which is entered if there is a malfunction or failure of a nuclear instrument.

Analysis. The failure to include information in the annunciator response procedure to trip an affected channel trip unit if another channel is already in bypass is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example could have significantly affected the operator's ability to perform the activity affecting quality, in this case, complying with the technical specification requirement to trip the failed trip unit channel. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include procedural guidance in the reactor protective system annunciator response procedure for tripping the trip unit channel if required. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the first of seven examples.

Example 2: No Procedural Criteria for Tripping Only Required Trip Units and Bypassing the Other Affected Trip Units

Introduction. The team identified a second example of the Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically the licensee failed to include procedural criteria for tripping only the required trip units and bypassing the other affected trip units in the case of a nuclear instrument failure concurrent with another instrument failure.

Description. During the initial license examination operating test development, the examiners created a scenario in which an initial condition was that a pressurizer pressure instrument was undergoing troubleshooting, placing the high pressurizer pressure and the thermal margin/low pressurizer pressure (TM/LP) trip units in bypass for that channel. A malfunction was then entered that failed a power range nuclear instrument on a different channel. Power range nuclear instruments are an input into the TM/LP trip unit and would require tripping that channel TM/LP trip unit. During the examination validation week, this scenario was run with these initial conditions and malfunctions and the validation crew struggled with their response to the nuclear instrumentation failure. The crew appropriately entered the Annunciator Response Procedure ARP-CB-4/A20 for the failure, which directed them to enter Abnormal Operating Procedure AOP-15, "Loss of Flux Indication or Flow Streaming," Revision 12. Step 1 states, "If ONE Power Range Safety Channel has failed or operates erratically, THEN disable the affected RPS Trip Units within one hour of failure by performing either step a or b." Step 1.a states, "Place ALL of the following RPS Trip Units on the inoperable channel in bypass within one hour: TU-1, TU-2, TU-9, TU-10, TU-12." Step 1.b states, "Place the affected RPS Trip Units on the inoperable channel in trip within one hour" by implementing the power trip test interlock and removing trip units TU-2 and TU-10. In the scenario developed by the examiners, per Technical Specification 2.15, the TM/LP trip unit (TU-9) must be tripped but the other trip units (TU-1, TU-2, TU-10, and TU-12) can be bypassed, which is desired to minimize the chances of a plant transient. By bypassing the trip units, it results in a reactor trip logic of 2 out of 3 remaining channels. If the trip units are tripped, the reactor trip logic is 1 out of 3, and the likelihood of a reactor trip is higher. However, the procedure as written does not allow the operators to just trip the TM/LP trip unit and bypass the other four affected trip units. The procedure as written would require the operators to trip all five affected trip units, increasing the likelihood of a reactor trip. During the examination validation of this scenario, the validation crew struggled with the appropriate response to the instrument failures and decided to deviate from the procedure as written in order to trip the TM/LP trip unit and to bypass the other four affected trip units, which is not specifically allowed by procedure, but is in accordance with the technical specifications. Therefore, the fact that this procedure does not include guidance on what actions to take for this situation significantly impacted the operator's ability to complete the task of tripping the TM/LP trip unit and bypassing the other four affected trip units. The licensee has entered this issue into their corrective action program as Condition Report 2012-03141.

Analysis. The failure to include criteria for tripping only the required trip units and bypassing the other affected trip units in the case of a nuclear instrument failure concurrent with another instrument failure is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example could have significantly affected the operator's ability to perform the activity affecting quality, in this case, complying with technical specification requirements to trip the required trip unit and bypass the other trip units. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria for tripping only the required trip units and bypassing the other affected trip units in the case of a nuclear instrument failure concurrent with another instrument failure. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03141, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the second of seven examples.

Example 3: Inadequate Procedure Use and Adherence Procedure

Introduction. The team identified a third example of the Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee failed to include criteria for implementation of procedure use and adherence associated with emergency operating procedures and abnormal operating procedures.

Description. The concept of procedure use and adherence associated with emergency operations procedures is mentioned in each of two procedures: SO-O-1, "Conduct of Operations," Revision 84, and OPD-04-09, "EOP/AOP Procedure Use and Adherence," Revision 12. Each of these procedures directs the operator to the other procedure for implementation of procedure use and adherence for emergency operating procedures and abnormal operating procedures. Neither of these documents contains the necessary guidance for procedure use and adherence. The result is that the operators routinely deviate from procedures, including both the abnormal operating procedures and the emergency operating procedures. Without clear guidance, the operators have difficulty knowing when/how/if they can work steps out of sequence or deviate from the procedures. The licensee has entered this issue into their corrective action program as Condition Report 2012-3143.

Analysis. The failure to include criteria for implementation of procedure use and adherence associated with emergency operating procedures and abnormal operating procedures is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example significantly affected the operator's ability to perform the activity affecting quality; in this case, during the examination validation week, the licensed operators routinely deviated from procedures, including normal operating instructions, annunciator response procedures, and emergency operating procedures. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include clear criteria for implementation of procedure use and adherence for emergency operating procedures. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03143, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the third of seven examples.

Example 4: No Procedural Guidance on Plotting 1/M Data Against Reactivity and Control Element Assembly Position

Introduction. The team identified a fourth example of the Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee failed to include criteria for plotting 1/M data against reactivity and the control element assembly position.

Description. During initial license examination operating test development, the examiners randomly selected performance of a 1/M plot for an administrative job performance measure. Although the shift technical advisors typically create the 1/M plots, it is listed as a task in the licensed operator job task list. When this was administered as an administrative job performance measure during the examination validation week, the licensed operator was unable to complete the task with the procedural guidance as written. Operating procedure OP-2A, "Plant Startup," Revision 102, Step 10.b states, "commence 1/M plots per OP-2A Attachment 2A." OP-2A, Attachment 2A provides instructions regarding calculation of the count rate. Step 4 of Attachment 2A states to calculate and record 1/M (Co/Ci), then plot the value for each channel. The first note in the attachment states that 1/M data shall be plotted against reactivity and control element assembly position but does not specify how to plot the data. In order to plot the 1/M data against reactivity using the control element assembly position, the operator would need to pick the appropriate figure from Figures II.B.2.a-f, depending on the current core burnup, in Technical Data Book TDB-II, "Reactivity Curves." The operator then must use this figure to convert from the control element assembly rod position to the sequential rod worth in %Δρ. The operator must then use this information to plot the data on the criticality 1/M plot, in which the calculated 1/M value is used as the y-axis coordinate and the sequential rod worth is used as the x-axis coordinate. The operating procedure OP-2A does not reference Technical Data Book TDB-II figures, nor does it explain how to convert from the control element assembly position to the sequential rod worth. The procedure also does not specify how to plot the 1/M and sequential rod worth data, specifically which data is plotted on the x-axis and which is plotted on the y-axis. Because the licensed operator was unable to complete the task during the examination validation week, the 1/M plot job performance measure was removed from the examination to allow the licensee to address the issue. The licensee entered this issue into their corrective action program as Condition Report 2012-03138.

Analysis. The failure to include criteria for plotting 1/M data against reactivity and the control element assembly position is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example significantly affected the operator's ability to perform the activity affecting quality; in this case, during the examination validation week, the licensed operator was unable to complete a 1/M plot for determining estimated critical position using the procedure as written. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria for plotting 1/M data against reactivity and the control element assembly position. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03138, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the fourth of seven examples.

Example 5: Operating Instruction OI-SI-1 Does Not Contain Sufficient Guidance for Operators to Successfully Accomplish Task

Introduction. The team identified a fifth example of the Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's safety injection tank fill procedure did not include criteria specifying which safety injection tank loop injection valves were associated with which safety injection tank leakage coolers.

Description. During initial license examination operating test preparation, the examiners randomly selected a safety injection tank low level alarm as a malfunction for one of the scenarios. The expected operator response was to refill the safety injection tank using the high pressure safety injection (HPSI) pumps per operating instruction OI-SI-1, "Safety Injection – Normal Operation," Revision 129, Attachment 4, "Filling SI Tank(s) Using HPSI Pumps." The procedure includes direction to close the selected leakage cooler discharge valve, either PCV-2929 or PCV-2969. The next step is to crack open, then fully open the loop injection valve for each safety injection tank line to be used, either HCV-314, Loop 1A HPSI Injection Valve, HCV-315, Loop 1A HPSI Injection Valve, HCV-320, Loop 2B HPSI Injection Valve, or HCV-321, Loop 2B HPSI Injection Valve. However, the procedure does not provide guidance on which loop injection valves are associated with which leakage coolers. During the examination validation week, when the safety injection tank SI-4B low level alarm was initiated, the licensed operators had to trace the safety injection line piping on the piping and instrumentation drawing to determine which valves were associated with which leakage coolers. The event was terminated after approximately one hour, when no progress had been made in operating the system. Therefore, the lack of this procedural guidance significantly impacted the operator's ability to perform the task. If safety injection tank level falls below 67%, Technical Specification 2.3(2)(f) applies, requiring restoration of the safety injection tank to operable status within twenty-four hours. Due to the inability of the licensed operator's to perform this task, the malfunction was removed from the examination to allow the licensee to address the issue. The licensee entered this issue into their corrective action program as Condition Report 2012-03139.

Analysis. The failure to include criteria specifying which safety injection tank loop injection valves were associated with which safety injection tank leakage coolers is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example significantly affected the operator's ability to perform the activity affecting quality; in this case, during the examination validation week, the licensed operators were unable to fill the safety injection tank using the procedure as written. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria specifying which loop injection valves were associated with which safety injection tank leakage cooler. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03139, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the fifth of seven examples.

Example 6: Procedure Does Not Address 43/FW Switch Placement For Failure of Standby Condensate Pump to Auto-Start

Introduction. The team identified a sixth example of the Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's alarm response procedure for a condensate pump trip with a failure of the standby pump to auto-start does not include the criteria to ensure the 43/FW switch is placed in OFF prior to attempting to start the standby condensate pump.

