May 20, 2003

Mr. Roy Anderson
Chief Nuclear Officer and President
PSEG Nuclear LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION SENIOR REACTOR OPERATOR LIMITED TO FUEL HANDLING INITIAL EXAMINATION REPORT NO. 50-354/03-301

Dear Mr. Anderson:

This report transmits the results of the Senior Reactor Operator Limited to Fuel Handling (LSRO) licensing examination conducted by the NRC during the period of April 1- 3, 2003. This examination addressed areas important to public health and safety and was developed and administered using the guidelines of the “Examination Standards for Power Reactors” (NUREG-1021, Revision 8, Supplement 1).

Based on the results of the examination, one LSRO applicant passed all portions of the examination. Two LSRO applicants failed the discussion scenario portion of the operating examination and one of these applicants also failed the written examination. Mr. A. Blamey discussed performance insights observed during the examination with Mr. J. Reid on April 3, 2003. On May 5, 2003, final examination results were given during a telephone call between Mr. A. Blamey and Mr. K. Krueger.

In accordance with 10 CFR 2.790 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). These records include the final examination and are available in ADAMS ({LSRO} Written - Accession Number ML031130318; {LSRO} Operating Section A - Accession Number ML031130629; {LSRO} Operating Section B - Accession Number ML031130710; and {LSRO} Operating Section C - Accession Number ML031130730). 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 regarding this examination, please contact me at (610) 337-5183, or by E-mail at RJC@NRC.GOV.

Sincerely,

/RA/

Richard J. Conte, Chief
Operational Safety Branch
Division of Reactor Safety

Docket Nos. 50-354
License No. NPF-57

Enclosure: Initial Examination Report No. 50-354/03-301

cc w/encl:
M. Friedlander, Director - Business Support
J. Carlin, Vice President - Engineering
D. Garchow, Vice President - Projects and Licensing
G. Salamon, Manager - Licensing
T. O'Connor, Vice-President - Operations
R. