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

July 7, 2009

Mr. Bruce H. Hamilton
Vice President
Duke Power Company,
LLC d/b/a Duke Energy Carolinas, LLC
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR PLANT- NRC EXAMINATION REPORT 05000369/2009301 AND 05000370/2009301

Dear Mr. Hamilton:

During the period May 11 - 15, 2009, the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the McGuire Nuclear Plant Units 1 and 2. At the conclusion of the tests, the examiners discussed preliminary findings related to the operating tests and the written examination submittal with those members of your staff identified in the enclosed report. The written examination was administered by your staff on May 22, 2009.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

If you have any questions concerning this letter, please contact me at (404) 562-4550.

Sincerely,

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

Docket Nos.: 50-369, 50-370
License Nos.: NPF-9, NPF-17

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

(cc: w/encl - See Page 3)

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(cc: w/encl - See Page 3)
* See previous concurrence

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DOCUMENT NAME: O:\MCGUIRE EXAMINATIONS\INITIAL EXAM 2009-301\CORRESPONDENCE\MG EXAM REPORT 2009-301 (REVISION 1) (2).DOC

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Letter to Bruce H. Hamilton from Malcolm T. Widmann dated July 7, 2009

SUBJECT: MCGUIRE NUCLEAR PLANT- NRC EXAMINATION REPORT 05000369/2009301 AND 05000370/2009301

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-369, 50-370
License Nos.:	NPF-9, NPF-17
Report No.:	05000369/2009301 and 05000370/2009301
Licensee:	Duke Power Company (DPC), LLC
Facility:	McGuire Nuclear Plant, Units 1 & 2
Location:	Huntersville, NC 28078
Dates:	Operating Tests - May 11 - 15, 2009 Written Examination - May 22, 2009
Examiners:	G. Laska, Chief, Senior Operations Examiner M. Bates, Senior Operations Engineer M. Riches, Operations Engineer T. Kolb, Senior Reactor Operations Engineer, NRR/IOLB (Re-Certification) K. Schaaf Operations Engineer (Certification)
Approved by:	Malcolm T. Widmann, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000369/2009301; 05000370/2009301; 05/11-15/09 & 05/22/09; McGuire Nuclear Plant, Units 1& 2, Operator License Examinations.

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

Members of the McGuire Nuclear Plant staff developed both the operating tests and the written examination.

The NRC administered the operating tests during the period of May 11-15, 2009. Members of the McGuire Nuclear Plant training staff administered the written examination on May 22, 2009. Three Reactor Operator (RO) and two Senior Reactor Operator (SRO) applicants, passed both the written examination and operating test. Three RO applicants passed the operating test but failed the written examination. Four applicants were issued licenses commensurate with the level of examination administered.

One RO applicant passed the operating test, but passed the written examination with a score between 80% and 82%. This applicant was issued a letter stating that he passed the examination and issuance of his license has been delayed pending any written examination appeal that may impact the licensing decision for the application.

There were four post examination comments.

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA5 Operator Licensing Initial Examinations

a. Inspection Scope

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

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

The NRC examiners evaluated six Reactor Operator (RO) and two Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. The examiners administered the operating tests during the period of May 11 – 15, 2009. Members of the McGuire Nuclear Plant training staff administered the written examination on May 22, 2009. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the McGuire Nuclear Plant, met the requirements specified in 10 CFR Part 55, "Operators' Licenses."

b. Findings

No findings of significance were identified. The NRC determined, using NUREG-1021, that the licensee's examination submittal was within the range of acceptability expected for a proposed examination.

Three RO and two SRO applicants passed both the operating test and written examination. Three RO applicants passed the operating test but failed the written examination.

One RO applicant passed the operating test, but passed the written examination with a score between 80% and 82%. This applicant was issued a letter stating that he passed the examination and issuance of his license has been delayed pending any written examination appeal that may impact the licensing decision for the application.

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

The licensee submitted two post-examination comments concerning the operating test and two comments concerning the written examination. A copy of the final written RO and SRO examinations and answer keys, with all changes incorporated, and the

licensee's post-examination comments may be accessed in the ADAMS system (ADAMS Accession Number(s) ML019700325, ML019700317, and ML019700343).