Description. During examination administration, one scenario had a malfunction where one condensate pump trips and the standby condensate pump fails to auto-start. During the performance of this scenario, the examiners observed the crew discuss the need the place the 43/FW switch to OFF prior to attempting to start the standby condensate pump. When the examiners asked the licensed operator candidates why they placed that switch in OFF prior to attempting to start the standby condensate pump, the candidates replied that the switch must be placed in OFF before attempting to start the condensate pump or the condensate pump will not start. The examiners then reviewed the annunciator response procedure for a condensate pump trip, ARP-CB-10,11/A12, Revision 12. Step 2 of ARP-CB-10,11/A12 states, "if FW-2A is tripped, then ensure that standby Condensate Pump FW-2B or C has automatically started." However, the procedure does not address the need to place the 43/FW switch to OFF before starting the standby condensate pump if the pump failed to auto-start. This switch is also tied to the auxiliary feedwater system and its standby start features also are wired through the 43/FW switch (except for engineered safeguards actuations). Consequently, this switch is a quality component and Appendix B applies to it as well as the procedure. The licensee wrote Condition Report 2012-03140 to address this issue.

Analysis. The failure to include criteria to ensure the 43/FW switch is placed in OFF prior to attempting to start the standby condensate pump for a condensate pump trip with a failure of the standby condensate pump is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example could have significantly affected the operator's ability to perform the activity affecting quality, in this case, starting the standby condensate pump with the 43 FW switch in the OFF position. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a Phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria to ensure the 43/FW switch is placed in OFF prior to attempting to start the standby condensate pump for a condensate pump trip with a failure of the standby condensate pump. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the sixth of seven examples.

Example 7: No Procedural Guidance for Starting Auxiliary Lube Oil Pump in AOP-28, "Auxiliary Feedwater System Malfunctions"

Introduction. The team identified a seventh example of the Green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawing." Specifically, the licensee failed to include the criteria to ensure the auxiliary lube oil pump was started prior to attempting to start a main feedwater pump for a loss of auxiliary feedwater event.

Description. During examination administration, a job performance measure was administered in which the initial conditions were that the plant was shutdown and auxiliary feedwater had just been lost. Per Abnormal Operating Procedure AOP-28, "Auxiliary Feedwater System Malfunctions," the candidate is supposed to attempt to start the turbine driven auxiliary feedwater pump, which fails, and is then supposed to attempt to start one main feedwater pump per Step 10. Step 10 does not contain actions to start the auxiliary lube oil pump prior to starting the main feedwater pump. During the examination administration, eight out of ten candidates were unable to start main feedwater pump FW-4B because they failed to start the auxiliary lube oil pump first. The licensee wrote Condition Report 2012-03973 to address this issue.

Analysis. The failure to include criteria to ensure the auxiliary lube oil pump was started prior to attempting to start a main feedwater pump for a loss of auxiliary feedwater event is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it is associated with the procedure quality attribute of the mitigating systems cornerstone and affects the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This example could have significantly affected the operator's ability to perform the activity affecting quality, in this case, starting the main feedwater pump. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the finding: (1) is not a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to May 11, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria to ensure the auxiliary lube oil pump was started prior to attempting to start the main feedwater pump for a loss of auxiliary feedwater event. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03973, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-01, "Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone." This was the seventh of seven examples.

**Failure to Periodically Update the Final Safety Analysis Report for the Emergency Diesel Generators**

Introduction. The team identified a Severity Level IV non-cited violation of Title 10 CFR Part 50.71(e), for failure to periodically update the final safety analysis report originally submitted as part of the application for the license to assure that the information included in the report contains the latest information developed. Specifically, the licensee failed to update the final safety analysis report to include the required continuous hour rating used for emergency diesel generator testing as referenced by Technical Specification Surveillance Requirement 3.7(1)(a)(ii).

Description. During the initial license written examination preparation, the examiner randomly selected a topic concerning the ability to monitor the effect of switching power supplies on instrumentation and controls to prevent exceeding design limits of the ac distribution system. During the examiner's review of the licensee provided reference material, the examiner reviewed the technical specification surveillance requirement associated with the emergency diesel generators. Technical Specification Surveillance Requirement 3.7(1)(a)(ii) states, in part, "The generator shall then be loaded to at least the continuous(2) KW rating and run for at least 60 minutes before being off-loaded and the diesel breaker tripped." This technical specification page is from Amendment 251 dated May 24, 2008. Reference (2) for this surveillance requirement is Updated Safety

Analysis Report (USAR) Section 8.4.1. Revision 13 for USAR 8.4.1, issued September 9, 2010, discusses the 2000 hour HP/KW rating and de-rating factors but does not specify what the continuous KW rating is for either diesel generator DG-1 or DG-2 as referenced by the technical specification. Both surveillance test procedures OP-ST-DG-0001 and 0002 for each emergency diesel generator contain a figure that provides a continuous hour rating based on the ambient temperature. The licensee determined that this continuous hour rating is based on a 1980 vendor letter which provides a specific continuous hour rating that was then de-rated using the USAR described de-rating factors. Therefore, the licensee determined that although the continuous hour rating is not specified in the USAR as referenced by the technical specification surveillance requirement, the surveillance test procedures are testing the diesel generators to the continuous hour rating to meet the technical specification surveillance requirement and the omission of this continuous hour rating did not result in an unacceptable change to facility procedures. The licensee wrote Condition Report 2011-06612 to address this issue.

Analysis. The failure to update the final safety analysis report to include information that is specifically referenced by the technical specifications is a performance deficiency. The inspectors considered this issue to be within the traditional enforcement process because it has the potential to impede or impact the NRC's ability to perform its regulatory function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. The violation is more than minor because the omitted final safety analysis report update information had a potential impact on safety and licensed activities in that the licensee did not have the required reference for a surveillance test. Consistent with the guidance in Section 2.2.2 and Section 6.1.d.3 of the NRC Enforcement Policy, the inspectors concluded that the violation is a Severity Level IV because the omitted information did not result in an unacceptable change to the facility or procedures.

Enforcement. Title 10 CFR 50.71(e) requires, in part, that the licensee shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed and that the revisions must reflect all changes up to a maximum of 6 months prior to the date of filing. Contrary to the above, the licensee failed to periodically update the final safety analysis report originally submitted as part of the application for the license to assure that the information included in the report contains the latest information developed. Specifically, on September 9, 2010, the licensee failed to update the final safety analysis report to include the required continuous hour rating used for emergency diesel generator testing as referenced by Technical Specification Surveillance Requirement 3.7(1)(a)(ii), dated May 24, 2008. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2011-06612, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy:
NCV 05000285/2012301-02, “Failure to Periodically Update the Final Safety Analysis Report for the Emergency Diesel Generators.”

**Failure to Comply with Technical Specification 2.3(1)(i) for Safety Injection Tank Operability**

Introduction. The team identified a Green non-cited violation of Technical Specification 2.3 (1)(i) for cross connecting Safety Injection Tanks during filling operations. This evolution made more than one Safety Injection Tank inoperable while the reactor was critical and requires entry into Technical Specification 2.01 and subsequent reactor shutdown within 6 hours unless all Safety Injection Tanks are returned to operable.

Description. During the initial operating test validation, the team discovered that OI-SI-1, "Operating Procedure Safety Injection," R129, describes directions for both sluicing (cross-connecting to equalize level) and filling more than one Safety Injection Tank (SIT) at the same time using the High Pressure Safety Injection (HPSI) Pumps. The team discovered that these evolutions have been performed at least two times in the plant. On 1/18/2010, the licensee cross-connected SIT SI-6A to SIT SI-6D to sluice from SIT 6D to SIT 6A. On 3/31/11 the licensee filled all four SIT's at the same time using the High Pressure Safety Injection Pumps. No more than one SIT is allowed to be inoperable per Technical Specification 2.3(1) while the reactor is critical. The licensee wrote Condition Reports 2012-01956 and 2012-04815 to address this issue.

Analysis. Cross-connecting Safety Injection Tanks while the reactor is critical is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it affects the mitigation system cornerstone, in that short term heat removal is degraded. Using the Significance Determination Process Phase 1 and 2 Worksheets, the finding was determined to affect the loss of system safety function and required entry into Appendix A of this process for screening. The senior reactor analyst screened the issue based on a less than one-hour exposure time and determined the risk to be 1 E-8, therefore this finding was determined to be of very low safety significance (Green). The finding has a cross-cutting aspect in the area of work control because the licensee failed to plan work activities to support long-term equipment reliability by not limiting safety system unavailability, specifically the Safety Injection Tanks [H.3(b)].

Enforcement. Technical Specification 2.3(1) c. requires four Safety Injection Tanks (SIT's) operable when the reactor is critical. Technical Specification 2.3(1) i. requires all valves, piping and interlocks associated with the SIT's and required to function during accident conditions be operable. Operability of the fill valves is met when each valve is shut. With multiple fill valves open when the reactor is critical. Contrary to the above, the licensee cross-connected SIT''s on at least two occasions in violation of this requirement. Specifically, on 1/18/2010, the licensee cross-connected SIT SI-6A to SIT SI-6D to sluice from SIT 6D to SIT 6A. On 3/31/11 the licensee connected all four SIT's together at the same time and filled them with water using the High Pressure Safety Injection Pumps. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Reports 2012-01956 and 2012-04815, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-03, "Failure to Comply with Technical Specification 2.3(1)(i) for Safety Injection Tank Operability."

**Inadequate Procedures with Five Examples for the Initiating Events Cornerstone**

The team identified a finding of very low safety significance (Green) involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” with five examples.

Example 1: No Annunciator Response Procedural Guidance for Starting an Auxiliary Lube Oil Pump

Introduction. The team identified the first of five examples of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” Specifically, the licensee’s annunciator response procedure was inadequate for a main feedwater pump trip because it does not include criteria for ensuring the auxiliary lube oil pump is started prior to attempting to start the main feedwater pump.

