UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
REGION II
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December 18, 2006

Southern Nuclear Operating Company, Inc.
ATTN: Mr. H. L. Sumner
Vice President
P. O. Box 1295
Birmingham, AL 35201

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC EXAMINATION REPORT 05000348/2006301 AND 0500364/2006301

Dear Mr. Sumner:

During the periods of October 23 - 26 and November 21, 2006, the Nuclear Regulatory Commission (NRC) administered operating examinations to employees of your company who had applied for licenses to operate the Farley Nuclear Plant. At the conclusion of the examination, the examiners discussed the examination questions and preliminary findings with those members of your staff identified in the enclosed report. The written examination was administered by your staff on October 30, 2006.

Four Senior Reactor Operator (SRO) applicants passed both the written examination and operating test. One SRO applicant failed the written examination. There were two post examination comments submitted. The NRC resolution to the post examination comments is included in this report as Enclosure 2. A Simulation Facility Report is included in this report as Enclosure 3.

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

Should you have any questions concerning this letter, please contact me at (404) 562-4607.

Sincerely,

\\RA\\

Robert C. Haag, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-348, 50-364
License Nos. NPF-2, NPF-8

Enclosures: (See next page)

Enclosures:
1. Report Details
2. NRC Resolution to the Farley Post Examination Comments
3. Simulation Facility Report

cc w/encls:

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NUCLEAR REGULATORY COMMISSION

REGION II

| | |
|---|---|
| Docket Nos.: | 50-348, 50-364 |
| License Nos.: | NPF-2, NPF-8 |
| Report No.: | 05000348/2006301, 05000364/2006301 |
| Licensee: | Southern Nuclear Operating Company, Inc. |
| Facility: | Joseph M. Farley Nuclear Plant |
| Location: | Columbia, AL 36319 |
| Dates: | Operating Tests - October 23 - 26 and November 21, 2006<br>Written Examination - October 30, 2006 |
| Examiners: | R. Baldwin, Chief, Senior Operations Examiner<br>S. Rose, Senior Operations Engineer<br>R. Aiello, Senior Operations Engineer |
| Approved by: | Robert C. Haag, Chief<br>Operations Branch<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

ER 05000348/2006301, 05000364/2006301; 10/23-26/2006, 10/30/2006, and 11/21/2006; Farley Nuclear Plant; Licensed Operator Examinations.

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

The NRC administered the operating tests during the periods of October 23 - 26 and November 21, 2006. Members of the Farley Nuclear Plant training staff administered the written examination on October 30, 2006. The written examinations and the operating test outlines were developed by the NRC. The operating test details were developed by the Farley Nuclear Plant training staff.

Four Senior Reactor Operators (SRO) passed both the operating test and written examination. One SRO applicant failed the written examination. Each applicant who passed the operating test and written examination with an overall score greater than 82% and SRO-only score greater than 74%, as applicable, was issued an operator license commensurate with the level of examination administered.

There were two post examination comments.

No findings of significance were identified.

## Report Details

## 4. OTHER ACTIVITIES

### 4OA5 Operator Licensing Initial Examinations

a. Inspection Scope

The NRC developed the operating tests outlines and the written examination and the licensee developed the operating tests details in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. The NRC and licensee's examination teams reviewed the proposed examinations. Examination changes agreed upon between the NRC and the licensee were made according to NUREG-1021 and incorporated into the final version of the examination materials.

The examiners reviewed the licensee's examination security measures while preparing and administering the examinations to ensure examination security and integrity complied with 10 CFR 55.49, "Integrity of examinations and tests."

The examiners evaluated five SRO applicants who were being assessed under the guidelines specified in NUREG-1021. The examiners administered the operating tests during the periods of October 23 - 26 and November 21, 2006. Members of the Farley Nuclear Power Plant training staff administered the written examination on October 30, 2006. The evaluations of the applicants and review of documentation were performed to determine if the applicants, who applied for licenses to operate the Farley Nuclear Power Plant, met requirements specified in 10 CFR 55, "Operators' Licenses."

b. Findings

No findings of significance were identified.

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

Four SROs passed both the operating test and written examination. One SRO applicant failed the written examination. Each applicant who passed the operating test and written examination with an overall score greater than 82% and SRO-only score greater than 74%, as applicable, was issued an operator license commensurate with the level of examination administered.

The SRO written examination, with knowledge and abilities (K/As), question references/answers, and examination references, may be accessed in the ADAMS system (ADAMS Accession Numbers ML061840214, ML061840215 and ML061840213).

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

4OA6 Meetings

Exit Meeting Summary

On October 27, 2006, the examination team discussed generic issues with Mr. Randy Johnson and members of his staff. The examiners asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

## PARTIAL LIST OF PERSONS CONTACTED

Licensee personnel

R. Johnson, Nuclear Plant General Manager
R. Bayne, Performance Analysis Supervisor
S. Chestnut, Engineering Support Manager
D. Christiansen, Nuclear Operations Training Supervisor
D. Hobson, Operations Superintendent
L. Hogg, Security Manager
G. Ohmstede, Plant Instructor-Nuclear
B. Oldfield, Quality Assurance Supervisor
C. Richter, Plant Instructor-Nuclear
T. Youngblood, Assistant General Manager

NRC personnel

C. Patterson, Senior Resident Inspector
E. Crow, Senior Resident Inspector
J. Baptist, Resident Inspector

## NRC RESOLUTION TO THE FARLEY POST EXAMINATION COMMENTS

A complete text of licensee's post exam comments can be found in ADAMS under Accession Number ML063310506.

**SRO QUESTION # 80**

The question proposes a time line in which two pressurizer problems are identified, the first being a pressurizer heater group (group 'A') being out of service and the second being an uncontrollable rise in pressurizer (PRZR) level at a rate of 4.5%/hr. The question concerns itself with Technical Specification (TS) 3.4.9, Pressurizer, in that, this TS requires, in the Limiting Condition for Operation (LCO), that PRZR level be maintained less than or equal to 63.5 percent (%) and that two groups of heaters be OPERABLE with the required power capacity.

The question asked the applicants to determine from, the list of distractors, the appropriate distractor that identified the **most limiting** actions required by Technical Specifications. The applicant was expected to evaluate the time PRZR level would reach its TS limit, as well as, determine when PRZR heater group 'A' being out of service would exceed its TS limit. Based on the time line the applicant was expected to select distractor 'C' based on the uncontrollable PRZR level rise, as the only correct answer. Distractor 'C' is "Unit 1 must be in Mode 3 no later than 1100 and must be in Mode 4 no later than 1700." This is obtained by applying Action A of TS 3.4.9, from the time the PRZR level starts to increase uncontrollably (0200). A change of PRZR level of 13.5% would raise level to the TS limit (63.5%), at the rate of 4.5%/hr change. This would take 3 hours to increase PRZR level to the TS limit and would occur at 0500. Once level increases to slightly above the TS limit, then Action A.A.1 (Be in Mode 3 with the reactor trip breakers open in 6 hours) would apply, thus requiring Mode 3 be entered at 1100 (6 hours from 0500). Then Action A.A.2 would required the plant to be in Mode 4 12 hours from the time the TS was not met, or 1700.

**LICENSEE COMMENT:**

The licensee recommends taking 'D' as the only correct answer instead of 'C.' The licensee contends, based on feedback for distractors 'A,' 'B,' and 'C,' that the exam development team obviously missed the equals sign (in the TS). And because of that, each one of those distractors used 63.5% as the final PRZR level and the times used in the dictractors were calculated using this value. The licensee further states that the original answer 'C' was not correct since the stem says "**most limiting actions required by Tech Specs**." The licensee further points out that the level of the PRZR (based on the time line) would be 63.5% at 0500. In order for the applicants to determine when level went out of specification, the applicants would have to determine when level exceeded the TS requirement. The licensee points out that if a value of 63.6% was used to determine PRZR level was out of specification, the calculation would show that it would take 3.02222 hours to reach 63.6%. This equates to 3 hours and several minutes. (This actually equates to 3 hours, 1 minutes and 19.8 seconds). The licensee contends that this would discount distractor ‘C’ as the correct answer since the time is greater than 1100 (actually 1101.19.9) that is in distractor ‘C.’ The licensee also states that “the only other choice to pick is ‘D’ as correct since it **does** meet the bounds of the PRZR heater Tech Spec **and** meets the bounds of the level Tech Spec, even though it is not the most limiting answer that **should** be on the page.” The licensee states that during the examination one applicant stated that distractor ‘C’ was technically not correct, but makes the best answer. The licensee believes that it would be appropriate to change the answer on the question to ‘D’ instead of ‘C’ since it (‘D’) is the only correct answer.

**NRC RESOLUTION:**

The NRC disagrees with the licensee’s recommendation.

The answer was based on a PRZR level rise from 50% to 63.5% resulting in a 13.5% rise in PRZR level. With the 4.5%/hr level increase, it would take 3 hours to achieve a level of 63.5%, the upper allowable TS limit (as stated by the licensee above). Since the PRZR level rise continues at the specified rate, the applicants should have realized that the TS limit of 63.5% would have been exceeded any time (actually seconds) after the 3 hours were reached.

Technical Specification Pressurizer level instruments use an analog scale with 2% divisions and the PRZR level recorder reads out in digital to the hundredth place. The accuracy to which an operator could read the TS instrument would be at half of the divisions (2) and then half again (0.5). Therefore, the best an operator could read the TS PRZR level instrument was 63.5%, providing the instruments were reading exact. The PRZR level recorder could be used to see a finer change in PRZR level, since it reads to the hundredth place, a level of 63.51% could be observed. Using the PRZR level recorder reading this would result in a time of 3.00222 hours, which would equate to 3 hours and 7.99 seconds, which would have been essentially logged as 3 hours.

Distractor ‘D’ was designed to identify that the PRZR heater issue was a plausible distractor and require the applicants to determine that the PRZR level issue was more limiting. This would have identified any knowledge weaknesses if the applicant did not know there was a **“more limiting**” PRZR TS level issue. Therefore, distractor ‘D’ would never have been the correct answer.

As soon as 63.5% was exceeded, operators are expected to follow Condition ‘A’ of the LCO. Therefore, the **most limiting** action required by TS would be to enter Condition ‘A,’ and in accordance with ‘A.1,’ be in Mode 3 with reactor breakers open within 6 hours AND ‘A.2.’ be in mode 4 during the following 12 hours. This equates to the original answer, ‘C.’

As stated in the above discussion once pressurizer level exceeds the TS limit of 63.5%, level continues to increase at the rate specified of 4.5%/hr. It would take a very short period of time (approximately 7.99 seconds) for pressurizer level to exceed the TS level limit.

The NRC maintains that distractor ‘C’ is the only correct answer.

**SRO QUESTION # 95:**

The question asks the applicant to determine if the RCS Pressure Safety Limit (SL) of Technical Specifications (TS) (2.1.2) is exceeded, then determine which one of the distractors would be more severe and the reason. The question provided three distractors in which the SL applied (Modes 1, 2 and 4), and another distractor that the SL does not apply (Mode 6). The applicants were expected to recall from memory that there is no SL for Mode 6 based on TS applicability and TS bases and then to determine from the remaining distractors that distractor C is the only correct answer. Which states that Mode 4 is more limiting since the reactor vessel temperature is lower and, therefore, less ductile. Distractor ‘D’, an incorrect answer, states that “Mode 6, (with the head on and the bolts not fully tensioned), since the reactor vessel temperature is lower, and therefore, less ductile.” Distractor ‘D’ was considered a very plausible distractor because the SL does not apply in Mode 6 and the temperature and pressure of Mode 6 is lower than any of the other Modes listed in the other distractors.