4OA6 Meetings

Exit Meeting Summary

On May 15, 2009, the NRC examination team discussed generic issues associated with the operating test with Mr. Regis Repko, Station Manager, and members of his staff. The examiners asked the licensee if any of the examination material was proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee personnel

S. Bradshaw, Training Manager
K. Crane, Regulatory Compliance
S. Helms, Operations Training Supervisor
F. Kirk, Operations Shift Supervisor
S. Mosteller, Operations-Training On Loan SRO
R. Pope, Operations Training manager
R. Repko, Station Manager
T. Rhodes, Operations Shift training Manager

NRC personnel

J. Brady, Senior Resident Inspector

NRC Resolution to the Facility Comments

A complete text of the licensee's post examination comments can be found in ADAMS under Accession Number ML019700343.

LICENSEE COMMENT 1

RO Admin JPM A-2 (Perform Daily Surveillance Checklist)

This JPM was performed on the Simulator with static setup. The candidates were given the task of performing PT/1/A/4600/003 B Enc 13.1 (Daily Surveillance Items Checklist). This PT contains a list of Tech Spec required surveillances required to be performed every 24 hours and a total of 4 abnormal indications were provided in the simulator setup. One of the discrepancies was associated with a required check of FWST solution temperature which was indicating approximately 97 Deg F on gage 1FWP-5030. Per the PT (Pg 2 of 8) the maximum FWST temperature is "Less than or equal to 100 F as read on 1FWP-5030 (Begin cooling FWST at 95 F)." This check in the PT contains a reference to Note 12 which states: "Actual Tech Spec limit is 100F. However, cooling of FWST should begin at 95 F to prevent exceeding Tech Spec limit." This note also references a PIP, (M96-2432) and was placed in the PT as a preventative measure to preclude exceeding the TS FWST temperature limit. This step in the PT satisfies SR 3.5.4.1 in which the FWST solution temperature is required to be verified to be $\geq$ 70 F and $\leq$ 100 F once per 24 hours. In addition, SLC surveillance TR 16.9.11.1 is also satisfied which contains the same temperature range and periodicity.

During performance of this JPM, several of the candidates did not identify or communicate to the examiner the need to begin cooling the FWST. While this would be considered a deficiency on the part of the candidate, this step was incorrectly identified as a critical step in the JPM standard and the omission of this notification should not be considered an automatic failure for this JPM. The justification of removal of this designation is that it does not impact the successful completion of PT. The Acceptance Criteria for this PT is as follows:

> "Each applicable surveillance item meets its specified Acceptance Criteria per Enclosure 13.1 (Daily Surveillance Items Checklist) or is identified as a discrepancy, evaluated per Tech Specs, and appropriate corrective action taken".

Due to the fact that both Tech Spec surveillances associated with this temperature check are satisfied with the indication given the acceptance criteria for this element is met, and therefore has no impact on the successful completion of the assigned PT.

NRC ANALYSIS:

The NRC agrees with the licensee that the PT acceptance criteria for FWST temperature were met and the step should not have been deemed a critical step.

NRC RESOLUTION:
The JPM will be graded with this step being a non-critical step.

LICENSEE COMMENT 2

RO/SRO Admin JPM A-1a (Determine Boric Acid Addition to the FWST)

During performance of this JPM, several of the candidates differed in how they interpreted Data Book Curve 7.7 (Refueling Water Storage Tank Level). The curve is a graphical representation of the FWST volume vs. tank level. This relationship is linear, and instead of just obtaining the volume from the graph for 440 inches and 480 inches, these candidates created a gallons per inch relationship based on information provided in an information block provided on the graph. This information block provided a value in gallons at 484 inches and 455.88 inches along with an "unusable" volume at 0 inches. If the candidate takes the difference between these levels and volumes, and subtracts the unusable volume, the result is an accurate means to determine the volume of water needed for the desired level change. Due to the very large capacity of the tank, the difference of the thickness of a line on the graph results in a 1000 gallon difference in the value obtained. Using the method, the value obtained differs from the value used in the standard of the JPM by approximately 1500 gallons (30000 vs. approx 31500).