Description. During preparation of the initial license examination operating test, the examiners randomly selected a main feedwater pump trip with a failure of the standby pump to auto-start as a malfunction for one of the scenarios. During the examination pre-validation visit, when this malfunction, occurred, the instructors attempted to start the standby pump without first starting its auxiliary lube oil pump, resulting in the failure of the standby pump to start. The examiners then reviewed the annunciator response procedure for a main feedwater pump trip, ARP-CB-10,11/A12, Revision 11. The annunciator response procedure states that if the main feedwater pump is tripped, ensure the standby feedwater pump has automatically started and that if two pumps are required, ensure that the other two feedwater pumps are running. The annunciator response procedure does not provide guidance that if the standby feedwater pump failed to auto-start, the auxiliary lube oil pump must be started before an attempt can be made to start the standby main feedwater pump. If the standby main feedwater pump cannot be started, the next action directed by the annunciator response procedure is to trip the reactor. This failure to include the information to start the auxiliary lube oil pump prior to attempting to start the standby main feedwater pump increases the risk of a plant transient.

Analysis. The failure to include criteria to ensure the auxiliary lube oil pump is started prior to attempting to start the main feedwater pump is a performance deficiency. It is more than minor because it adversely affects the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This failure could result in the operators being unable to start the standby main feedwater pump, resulting in an unnecessary plant transient. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a phase 1 screening was performed and the finding was determined to contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available and required a phase 3 analysis. A senior reactor analyst determined that the finding was of very low safety significance because the calculated bounding delta core damage frequency was 1.4 E-7. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria for ensuring the auxiliary lube oil pump was running prior to attempting to start the main feedwater pump. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-04, "Five Examples of Inadequate Procedures for the Initiating Events Cornerstone." This was the first of five examples.

Example 2: Lack of Criteria for Ensuring Instrument Failure Is Fixed Prior to Restoring Letdown Isolation Valve Interlock

Introduction. The team identified the second of five examples of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The licensee failed to include criteria to ensure that an instrument or controller failure is corrected prior to restoring the letdown isolation valve interlock in the alarm response procedure for a letdown isolation.

Description. During preparation of the initial license examination operating test, the examiners randomly selected a failure of a pressure controller for letdown pressure downstream of the backpressure regulator. This pressure controller is an input into the interlock for the letdown isolation valve. The malfunction causes the pressure controller to fail high, which results in an isolation of letdown. Once the letdown isolation valve interlocks are defeated, letdown can be restored, but the interlocks must remain defeated until the controller is repaired. Otherwise, letdown will re-isolate. Annunciator Response Procedure ARP-CB-1,2,3/A2, Revision 33, provides instructions to the operators to defeat the letdown isolation valve interlocks and then to restore letdown per OI-CH-1, "Chemical and Volume Control System Normal Operation." The annunciator response procedure does not provide any instructions regarding the need to maintain the interlock defeated until such time as the controller is repaired. During letdown restoration per OI-CH-1, the procedure directs the operators to place the interlock from DEFEAT back to NORMAL, reinstating the interlocks and resulting in a re-isolation of letdown since the failure has yet to be repaired. During examination validation, the operating crew recognized this problem and performed a procedure deviation to omit performance of the step reinstating the interlocks until the failure could be repaired. This failure to include guidance regarding the need to wait to reinstate the interlocks until after the failure is repaired increases the risk of another letdown isolation unless the operator deviates from the procedure as written and could significantly impact the operators ability to successfully restore letdown.

Analysis. The failure to include criteria to ensure that an instrument or controller failure is corrected prior to restoring the letdown isolation valve interlock is a performance deficiency. It is more than minor because it adversely affects the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This failure could result in a re-isolation of letdown flow. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria for ensuring that an instrument failure is corrected prior to restoring the letdown isolation valve interlocks. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-04, "Five Examples of Inadequate Procedures for the Initiating Events Cornerstone." This was the second of five examples.

Example 3: No criteria in Alarm Response Procedure for Determining if Auxiliary Feedwater Actuation is Inadvertent

Introduction. The team identified a third of five examples of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The licensee failed to include appropriate criteria in the annunciator response procedure to determine whether or not the auxiliary feedwater actuation is inadvertent.

Description. During preparation of the initial license examination operating test, the examiners randomly selected an inadvertent auxiliary feedwater actuation while the reactor is at power as one of the events in a scenario. During review of the licensee provided reference material, the examiners discovered that Annunciator Response Procedure ARP-AI-66A/A66A, Revision 13, does not contain guidance to determine if the auxiliary feedwater actuation is inadvertent, nor does it contain guidance to enter Abnormal Operating Procedure AOP-28, "Auxiliary Feedwater System Malfunctions," Revision 15, if the operators determine that the actuation is inadvertent. Specifically, if an inadvertent auxiliary feedwater actuation occurs, the operators need to enter AOP-28 for instruction in recovering from the actuation. Abnormal Operating Procedure AOP-23, "Reset of Engineered Safeguards" provides instructions for recovering from an inadvertent actuation of engineered safeguards, except for an inadvertent auxiliary feedwater actuation which is contained in AOP-28. This failure to direct the operators to determine if the auxiliary feedwater actuation is inadvertent and to direct entry into AOP-28 if it is determined to be inadvertent could significantly impact the operator's ability to recover from an inadvertent auxiliary feedwater event, including securing from the additional feedwater flow to the steam generators, which could result in a turbine trip on overspeed and subsequent reactor trip, and restoring auxiliary feedwater to service following termination of the additional feedwater flow.

Analysis. The failure to include criteria to ensure that an instrument failure is corrected prior to restoring the letdown isolation valve interlock is a performance deficiency. It is more than minor because it adversely affects the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This failure could result in overfilling the steam generators and causing a plant transient. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria in the alarm response procedure to determine whether or not the auxiliary feedwater actuation was inadvertent. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement

Policy: NCV 05000285/2012301-04, “Five Examples of Inadequate Procedures for the Initiating Events Cornerstone.” This was the third of five examples.

Example 4: No Criteria for Entering Reactor Coolant Pump Abnormal Operating Procedure for a Reactor Coolant Pump Seal Cooler Leak

Introduction. The team identified a fourth of five examples of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The licensee failed to include criteria in the annunciator response procedure to enter the reactor coolant pump abnormal operating procedure when evidence of a reactor coolant pump seal cooler leak exists.

Description. During preparation of the initial license examination operating test, the examiners randomly selected an intersystem loss of coolant accident as the major transient in one of the scenarios. The examiners chose to use a reactor coolant pump seal cooler leak into the component cooling water system as the intersystem loss of coolant accident. During a reactor coolant pump seal cooler leak, the reactor coolant pump’s component cooling water seal cooler high temperature annunciator will be the first to alarm. The actions for this annunciator, per Annunciator Response Procedure ARP-CB-1,2,3/A1, Revision 33, include increasing component cooling water flow and if the flow cannot be increased, to enter Abnormal Operating Procedure AOP-11, “Loss of Component Cooling Water,” for a loss of component cooling water flow. Neither of these actions would address the actual transient. Abnormal Operating Procedure AOP-35, “Reactor Coolant Pump Malfunctions,” Revision 5, provides the instruction to trip the reactor and trip the affected reactor coolant pump if the reactor coolant pump lower seal cavity temperature exceeds 200$^{o}$F. The annunciator response procedure does not provide guidance to enter AOP-35, nor does it proved instruction to check other reactor coolant pump temperatures, such as the lower seal cavity temperature, that would require stopping the pump. This failure to include criteria in the annunciator response procedure to check other reactor coolant pump temperatures or to enter AOP-35 could result in a delay by the operators in tripping the reactor and stopping the affected reactor coolant pumps.

Analysis. The failure to include appropriate criteria in the annunciator response procedure to enter the reactor coolant pumps abnormal operating procedure when evidence of a reactor coolant seal cooler leak exists is a performance deficiency. It is more than minor because it adversely affects the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This procedural omission could impact the timeliness of the required action to trip the plant and secure the reactor coolant pump. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 12, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria in the alarm response procedure to enter the abnormal operating procedure for reactor coolant pump malfunctions if there is evidence of a reactor coolant pump seal leak. Because this violation is of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-04, “Five Examples of Inadequate Procedures for the Initiating Events Cornerstone.” This was the fourth of five examples.

Example 5: No Procedural Guidance for a Volume Control Tank Level Instrument Failure

Introduction. The team identified a fifth of five examples of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The licensee failed to include instructions for a volume control tank level instrument failure in the alarm response procedure.

Description. During preparation of the initial license examination operating test, the examiners randomly selected a volume control tank level instrument failure as a malfunction for one of the scenarios. During review of the licensee provided reference materials, the examiners discovered that Annunciator Response Procedure ARP-CB-1,2,3/A2, Revision 39, did not contain procedural guidance for a failure of the volume control tank level instrument. If the controlling volume control tank level instrument fails low, the suction source for the charging pumps swaps over to the safety injection/refueling water tank, which has a higher boron concentration than the reactor coolant system and volume control tank. Without procedural guidance instructing the operator to swap the controlling volume control tank level instrument away from the failed channel, the charging pump suction will continue to draw from the more highly borated safety injection/refueling water tank, resulting in a larger plant transient than if such guidance was proceduralized. This malfunction was removed from the examination due to the lack of procedural guidance for responding to the event.

Analysis. The failure to include instructions for a volume control tank level instrument failure is a performance deficiency. It is more than minor because it adversely affects the procedure quality attribute of the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This procedural omission could impact the timeliness of the required action to switch the level control channel, increasing the time that the charging pumps take a suction from the safety injection refueling water storage tank and increasing the severity of the plant transient. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include actions in the alarm response procedure to swap the volume control tank level control channels in the event of an instrument failure. Because this violation is of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report 2012-03140, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-04, “Five Examples of Inadequate Procedures for the Initiating Events Cornerstone.” This was the fifth of five examples.

**Failure to Correctly Translate the Design Basis for the Containment Spray System**

Introduction. The team identified a finding of very low safety significance involving a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that, design control measures did not assure that the design basis for safety related systems were correctly translated into procedures. Specifically, the licensee failed to correctly translate the design basis of the containment spray system into Technical Basis Document Procedure TBD-EOP-05, “Uncontrolled Heat Extraction.”