**LICENSEE COMMENT:**

The licensee recommends taking ‘D’ as the only correct answer instead of ‘C.’ The licensee states that “the stem of the question asks which one of the modes is most severe and the reason if the TS limit is exceeded.” They further state “since there is no qualifier as to what the severity is based on, such as Tech Spec Bases, the only other option is to apply Generic Fundamental training to the question.” The licensee argues that ‘D’ is correct for the question as written because the stem stating RCS (Reactor Coolant System) pressure has exceeded T.S. 2.1.1 (the licensee actually meant TS 2.1.2) (>2735 psig), would indicate that Mode 6 is more limiting than Mode 4 due to the temperature of the vessel and the ductility of the vessel. The licensee argument pivots around Operator fundamentals training, in that brittle fracture of the Reactor Vessel would be more likely at **the lower temperature** associated with Mode 6 (≤ 200 °F) than that of Mode 4 (> 200 °F and < 350 °F) at the same pressure of > 2735 psig. The licensee identifies pages 5-8 of General Physics Corporation Thermodynamics, Chapter 10, Brittle Fracture & Thermal Stress, as the basis for the lower temperature in Mode 6 being more severe.

**NRC RESOLUTION:**

The NRC disagrees with the licensee’s recommendation.

Safety Limit TS 2.1.2 applies in Modes 1, 2, 3, 4, and 5, and requires that RCS pressure shall be maintained ≤ 2375 psig as defined on page 2.0-1 of Farley Units 1 and 2, Amendment No. 151 (Unit 1) and Amendment No. 143 (Unit 2). Based on this reference it can be seen that Mode 6 is not defined for this SL. Technical Specification 2.1.2, Bases (page B 2.1.2-2) further identifies under the APPLICABILITY paragraph that “SL 2.1.2 applies in MODES 1, 2, 3, 4 and

5 because this SL could be approached or exceeded in these MODES due to over-pressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized." Table 1.1-1 Definitions, identifies MODE 6 as "Refueling [c)]." Where c) is further defined as "one or more vessel head closure bolts less than fully tensioned. The licensee's inclusion of the General Physics reference material clouds the issue as to why Distractor 'D' should be considered as the only correct answer and not Distractor 'C.'

The licensee suggests in their discussion that the applicants were applying the actual pressure increasing above the 2375 psig SL. The question clearly asks if the SL TS is exceeded which of the distractors would be more severe and the reason. The applicants are expected to know the requirements for TS with actions of 1 hour or less. In this case, the TS SL does not apply in MODE 6. If a violation of SL 2.1.2 occurred, then the applicants were expected to know, from memory, that in accordance with 2.2.2.1, while in MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour, or 2.2.2.2, while in MODE 3, 4 or 5, restore compliance within 5 minutes.

On page B2.1.2-3 under the SAFETY LIMIT VIOLATIONS it states "If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RSC pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile."

Since TS's does not identify a pressure SL in TS 2.1.2 for Mode 6 it therefore cannot apply.

The NRC maintains that distractor 'C' is the only correct answer.

## SIMULATOR FIDELITY REPORT

Facility Licensee: Farley Nuclear Power Plant

Facility Docket Nos.: 05000348/05000364

Operating Tests Administered: October 23 - 26 and November 21, 2006

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

No simulator fidelity or configuration items were identified.

December 18, 2006

Southern Nuclear Operating Company, Inc.
ATTN: Mr. H. L. Sumner
Vice President
P. O. Box 1295
Birmingham, AL 35201

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC EXAMINATION REPORT 05000348/2006301 AND 0500364/2006301

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Docket Nos. 50-348, 50-364
License Nos. NPF-2, NPF-8
Enclosures: (See next page)

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