Using the numbers in the information block on Curve 7.7, there are two methods to determine the gallons / inch in the tank which is then multiplied by the desired change in tank level (in inches) to determine the total makeup volume to the FWST.

Using one method, the candidate would take either of the volumes listed in the information block (394,089 gallons or 372,100 gallons), subtract the unusable volume at zero inches (15,638 gallons), and then divide by the number of inches at the respective volume to obtain a gallons/inch value for the FWST.

Using the second method, the candidate would take the difference between the two tank volumes in the information block and then divide by the difference in inches between those two volumes.

In order to successfully complete the JPM, the candidate must first correctly determine the required volume of water to add as described above. Then, the candidate must determine the required amount of boric acid required to maintain the water addition at the Tech Spec required boron concentration. Due to the large variation in volume, (from even being the thickness of a line different in reading curve), a single value of 30000 gallons was chosen for the desired make up volume and no range was given in the standard. This was done to enhance the discriminatory value of the boric acid calculation. Even a small range provided in interpreting the curve resulted in a very large range in the resultant boric acid volume required. For the total make up volume used, the range provided in the standard is reasonable, but does not allow for much variation in determining the total volume needed. This resulted in the candidates who actually used an accurate means to determine the makeup volume, coming up with a boric acid volume which was outside the acceptable range as given in the JPM standard.

Based on the discussion above, the candidates who use the gallons per inch method should pass this JPM as long as the boric acid volume required is correct for the total makeup volume used. The difference in makeup volume would not challenge either the operation or the operability of the FWST. As long as the correct desired Boron concentration, RHT boron concentration, and BAT concentration are used, the difference in the total makeup volume only changes the final level in the FWST. The final FWST boron concentration would be the same.

NRC ANALYSIS:

The NRC reviewed the information the licensee submitted and determined that either method was acceptable to determine how much water must be added to raise FWST level from 440" to 480." Using the curve and determining 30,000 gallons were required, as read off of the graph was the method described in the Job Performance Measure. The method of using the volumes listed in the information block (394,089 gallons or 372,100 gallons), subtracting the unusable volume at zero inches (15,638 gallons), and then dividing by the number of inches at the respective volume to obtain a gallons/inch value for the FWST was also deemed to be acceptable.

NRC RESOLUTION:

The JPM will be graded as satisfactory if the correct values for either method are used throughout the calculation.

RO QUESTION 74

LICENSEE COMMENT:

The licensee's comments pertain to the second part of each answer choice; therefore, the following discussion is limited to that piece of each answer choice. The licensee states that conditions of a LOCA and SI were presented in the stem; however, the severity of the LOCA was not provided. The licensee also states that the correct answer depends on the severity of the LOCA and that the applicant was forced to make an assumption on the severity of the LOCA due to the lack of information in the stem. In other words, the licensee contends that the applicant was forced to make an assumption on whether or not a phase B isolation signal (containment pressure > 3 psig) was present as a result of the LOCA. They state that if the applicant assumed that a phase B isolation signal was NOT present, then the applicant would have determined that "C" (the originally intended correct answer) was the correct answer. They state that if the applicant assumed that a phase B isolation signal WAS present, then the applicant would have determined that "D" was the correct answer. Therefore, due to the lack of necessary information in the stem, the licensee proposes that both "C" and "D" be accepted as correct answers.

NRC ANALYSIS:

The NRC partially agrees with the licensee's contention. Because not enough information was presented in the stem to determine the magnitude or location of the LOCA, or the time at which the question referred, the applicant was forced to make an assumption on the presence of a phase B isolation signal. As stated above, depending on that assumption, either "C" or "D" would be a correct answer choice.