Description. During the initial license written examination development, the examiners randomly selected a topic concerning the failure of a spray pump and its impact on the containment spray system. During review of licensee provided materials, the examiners noted an inconsistency between descriptions of the containment spray system. Specifically, Updated Safety Analysis Report (USAR) Section 6.3.4.2, System Training Manual Volume 15, “Emergency Core Cooling System,” Section 2.221, and Lesson Plan 07-11-22, “Safety Injection, Containment Spray and Shutdown Cooling” all state that containment spray pump SI-3A is interlocked with containment spray header isolation valve HCV-345 and containment spray pump SI-3B is interlocked with containment spray header isolation valve HCV-344 such that it requires a containment spray actuation signal and the pump motor breaker closed before the respective valve will open. However, Technical Basis Document Procedure TBD-EOP-05, “Uncontrolled Heat Extraction” states that a modification installed an interlock such that if only one containment spray pump starts, containment spray header isolation valve HCV-344 will close to prevent containment spray pump damage, and a later modification extended this interlock to include containment spray header isolation valve HCV-345. The technical basis document therefore indicates that if only one containment spray pump fails to start, neither containment spray header isolation valve will open, which is not per the design of the containment spray system and is incorrect.

Analysis. The failure to correctly translate the design basis of the containment spray system into an emergency operating procedure technical basis document is a performance deficiency. It is more than minor because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The incorrect guidance in the emergency operating procedure basis document could result in a licensed operator taking incorrect action to secure containment spray based on a faulty understanding of the expected system response. Securing containment spray during a main steam line break would challenge the safety function of the containment building, increasing the risk to public health and safety. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding: (1) did not represent only a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool; (2) did not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; (3) did not represent an actual open pathway in the physical integrity of reactor containment system, containment isolation system, and heat removal components; and (4) did not involve an actual reduction in function of hydrogen ignitors in the reactor containment. This finding was determined not to have a cross-cutting aspect because it is not indicative of current plant performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that “measures shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.” Contrary to the above, the licensee failed to assure the design basis for the containment spray system was correctly translated into specifications, drawings, procedures and instructions. Specifically, in October 2006, the licensee did not correctly translate the design basis of the containment spray system into the Technical Basis Document Procedure TBD-EOP-05, “Uncontrolled Heat Extraction.” Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2011-06802, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-05, “Failure to Correctly Translate the Design Basis for the Containment Spray System.”

**Inadequate Procedures with Four Examples for the Barrier Integrity Cornerstone**

The team identified a finding of very low safety significance (Green) involving a non-cited violation of Title 10 CFR Part 50, Appendix, B, Criterion V, "Instructions, Procedures, and Drawings," with four examples.

Example 1: Failure to Include Appropriate Quantitative or Qualitative Acceptance Criteria in Reactor Coolant Pump Procedures

Introduction. The team identified the first of four examples of a non-cited violation of Title 10 CFR Part 50, Appendix, B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the normal operating instruction for reactor coolant pumps, OI-RC-9, "Reactor Coolant Pump Operation," contains pump trip requirements that conflict with the pump trip requirements provided in Abnormal Operating Procedure AOP-35, "Reactor Coolant Pump Malfunctions."

Description. During examination administration, one of the major transients for a scenario was an intersystem loss of coolant accident from the reactor coolant system to the component cooling water system through a leak in one reactor coolant pump (RCP) seal cooler. This resulted in numerous high seal temperature alarms on reactor coolant pump RC-3A. The candidate in the at-the-controls position tripped the reactor, performed all the standard post trip actions for his position, and then was cued by the control room supervisor to trip reactor coolant pump RC-3A, approximately seven minutes after the reactor coolant pump trip criteria was met for the high lower cavity seal temperature. During followup questions after the scenario, the candidate reviewed operating instruction OI-RC-9, "Reactor Coolant Pump Operation." Precaution 19 of OI-RC-9 states, in part, "A running RCPs lower seal (cavity) temperature shall not exceed 200$^{o}$F. If the Reactor is critical, the Reactor shall be tripped and the RCP immediately stopped." The candidate explained to the examiners that the precaution meant that the reactor would be tripped, verified tripped, and then the reactor coolant pump would be tripped before continuing with the rest of the standard post trip actions. The examiners then reviewed Abnormal Operating Procedure AOP-35, "Reactor Coolant Pump Malfunctions," Section I, Steps 1 and 1.1, which state that "if the lower seal cavity temperature exceeds 200$^{o}$F and the reactor is critical, then trip the reactor, implement Emergency Operating Procedure EOP-00, "Standard Post Trip Actions," and then stop the affected reactor coolant pumps." These two procedures conflict with each other on when the reactor coolant pump must be tripped once the lower seal cavity temperature exceeds its limits and the reactor is tripped. Per vendor document WCAP-16175-P-A, "Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants," dated March 2007, the reactor coolant pump must be tripped twenty minutes after the lower seal cavity temperature has exceeded its limits to prevent reactor coolant pump seal damage. Standard post trip actions typically take approximately ten minutes to complete, which is less than the twenty minutes calculated in the analysis. Therefore this procedural conflict on whether the reactor coolant pump must be tripped immediately or if it can be tripped after all the standard post trip actions are complete would not result in damage to the reactor coolant pump seals.

Analysis. The failure to include appropriate acceptance criteria in reactor coolant pump procedures is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This conflicting information could have significantly affected the operator's ability to trip the reactor coolant pumps at the appropriate time. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the finding: (1) did not represent only a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool; (2) did not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; (3) did not represent an actual open pathway in the physical integrity of reactor containment system, containment isolation system, and heat removal components; and (4) did not involve an actual reduction in function of hydrogen ignitors in the reactor containment. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, prior to April 20, 2012, the licensee had a normal operating instruction and an abnormal operating procedure that conflicted with each other on when reactor coolant pumps must be tripped. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-03144, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-06, “Inadequate Procedures with Four Examples for the Barrier Integrity Cornerstone.” This was the first of four examples.

Example 2: Incorrect Criteria for Restoring Proper Control Element Assembly Group Overlap

Introduction. The team identified a second of four examples of a non-cited violation of Title 10 CFR Part 50, Appendix, B, Criterion V, “Instructions, Procedures, and Drawings.” Specifically, the licensee’s control element assembly regulating group excessive overlap Alarm Response Procedure ARP-DCS-SCEAPIS directs the operators to restore the control element assemblies to proper overlap using MANUAL GROUP mode, which is blocked on excessive overlap. The control element assemblies must be moved using MANUAL INDIVIDUAL mode.

Description. During initial license written examination development, the examiners randomly selected a topic concerning the operation of the control element assembly drive system. During the examiners' review of licensee provided reference material, the examiners noted a discrepancy in Annunciator Response Procedure ARP-DCS-SCEAPIS. System Training Manual Volume 11, "Control Rod Drive System," paragraph 2.62 states that an overlap rod block will stop rod motion to prevent excessive overlap between groups and that rod block bypass may be used in the MANUAL INDIVIDUAL mode with an overlap present. Paragraph 2.104 states that the rod-block relays are energized by the secondary control element assembly position indication system (SCEAPIS) digital control system if it detects an out-of-sequence/overlap condition. Paragraph 2.106 states that a rod-block condition may be bypassed via the rod-block relays and that only one rod at a time may be bypassed. However, in the SCEAPIS digital control system Annunciator Response Procedure, ARP-DCS-SCEAPIS, Revision 1, for annunciator process alarm "Regulating Group Excessive Overlap," the procedure directs the operators check for improper overlap for the control element assembly groups, and if improper overlap exists, to return the control element assembly group to the proper overlap in MANUAL GROUP per OI-RR-1, "Reactor Regulating System Normal Operation." If the operators were to attempt to restore the control element assembly group to the proper overlap in MANUAL GROUP mode, the control element assemblies would not be able to be moved due to the rod-block that is inserted when a regulating group excessive overlap is detected. The only way to move the control element assemblies is to bypass the rod-block using the bypass switch and to move each individual control element assembly to the proper overlap in MANUAL INDIVIDUAL mode. Therefore, the Annunciator Response Procedure ARP-DCS-SCEAPIS is incorrect as written and the task to restore proper control element assembly group overlap cannot be accomplished in accordance with the procedure as written. The licensee entered this issue into the corrective action program as Condition Report 2011-07172, and has since issued a procedure change to ARP-DCS-SCEAPIS to correct the issue.

Analysis. The failure to include correct criteria for restoring proper control element assembly group overlap is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This example could have significantly affected the operator's ability to perform the activity affecting quality, in this case, restoring control element assemblies to the proper overlap to prevent exceeding fuel design limits. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the fuel cladding barrier was affected but did not affect the reactor coolant system or containment barriers. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to November 22, 2011, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include appropriate criteria for restoring proper control element assembly group overlap. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2011-07172, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-06, "Inadequate Procedures with Four Examples for the Barrier Integrity Cornerstone." This was the second of four examples.

Example 3: Procedure Does Not Address Excessive Overlap between Control Element Assembly Groups Caused By Operator Error

Introduction. The team identified a third of four examples of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's Annunciator Response Procedure ARP-DCS-SCEAPIS for excessive overlap between control element assembly groups only contains actions for a failure of the digital control system and does not address the possibility of operator error. Additionally, Abnormal Operating Procedure AOP-02, "CEA and Control System Malfunctions," also does not contain actions for this condition.

Description. During initial license written examination preparation, the examiners randomly selected a topic on the control element assembly drive system. During review of licensee provided reference materials, the examiners noted that although rods can be moved in MANUAL GROUP or MANUAL INDIVIDUAL mode, which does not use the digital control system sequencing, the Annunciator Response Procedure ARP-DCS-SCEAPIS, Revision 2, for rod motion out of sequence addresses only a digital control system failure. Specifically, for the "Regulating Groups Out of Sequence Movement" alarm, the procedure directs the operators to have an instrument and controls technician determine the cause of the alarm and if there is improper sequencing, to go to AOP-02, "CEA and Control System Malfunctions," Revision 8. Abnormal Operating Procedure AOP-02, Section V, "Loss of Position Indication or Functions," is entered if there is improper overlap or sequencing observed. This abnormal operating procedure specifies actions to take to correct a digital control system failure and to verify proper operation of the digital control system, but does not address the possibility that the improper sequencing was caused by operator error. Normal operating practice is to move the control element assemblies in MANUAL SEQUENTIAL mode, which does use the digital control system sequencing. The licensee wrote Condition Report 2012-09654 to address this issue.

Analysis. The failure to include criteria to address excessive overlap between control element assembly groups caused by operator error is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This example could have significantly affected the operator's ability to perform the activity affecting quality; in this case, restoring control element assembly group overlap to prevent exceeding fuel design limits. In accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," a phase 1 screening was performed and determined that the finding was of very low safety significance (Green) because the fuel cladding barrier was affected but did not affect the reactor coolant system or containment barriers. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to November 27, 2011, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria to address excessive overlap between control element assembly groups caused by operator error. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2012-09654, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-06, "Inadequate Procedures with Four Examples for the Barrier Integrity Cornerstone." This was the third of four examples.

Example 4: Conflicting Abnormal Operating Procedure Guidance for Reactor Shutdown Requirements in AOP-21 "High RCS Activity"

Introduction. The team identified a fourth example of a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's Abnormal Operating Procedure AOP-21, "Reactor Coolant System High Activity," has multiple values for high RCS activity requirements that are confusing for operators to use in order to determine if it is necessary to initiate a plant shutdown. Additionally, this procedure is not current with the most recent SO-O-43 "Fuel Reliability Management Plan" action levels. The Fuel Reliability Management Plan

SO-O-43 currently lists four action levels, while the actions in procedure AOP-21 are based on five action levels. The fifth action level actions would not be performed since no fifth action level is defined in SO-O-43.

Description. During preparation of the initial license examination operating test, the examiners randomly selected high reactor coolant system activity levels for one of the required technical specification entry determinations for the control room supervisor. During the examination pre-validation visit, the instructors were unable to use Abnormal Operating Procedure AOP-21, "Reactor Coolant System High Activity," to determine the necessary course of action based on the given reactor coolant system activity levels. A cue of a reactor coolant system activity level of >1 µCi/gm but less than 18.8 µCi/gm was provided as the initiating event. AOP-21 has multiple values for high reactor coolant system activity requirements that are difficult to use in order to determine if it is necessary to initiate a plant shutdown. Specifically, Step 6 states to initiate a reactor shutdown if reactor coolant activity exceeds 1 µCi/gm Dose Equivalent I-131 for more than 100 hours during one continuous time interval, which in this case, was not exceeded. However, the next step states that if Dose Equivalent I-131 is greater than 0.2 µCi/gm, initiate a reactor shutdown at a rate of 3% per hour. These two steps appear to be in conflict with each other, in that per Step 6, reactor coolant activity could exceed 1 µCi/gm for up to 100 hours without requiring a reactor shutdown, and yet Step 7 states that if reactor coolant activity exceeds 0.2 µCi/gm, which is less than the value in Step 6, a reactor shutdown is required. Additionally, during review of reference material associated with the procedure, the examiners discovered that although the "Fuel Reliability Management Plan," SO-O-43, had recently been revised to reduce the required action levels based on reactor coolant system activity levels from five to four, the abnormal operating procedure has not been changed to reflect the new activity levels. Specifically, a note in the abnormal operating procedure states that if chemistry is improved to within the requirements of Action Level 4 or lower of SO-O-43, prior to plant shutdown, power operation may continue, subject to the requirements of other Action Levels and technical specifications. However, because SO-O-43 has been changed so that there are only four action levels, chemistry will always be within the requirements of Action Level 4 or lower, and the note would then allow continued plant operation, even though Action Level 4 in the revised SO-O-43 would require a plant shutdown. This adds to the difficulty in using the procedure. During the examination pre-validation visit, the instructors were unable to come to a consensus on whether or not a reactor plant shutdown was required by procedure or not. Therefore, this failure to include the appropriate quantitative criteria to ensure that the reactor shutdown is accomplished when required could have significantly impacted operator's ability to perform the task had a high reactor coolant system activity level been identified. The licensee wrote Condition Report 2012-03141 to address this issue.

Analysis. The failure to include appropriate quantitative criteria for ensuring that the reactor shutdown is accomplished when required is a performance deficiency. The performance deficiency is more than minor and is therefore a finding because it adversely affects the procedure quality attribute of the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. This procedure specifies actions to take to maintain fuel barrier integrity within design limits. In accordance with Inspection Manual Chapter 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings,” a phase 1 screening was performed and the finding was determined to be of very low safety significance (Green) because the fuel cladding barrier was affected but did not affect the reactor coolant system or containment barriers. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a corrective action program with a low threshold for identifying issues in that licensed operators deviate from procedures when procedures cannot be implemented as written without writing necessary condition reports to fix the deficient procedures [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to April 20, 2012, the licensee failed to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to ensure that the abnormal operating procedure for high reactor coolant system activity contained clear and direct actions to be taken for specific reactor coolant activity levels and that the procedure was current with respect to the fuel reliability management plan. Because this violation is of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report 2012-03141, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012301-06, “Inadequate Procedures with Four Examples for the Barrier Integrity Cornerstone.” This was the fourth of four examples.

**Minor violation**

The team also identified a minor violation of Title 10 CFR Part 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the licensee failed to include acceptance criteria in Abnormal Operating Procedure AOP-39, “Toxic Gas,” Revision 2, for determining if a site evacuation is required and if personnel can be evacuated prior to plume arrival. The result is that two senior reactor operators could use the same procedure to come to two different conclusions regarding evacuation. This was identified by the examination team during initial license examination preparation and was removed from the examination to allow the licensee to address the issue. In discussion with regional experts in the area of emergency preparedness, the examiners discovered that although the information necessary to make a decision regarding evacuation is not located in the abnormal operating procedure, adequate information is included in other emergency preparedness documents to ensure that an adequate decision would be made regarding site evacuation in the case of a toxic gas event. The violation is minor because, although there is no minor example in Inspection Manual Chapter 0612, Appendix E, that is similar to this case, the answer to all of the more than minor screening questions is no. The licensee has written Condition Report 2011-06612 to address this issue.

Other Examination Development Observations

Several training materials, lesson plans, system training manuals, and other non-quality documents had incorrect information in them when compared to design basis and other quality documents. These items were discovered during examination development and administration. Once identified, these issues were documented in Condition Report 2012-04749.

.3 Operator Knowledge and Performance

a. Scope

On April 13, 2012, the NRC proctored the administration of the written examinations to all ten applicants. Although the NRC graded the written examinations, the licensee was still required to analyze the results and send to the regional office. These results were received by the regional office on April 26, 2012, and may be accessed in the post-examination comments and analysis file, placed in the NRC's document system (ADAMS) under the accession numbers noted in Enclosure 2.

The NRC examination team administered the various portions of the operating tests to all ten applicants during the week of April 16, 2012.

b. Findings

No findings were identified.

Six of the applicants passed the written examination and all ten passed all parts of the operating test. The final written examinations, final operating test, and post-examination analysis may be accessed in the ADAMS system under the accession numbers noted in Enclosure 2, which also includes nine post-examination comments.

The examination team noted the following generic weaknesses either during examination development (with licensed operators and instructors) or during administration:

(1) Most of the applicants and several licensed operators failed to start the auxiliary oil pump prior to attempting a start of a main feedwater pump during a job performance measure,
(2) The validation crew had trouble with several events, including the tracking and correct positioning of trip units when instruments failed,
(3) The validation crew performed multiple procedure deviations,
(4) Circle/slash techniques were inconsistent and incorrect in at least one case during job performance measure administration,
5) Use of alarm response procedures by the applicants was weak,

(6) Several crews delayed tripping the reactor when required for an Anticipated Transient Without Scram (ATWS) event,
(7) There were several communication breakdowns by the applicants during the scenarios as they progressed, and
(8) Two applicants were stopped during job performance measure administration due to the potential risk of bodily injury because they were not following the procedure for racking out a 480 Vac breaker.

These generic weaknesses were entered into the corrective action program as Condition Report 2012-05804.

The licensee is also performing an Apparent Cause Analysis 2012-04110 for the high failure rate on the written examination.

.4 Simulation Facility Performance

a. Scope

The NRC examiners observed simulator performance with regard to plant fidelity during examination validation and administration.

b. Findings

No findings were identified.

.5 Examination Security

a. Scope

The NRC examiners reviewed examination security during both the onsite preparation week and examination administration week for compliance with 10 CFR 55.49 and NUREG-1021. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. Findings

No findings were identified.

**4OA6 Meetings, Including Exit**

The chief examiner presented the preliminary examination results to Messrs. Mike Prospero, Site Vice President, and his staff on April 20, 2012. A final telephonic exit was conducted on June 7, 2012, between Messrs. Kelly Clayton, Chief Examiner, and Thomas Giebelhausen, Manager, Operations Training.