NUREG-1021, Operator Licensing Examination Standards for Power Reactors, Revision 9, Supplement 1, states (ES-403, Section D.1.c):

> *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.*

"C" states that RV containment isolation valves are open and "D" states that RV containment isolation valves are closed. These two answers are conflicting. These answer choices are analogous to the above referenced example from ES-403, which states that a manual scam being required and a manual scram not being required are conflicting. Due to the conflicting nature of "C" and "D", the question will be deleted in accordance with the guidance of ES-403, Section D.1.c.

NRC RESOLUTION:

Delete question.

SRO QUESTION 81

LICENSEE COMMENT:

The licensee's comments pertain to the second part of each answer choice; therefore, the following discussion is limited to that piece of each answer choice. The licensee states that, in part, the question was written to test the sources of water that are assumed to contribute to the design basis containment flooding accident as documented in the Westinghouse Owners Group (WOG) Basis Document for FR-Z.2 (Response to Containment Flooding). However, the licensee states that the WOG Basis Document is a generic document that is written to support all Westinghouse units, most of which do not have ice condensers. Therefore, the licensee contends that basing the question on the WOG Basis Document was inaccurate in that it does not take into consideration the unique design feature of ice condenser containments, such as the one at McGuire. The licensee contends that the question be based on their own background document for FR-Z.2, which resides in their lesson plan (OP-MC-EP-FRZ, Rev. 17). The licensee states that this is material to the accuracy of the answer because a major source of water assumed in the McGuire Design Basis Containment Flood Analysis is the ice condenser. Furthermore, the licensee contends that the background document for FR-Z.2 does not list the CA storage tank as a contributor to the maximum sump level. The licensee states that the question asked for the sources of water that would contribute to the maximum flood level; therefore, not including contributions from the ice condenser AND listing the CA storage tank as a contributor resulted in there being no correct answer. Thus, the licensee requests that the question be deleted from the exam.

NRC ANALYSIS:

The NRC partially agrees with the licensee's contention. The question specifically asks for the sources of water which contribute to the maximum sump level with regards to the Analysis for the Design Basis Containment Flooding Event. The question does not ask anything about FR-Z.2, the contents of the licensee's Basis Document for FR-Z.2, or the WOG Basis Document for FR-Z.2. Therefore, the only relevant supporting documentation for the question resides within the Analysis for the Design Basis Containment Flooding Event.

Again, the question statement asks for the sources of water which contribute to the maximum sump level with regards to the Analysis for the Design Basis Containment Flooding Event. The Analysis for the Design Basis Containment Flooding Event does not use the CA storage tank as a contributor to the maximum sump level. Answer choice "B" lists the CA storage tank as a contributor to the maximum sump level in the Analysis for the Design Basis Containment Flooding Event. Therefore, answer choice "B" is not a correct answer choice.

The Analysis for the Design Basis Containment Flooding Event lists the ice condenser as a contributor to the maximum sump level. Answer choice "D" states that ONLY the entire contents of the NC system, FWST, and NI accumulators are contributors to the maximum sump level in the Analysis for the Design Basis Containment Flooding Event. Since "D" does not state that the ice condenser is a contributor to the maximum sump level, it too is not a correct answer choice, thus resulting in no correct answer being listed for the question.

NRC RESOLUTION:

Delete question.

SIMULATOR FIDELITY REPORT

Facility Licensee: McGuire Nuclear Plant

Facility Docket No.: 05000369/2009-301 and 05000370/2009-301

Operating Test Administered: May 11 through May 15, 2009

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and, without further verification and review in accordance with Inspection Procedure 71111.11 are not indicative of noncompliance with 10 CFR 55.46. No licensee action is required in response to these observations.

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

Item	Description
VX Controls: RAF-D2 RAF-D-4	Controls on the simulator did not operate the same way the plant operates. Applicants attempted to open these dampers on the simulator by taking the control switch to open. The simulator required the switches to be held in the open position until the fully open signal was obtained. In the plant the operator need only take the control switch to the open position, and the damper will open fully. PIP-M-09-2566 was written to address this issue.
"B" Train DRPI	"B" Train digital rod position indication did not indicate the same for the same malfunction in one of the three scenarios administered. SDR 2905 was written to address this issue.