All proprietary information was returned to the licensee at the conclusion of the examination and exit meeting.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

### KEY POINTS OF CONTACT

Licensee Personnel

M. Prospero, Plant Manager
M. Smith, Operations Manager
C. Verdoni, Shift Manager, Standards
R. Haug, Training Manager
T. Giebelhausen, Operations Training Manager
R. Cade, Operations Training Supervisor, Support
R. Lowery, Operations Training Supervisor, Initial Programs
D. Acker, Licensing Engineer

NRC Personnel

John Kirkland, Senior Resident Inspector

### LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000285/2012301-01 | NCV | Seven Examples of Inadequate Procedures for the Mitigating Systems Cornerstone (Section 4OA5) |
| 05000285/2012301-02 | NCV | Failure to Periodically Update the Final Safety Analysis Report for the Emergency Diesel Generators (Section 4OA5) |
| 05000285/2012301-03 | NCV | Failure to Comply with Technical Specification 2.3(1)(i) for Safety Injection Tank Operability (Section 4OA5) |
| 05000285/2012301-04 | NCV | Five Examples of Inadequate Procedures for the Initiating Events Cornerstone (Section 4OA5) |
| 05000285/2012301-05 | NCV | Failure to Correctly Translate the Design Basis for the Containment Spray System (Section 4OA5) |
| 05000285/2012301-06 | NCV | Four Examples of Inadequate Procedures for the Barrier Integrity Cornerstone (Section 4OA5) |

### ADAMS DOCUMENTS REFERENCED

Accession No. ML12156A005 - FINAL WRITTEN EXAMINATION (delayed release until April 20, 2014)
Accession No. ML12156A003 - FINAL OPERATING TEST
Accession No. ML12156A007 - POST EXAMINATION ANALYSIS AND COMMENTS

## NRC Review of FCS Written Post-Examination Comments

**Note: A complete text of the licensee's post examination analysis and comments can be found in ADAMS under Accession Number ML12156A007.**

**Question 5**

Given the following:

A loss of off-site power has occurred.
Both emergency diesel generators have started and are loaded on their respective buses.
Breaker 1A4-10 trips, de-energizing 480V bus 1B4A

Which of the following LPSI isolation valves would **NOT** automatically reposition following a SIAS?

A. HCV-327
B. HCV-329
C. HCV-331
D. HCV-333

Key Answer: B

**Licensee Comments for Question 5:**
Fort Calhoun Station requests that this question be deleted from the examination per ES-403 D.1.b because this question is not linked to job requirements.

As discussed during examination validation, FCS believes that this question requires the recall of knowledge that is too specific for the closed reference test mode (i.e., it is not required to be known from memory).

In the control room, operators can quickly determine the power supply to these valves by looking at the control switch labels. If an electrical bus is lost, operators are trained to use AOP-32, "LOSS OF 4160 VOLT or 480 VOLT BUS POWER," to determine the affected plant equipment. Section XII of AOP-32 addresses the loss of bus 1B4A. AOP-32 Attachment B provides a complete list of components powered from Bus 1B4A including MCC-4A1 (page 314). It also includes a list of components powered by MCC-4A1 including HCV-329. (page 324)

This K/A has a RO Importance factor of 2.7* (An importance factor barely above 2.5 with variability in the rating responses.) In referring to the asterisk, NUREG-1122, Rev 2, states "These marks indicate a need for examination developers to review plant-specific materials to determine whether or not that knowledge or ability is indeed appropriate for inclusion in any given examination." Since the power supplies are clearly indicated in the control room and the operators are trained in the use of AOP-32, requiring Operators to memorize the bus power supplied to MOVs would NOT increase protection of the health and safety of the public. Therefore, this knowledge is NOT appropriate for inclusion in the examination.

**NRC Resolution of Question 5**

Firstly on job link requirements:

10CFR55.41(b)(7) requires operators be examined on "Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features." This question tests the applicants' knowledge of the failure mode on loss of power for a Low Pressure Injection System (LPSI) motor-operated valve (HCV-329) during an emergency condition at the plant. Because LPSI is a safety system important in preventing core damage during certain emergency conditions, this implicitly meets the requirements in the code of federal regulations for a safety system and failure modes. Therefore the NRC concluded that this topic is linked to the job requirements for a Reactor Operator as defined in the Code of Federal Regulations.

Secondly regarding recall of knowledge from memory:

Fort Calhoun Station specifically communicated to the NRC chief examiner that operators are not required to know any power supplies from memory beyond large 4160vac loads because they can use plant procedures and board indications to determine what equipment has lost power during loss of power events. The NRC disagrees with this approach because there are numerous K2 K/A's in the catalog that are considered to be very important to the NRC and this is one of them. The importance of power supply knowledge is reflected by the number of K2 K/As in the catalog and because many of the associated importance factors are greater than 2.5, including this question. This question is a direct match for the K/A it was written for and is also covered by another K/A for system 006, Emergency Core Cooling Systems. Specifically, K2.04, with an importance rating on 3.6 for a reactor operator, requires power supply knowledge for ESFAS operated valves.

On the NRC's Operator Licensing website under "Frequently Asked Questions," question number 401-42 is answered below with <u>underlined text</u> as emphasis specifically tied to the Question 5 comments from the licensee. Note that Fort Calhoun Station's comments regarding the use of procedures and indications in the control room to determine power supplies for equipment instead of knowing these items from memory is countered in this specific question posted on the NRC's public website:

> **401.42 - Why is it valid to use a closed reference exam for initial license exams when it is really important that the operator use all of the tools available to him on shift?**
>
> Open-reference items on the initial license examination should be used judiciously and sparingly because the examination should focus on the broader content areas that rely primarily upon learned information, committed to memory.
>
> In nearly every field of study (e.g., medicine, law, and education), the testing required for initial licensing or certification is more demanding than that required to maintain certification. <u>The rationale is that newly licensed personnel should possess a broad body of knowledge and ability to perform their job independently and without the aid of supplemental knowledge contained in procedures.</u> This by no means suggests that procedures should not be used, but rather that initial license testing should emphasize those areas where procedures need not be used.
>
> Through their training, operators must learn set points, immediate actions, system designs and interrelationships, administrative procedures, and applications of knowledge to the job. <u>The knowledge that is learned is expected to be demonstrated through the</u>

NRC examination format that measures recognition and recall of safety-significant knowledge without relying on references. This approach is consistent with the timely retrieval of information that may be required during the licensed operators' job and that might otherwise not be possible if the applicants prepared only for open-reference examinations. If too many open-reference questions are allowed on the initial licensing examination, the need and ability to learn and retrieve a broad body of knowledge would be lessened. Similarly, the confidence that the baseline body of knowledge had been truly established could be questioned.

Once initial competency is assured, then ongoing training and testing, which is more review-like, focused and specialized in nature, can make more appropriate use of the open-reference format, as is done on requalification examinations. However, for the reasons stated above, the NRC does not plan to increase the limited and judicious use of open-reference questions on the initial license examination.

Finally regarding the asterisk in the K/A catalog:
FCS commented that the asterisk in the K/A catalog (NUREG-1122, Revision 2, Supplement 1) next to the importance factor for this K/A requires a plant-specific review to determine if it is appropriate for inclusion in the examination. These words are taken directly from the K/A catalog and were considered during examination development when FCS first commented that the question was not a fair question because the station does not train on it. This specific K/A had been discussed with Region IV staff and another licensee in 2007 and the NRR program office was consulted for assistance. At that time, NRR concluded that this is an important K/A and therefore could not be removed from the written examination based on a lack of training. The chief examiner consulted NRR again for this case prior to examination approval with the same direction from them to leave the topic on the examination because this knowledge is appropriate for inclusion on the examination.

The contention was also made during examination reviews that this question was not a fair question because FCS does not train on power supplies at the 480vac level. Failure to train on a given topic is no basis for removal of the topic from the examination as indicated in NUREG-1021, Revision 9, Supplement 1, page 4 of ES-401, section D.1.b, second paragraph from the bottom of the page.

FCS recommended that this question be deleted from the examination based on their belief that this question is not linked to job requirements. The NRC Region IV staff and NRR program office disagreed with this position and therefore the question was not deleted from the examination.

**Question 23**

Current letdown heat exchanger outlet temperature is 113 °F as measured by temperature elements TE-2897A and TE-2897B. Assuming the 2" letdown controller TC-2897B setpoint is set at the bottom of its normal band, 8" letdown controller TC-2897A is set to its normal setpoint, and both controllers TC-2897A and TC-2897B are in the AUTO position, what is the expected response of letdown heat exchanger CCW outlet valves TCV-2897A and TCV-2897B?

A. TCV-2897A is closed; TCV-2897B is closed
B. TCV-2897A is closed; TCV-2897B is modulated open
C. TCV-2897A is modulated open; TCV-2897B is closed
D. TCV-2897A is modulated open; TCV-2897B is modulated open

Key Answer: B

**Licensee Comments for Question 23:**
Fort Calhoun Station requests that both choices B and D be accepted as correct per ES-403 D.1.b because the question stem did not provide all the information necessary to differentiate between choices B and D.

The normal setpoint for TC-2897B is 110°F to 115°F with an allowable band of ± 2°F per OI-CH-1, Attachment 12. Thus the bottom of the band is 110°F. Per OI-CH-1,TC-2897A is normally set approximately 5°F higher than TC-2897B or 115°F for the setpoint described in the stem. However, according to OI-CH-1, the setpoint could be set as high as 120°F. The allowed range for TC-2897A per IC-CP-01 is ± 7.5°F, therefore its actual setpoint could be as low as 107.5°F, in which case, TCV-2897A would be open at 113°F. It could also be as high as 127.5°, in which case TCV-2897A would be closed at 113°F. Therefore both choices B and D could be correct depending on applicant assumptions.

**NRC Resolution of Question 23:**
The NRC agreed with FCS's position that there could be two correct answers for this question based on new information (Document IC-CP-01-2897) provided during the post-examination comments phase. Because of the larger band for this valve, it could be possible from the stem conditions that the valve could be open or it could be closed at the given stem temperature. This document was never provided to the NRC examination author despite three written examination validations by licensed operators and training staff at FCS. After reviewing the new information provided by the station, the NRC agreed with the station's position that both answer B and distractor D could be correct based on the setup in the stem of the question. However, the NRC disagreed with the recommendation to accept both answers as correct because these choices are direct opposites of each other. Answer B has valve TCV-2897A closed while distractor D has valve TCV-2897A modulated open. NUREG-1021 specifically states at the bottom of page 3 and the top of page 4 of ES-403 D.1.c that "If it is determined that there are two correct answers, both answers will be accepted as correct. If, however, both answers contain conflicting information, the question will likely be deleted. For example, if part of one answer states that operators are required to insert a manual reactor scram, and part of another answer states that a manual scram is not required, then it is unlikely that both answers will be accepted as correct, and the question will probably be deleted." Therefore, because the two answers contain conflicting information, the NRC deleted the question from the examination.

**Question 30**

With the reactor at 100% power, main steam safety valve MS-280 fails open. What is the initial steam generator level response and the reason for that response?

A. Steam generator level decreases due to decreased recirculation flow
B. Steam generator level decreases due to increased recirculation flow
C. Steam generator level increases due to decreased recirculation flow
D. Steam generator level increases due to increased recirculation flow

Key Answer: D

**Licensee Comments for Question 30:**
Fort Calhoun Station requests that the key answer be changed to "C" per ES-403 D.1.b based on newly discovered technical information that supports a change to the answer key.

A calculation based on performance data for the FCS replacement steam generators indicates that the recirculation flow decreases from 90% to 100% power from $1.063 \times 10^7$ lb/hr at 90% power to $9.987 \times 10^6$ lb/hr at 100% power.

In the attached Table 2-9-1 (redacted because it is Proprietary Information), values for circulation ratio, steam flow and feedwater flow at 90% and 100% power are given. In the table note, circulation ratio is defined as circulation flow rate / steam flow rate.

Therefore circulation flow rate = circulation ratio X steam flow rate.

STM 25 states "The recirculation flow plus the feed flow yield the circulation flow."

Therefore recirculation flow = circulation flow – feed flow.

Using these formulas recirculation flow @ 90% power = $1.063 \times 10^7$ lb/hr and recirculation flow @ 100% power = $9.987 \times 10^6$ lb/hr showing that recirculation flow decreases with an increase in steam flow supporting C as the correct answer. The statement in the STM that indicates that recirculation flow increases with an increase in steam flow will be corrected.

**NRC Resolution of Question 30:**
The NRC reviewed the newly discovered technical information submitted during the post-examination phase of the review. This document from Mitsubishi Heavy Industries (MHI) defines recirculation ratio for the replacement steam generators (2007) and provides different results for this parameter than the steam generators that were installed during construction. This document also demonstrates that the System Training Manual (STM-25) that the NRC used to develop the question is incorrect. The station wrote a condition report because of the risk of negative training for operators on this concept and it is understood that this will be corrected in STM-25 and all associated training documents. Therefore, the NRC concluded that the proposed correct answer D is incorrect and the correct answer should be distractor C.

**<u>Question 33</u>**

The reactor was just shutdown for refueling. OI-RM-1, RADIATION MONITORING is being performed to verify the area monitor setpoints for monitor RM-072, Containment Main Floor, East (1022'-10"). The as-found warn/alert setpoint was set at 20 mr/hr. The as-found high alarm setpoint was set at 30 mr/hr. These setpoint values are the same as the last time this check was performed, while the reactor was still at power.

Per OI-RM-1, the warn/alert setpoint _____(1)_____ and the high alarm setpoint _____(2)_____.

A. Needs adjustment; needs adjustment
B. Needs adjustment; does not need adjustment
C. Does not need adjustment; needs adjustment
D. Does not need adjustment; does not need adjustment

Key Answer: B

**Licensee Comments for Question 33:**
Fort Calhoun Station requests that this question be deleted from the examination per ES-403 D.1.b because this question is not linked to Reactor Operator job requirements and it is at the wrong license level.

As discussed during examination validation, this question is not linked to operator job requirements because reactor operators do not adjust the area radiation monitor setpoints.

This question has an asterisk following its importance rating. In referring to the asterisk, NUREG-1122, Rev 2, states "These marks indicate a need for examination developers to review plant-specific materials to determine whether or not that knowledge or ability is indeed appropriate for inclusion in any given examination." It is important that the operators know how to respond to a radiation monitor alarm. However, this question is not linked to operator job requirements because reactor operators do not adjust the radiation monitor set points. Set point adjustments are performed by I&C.

There is a task on the Licensed Operators task list for changing the setpoints for process radiation monitors but not for changing area monitor setpoints.

OP-3A does direct the SRO to notify I&C to adjust the area radiation monitor setpoints.
In addition, the STA changes setpoints for the computer.

In addition, the K/A discusses the "Ability to manually operate and/or monitor in the control room: Alarm and interlock setpoint checks and adjustments." However, the question addresses a knowledge rather than an ability. Therefore, there is also a K/A mismatch.

**NRC Resolution of Question 33:**
Firstly on job link requirements and wrong license level:
10CFR55.41(b)(11) requires [reactor] operators be examined on "Purpose and operation of radiation monitoring systems, including alarms and survey equipment." This question tests the

applicants' general radiation knowledge that the radiation levels decrease when the reactor is shutdown and therefore the warn setpoint would need to be adjusted down. Also, the question is requiring that the applicants know that the alarm setpoint would not be adjusted. The applicants were not asked how to make the adjustment or what the new value would need to be after shutdown. Because this topic contains the operation of radiation monitoring equipment, including alarms, it meets the requirements contained in the reactor operator applicant section of the Code of Federal Regulations. Therefore the NRC concluded that this topic is linked to the job requirements for a Reactor Operator as defined in the Code of Federal Regulations.

Secondly the point that Reactor Operators do not adjust area radiation monitors:
FCS stated that reactor operators do not make the adjustments for these radiation monitors and therefore during the examination validation the licensee wanted the question removed from the examination. The NRC did not agree with this position because licensed operators usually do not make adjustments on even more important instruments than radiation monitors, including Reactor Protection System (RPS) setpoint calibrations and adjustments. However, they are required to know when RPS setpoints are required to be adjusted therefore this is not a valid reason to remove the topic from the examination. Again, failing to train on a topic that was selected for the examination is not a valid basis to have the topic (question) removed from the examination. Licensed operators use these radiation monitoring instruments in the control room to make important decisions during plant emergencies for various conditions, including shutdown and at-power conditions. The NRC might agree that Reactor Operators do not adjust area radiation monitors but the NRC believes that licensed reactor operators are required to know that the warn/alert setpoint is required to be changed during different plant conditions.

Thirdly regarding the asterisk in the K/A catalog and K/A mismatch:
FCS commented that the asterisk in the K/A catalog (NUREG-1122, Revision 2, Supplement 1) next to the importance factor for this K/A requires a plant-specific review to determine if it is appropriate for inclusion in the examination. These words are taken directly from the K/A catalog and were considered during examination development when FCS first commented that the question was not a fair question because the station does not train on it. The station again commented that licensed reactor operators do not adjust radiation monitor setpoints. See discussion above on this topic.

The K/A for this question reads, "072 Area radiation Monitor (ARM) System- Ability to manually operate and/or monitor in the control room: Alarm setpoint checks and adjustments," with an importance rating of 3.0* for a reactor operator applicant. Therefore this question meets the K/A because it requires the ability to monitor a setpoint adjustment (down in this case) and the importance rating indicates it is much greater than 2.5 and should be included on the examination for these reasons. If the area radiation monitor in question did not have an adjustment then that would be a reason to throw out this K/A even though its importance factor is 3.0,* however this is not the case for this question at FCS. Therefore, this question was not deleted from the examination.

**Question 50**

Given the following:

• Reactor has tripped
• S/G RC-2A is 540 psia and stable
• S/G RC-2A is at 30% WR level and stable
• S/G RC-2B is 480 psia and decreasing
• S/G RC-2B is 25% WR level and decreasing
• T-avg is 500°F and decreasing
• SGLS has isolated both S/Gs

What is the current expected response of AFW and the primary purpose for that response?

A. AFW will feed both steam generators to maintain heat sink
B. AFW will feed only S/G RC-2A to maintain heat sink of intact steam generator
C. AFW will feed only S/G RC-2A to minimize cooldown of RCS
D. AFW will NOT feed either steam generator to minimize cooldown of RCS

Key answer: C

**Licensee Comments for Question 50:**
Fort Calhoun Station requests that choices B and C be accepted as correct per ES-403 D.1.b because the unclear wording in the choices made choices B and C both correct. This comment was made by one applicant and supported by all other applicants during the post-examination review. Applicants stated they had difficulty deciding between choices B and C because both are important to reactor safety. Most picked B because the wording addressed feeding the intact steam generator rather than isolating the faulted steam generator.

The AFW system is designed to isolate the faulted S/G to minimize RCS cooldown while maintaining flow to the intact S/G to ensure RCS heat removal. AFW will be isolated from S/G RC-2B to minimize the cooldown of the RCS making the key answer, C, correct. AFW will be supplied to S/G RC-2A (i.e. both steam generators will not be isolated) to maintain the intact S/G as a heat sink to satisfy the RCS Heat Removal Safety Function making choice B equally correct.

The AFW STM supports both answers as being correct.

Per the Auxiliary Feedwater STM "The auxiliary feed actuation logic is designed to provide AFW to the intact steam generator. The additional cooldown from automatic auxiliary feedwater actuation is prevented by a low steam generator pressure condition in the ruptured steam generator and by an adequate level in the intact unit."

**NRC Resolution of Question 50:**
The NRC agrees with FCS that the stem and answer selections allow both Answer C to be correct and distractor B to be correct because of poor wording in the answer choices. Feeding only the good generator ensures the heat sink is maintained via the RC-2A generator and by not feeding the faulted generator (RC-2B) it also minimizes the cooldown, therefore answer C and distractor B are both correct. The NRC accepts both answer C and distractor B as correct.

**Question 62**

Given the following plant conditions:
A station blackout has occurred
D/G#2 has been restored and loaded
RCS pressure is 2090 psia
1 charging pump is running
WR S/G levels indicate 29% in both steam generators
FW-10 is mechanically bound
FW-54 has failed to start
Tcold has risen 9°F in the last few minutes and is continuing to increase

Which ONE of the following actions should the operators take next?

A. Start motor driven auxiliary feedwater pump FW-6 to provide feedwater to the steam generators

B. Use the demineralized water system to provide feedwater to the steam generators

C. Establish once-through cooling by opening PORV PCV-102-2 and starting HPSI pump SI-2B ONLY

D. Establish once-through cooling by aligning power as necessary to start two HPSI pumps and open both PORVs

Key answer: D

**Licensee Comments for Question 62:**
Fort Calhoun Station requests that both choices C and D be accepted as correct per ES-403 D.1.b due to an unclear stem that did not provide the applicants with information need to differentiate between choices "C" and "D."

The actions in choice "C" are part of the actions in choice "D."

The ultimate goal is to establish once-through-cooling using both PORVs and two HPSI pumps. However, If bus 1A3 is unavailable, EOP-20, HR-4 directs the operator to step 2 where he is directed to establish "partial once-through-cooling" by starting HPSI pump SI-2B and opening PCV-102-2. Therefore, choice "C" is clearly correct. In step 3, the operators are directed to crosstie buses and start HPSI pump SI-2C, Steps 8-10 directs the operators to crosstie breakers and open PORV, PCV-101-1 making the key answer "D" also correct.

**NRC Resolution of Question 62:**
In this question the stem clearly asks "Which ONE of the following actions should the operator take next?" Because of the stem conditions, the procedure to be in for this event is EOP-20. Because one vital bus is de-energized, the contingency action step 1.1 is entered and it asks if once through cooling can NOT be established because a Vital 4160 V bus is de-energized (and it is) then step 1.1a is the correct next appropriate step and it directs you to go to step 2. In step 2 it states "Establish partial once-through cooling by performing the following:"

a. Stop all RCPs
b. De-energize all PZR Heaters
c. Open all HPSI Loop Isolation Valves
d. Start SI-2B, HPSI Pump
e. Open PCV-102-2, PORV

Partial once-through cooling is defined at FCS to mean one injection pump and one PORV. Once-Through cooling at FCS requires two injection pumps and two PORVs, therefore distractor C is clearly not correct because it is a TRUE-FALSE statement that stands alone as FALSE because this distractor is defined to be partial once-through cooling and with the ONLY in bold at the end of distractor C cannot be correct (psychometrically flawed as well and needs to be corrected before inclusion in the bank). Answer D is also not correct because this is clearly not the next step in the procedure. It is the step after the step for establishing partial once-through cooling (ie Step 3 in EOP-20). This makes answer D incorrect for the stem wording and therefore the NRC concludes that there is no correct answer for this question and therefore it must be deleted from the examination.

**Question 67**

Which one of the following instrument errors would cause the reactor power calculated by XC-105 to be greater than actual reactor power?

A. Feedwater pressure indicating lower than actual.
B. Feedwater temperature indicating higher than actual.
C. Feedwater flow indicating higher than actual.
D. Reactor Coolant Pump Power indicating lower than actual

Key Answer: C

**Licensee Comments for Question 67:**
Fort Calhoun Station requests that choices C and D be accepted as correct per ES-403 D.1.b because newly discovered technical information supports this change to the key. This comment was made by an applicant during the post-examination review. The calorimetric equation is:

$Q_{rx} = M_{fw} (h_{stm} - h_{fw}) - Q_{RCP} + Q_{losses}$

Feedwater flow indicating higher than actual will cause the calculated reactor power to be greater than the actual reactor power making the key answer, “C,” correct. RCP power indicating lower than actual would also cause the calculated reactor power to be greater than actual reactor power making choice “D” also correct. RE-CPT-RX-0003, Attachment 9.1 also supports choices C and D. It indicates that having feedwater flow greater than actual will have a conservative effect (XC-105 will indicate higher that actual power). It also indicates that having RCP power less than actual will have a conservative effect on XC-105.

**NRC Resolution of Question 67:**
The NRC examiners pulled this question from the FCS bank and noted that it had been used on the 1999 NRC examination. During this time only two distracters were required to be credible although the NRC expected licensees to create all distracters as credible. The one distractor that was not credible was replaced with a new distractor and the mistake was not caught by during multiple validations by the licensee or the NRC that this new distractor was actually correct as well. The NRC accepts answer C and distractor D as both correct.

**<u>Question 86</u>**

The reactor is currently at 100% power. It was determined at 0815 today, April 13th, that one safety injection tank had a boron concentration that was less than the refueling concentration.

Assuming boron concentration cannot be restored to within limits, the plant must be in HOT SHUTDOWN no later than _____(1)_____.

A. April 14th at 2015
B. April 15th at 0815
C. April 16th at 2015
D. April 17th at 0815

Key answer: C

**Licensee Comments for Question 86:**
Fort Calhoun Station requests that answers A and C be accepted as correct per ES-403 D.1.b because the question stem did not provide all of the information necessary to differentiate between choices A and C.

Numerous SRO applicants made this comment during the post-examination review. The stem states that "assuming boron concentration cannot be restored to within limits," and does not provide a reason why normal SIT boration cannot be performed.

<u>Most applicants made a similar assumption that a problem with valves, interlocks or piping is preventing restoration of boron concentration.</u>

Technical Specifications 2.3(2)e through h state:
e. Any valve, interlock or piping associated with the safety injection and shutdown cooling system which is not covered under d. above but which is required to function during accident conditions may be inoperable for a period of no more than 24 hours.

f. One safety injection tank may be inoperable for reasons other than g. or h. below for a period of no more than 24 hours.

g. Level and/or pressure instrumentation on one safety injection tank may be inoperable for a period of 72 hours.

h. One safety injection tank may be inoperable due to boron concentration not within limits for a period of no more than 72 hours."

Technical Specification 2.3(2) contains the following statement:
"If the system is not restored to meet the minimum requirements within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours."

Therefore, if an applicant assumed that there was a problem with piping, valves or interlocks, then Technical Specifications 2.3(2)e or 2.3(2)f would apply. Both of these situations are 24 hour LCO's.

If These LCO's are not met, then the plant must be placed in hot shutdown within 12 hours per Technical Specification 2.3(2) for a total of 36 hours to get to hot shutdown.

Therefore, 4/13 @ 0815 + 36 hours = 4/14 @ 2015 making choice "A" also correct.

FCS Training staff concurs that the assumption made by several of the applicants is reasonable based on needed information not provided in the stem and that "A" should also be accepted as correct.

**NRC Resolution of Question 86:**
The stem of this question clearly does not provide any information that there is a problem with valves or piping because it was not desired for the applicants to enter LCO 2.3(2)e or f because the stem of the question stated "Assuming boron concentration cannot be restored to within limits" which is close to cueing the applicant that LCO 2.3(2)h is the correct statement to enter because very close wording between the LCO and stem exists for this question. The explanation provided in the question worksheet is clear and allows only one correct answer, Answer C, based only on conditions provided in the stem.

NUREG-1021, Revision 9, Supplement 1, Appendix E, page 2, item 7, paragraph 2, clearly states that
*"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 or the alarm is expected to activate as a result of the conditions that are stated in the question."*

There were no questions asked on this stem during examination administration and the Appendix E brief was performed twice by the NRC to ensure these concepts were emphasized. Additionally, the licensee also read the Appendix E brief to the applicants before the NRC arrived to administer the written examination. Applicants were therefore briefed that they could not make any assumptions that were not included in the stem of the question.

The NRC concludes that this question is correct as written with only one correct answer, Answer choice C.

**Question 89**

The reactor was operating at full power when a fire was reported in Room 18 (CCW Heat Exchanger Room). The reactor has been tripped and plant status is as follows:

- Raw water pumps AC-10A and AC-10C are running
- CCW Heat Exchanger AC-1C is operating with a CCW exit temperature of 125 deg. F
- Procedure AOP-06 FIRE EMERGENCY FOR AUXILIARY BUILDING RADIATION CONTROLLED AREAS AND CONTAINMENT was entered

What should the control room supervisor direct the crew to do to achieve and maintain the plant in a safe shutdown condition and what procedure should be used to complete these actions?

A. Implement Procedure AOP-11, LOSS OF COMPONENT COOLING WATER and establish feed and bleed cooling for the CCW system components

B. Implement Procedure AOP-18, LOSS OF RAW WATER and use the fire protection system water to cool the CCW system components

C. Stay in Procedure AOP-06 and establish feed and bleed cooling for the CCW system components

D. Stay in Procedure AOP-06 and use Demineralized water to cool the CCW system components

Key answer: B

**Licensee Comments for Question 89:**
Fort Calhoun Station requests that the key answer be changed to “C” per ES-403 D.1.b because of newly discovered technical information that supports a change to the answer key.

One applicant made the following comment during the post-examination review. AOP-18, Attachment B, “Fire Protection System Backup” steps 2 and 3 require that hoses be run from BOTH Fire Hose Cabinets FP-7C (Room 19) AND FP-7D (Corridor 4). Applicants eliminated the key answer “B” because for a fire in Room 18, Hose Cabinet FP-7D would be used for fire fighting and would not be available for Fire Protection System Backup.

AOP-18, Contingency step 12.1 states “IF the Fire Protection System is NOT available, THEN GO TO Step 14.” Step 14 directs the establishment of feed and bleed cooling by opening a CCW drain valve. AOP-06 would not be exited, making choice “C” the correct answer. The FCS Training staff concurs with this comment.

**NRC Resolution of Question 89:**
The NRC recognizes that in order to fight the fire in room 18 that there is only one cabinet, FP-7D, that a hose can be connected to in order to fight the fire. This means that it would be extremely difficult and is not proceduralized on what actions to take to connect fire main water to

the CCW system to cool it as well for the stem conditions given. The NRC did not have knowledge that there was only 1 fire main connection for this event and concluded that the procedure is inadequate as written and this issue should have been caught by FCS staff during validation or during fire protection system walk-downs with procedures. This procedure issue makes AOP-18 incorrect because it does not provide guidance on how to connect to the fire main system for CCW cooling while a fire is occurring in this room. This makes Answer B incorrect. Because AOP-06 directs you to AOP-18 for this event to use fire main water to cool CCW this makes distractor C incorrect as well. Regarding AOP-06 and distractor C, there are no steps in this procedure for feed and bleed cooling, again making distractor C incorrect. Therefore the NRC concludes that there is no correct answer for this question, the procedure and training are inadequate for this event, and this question is therefore deleted from the examination.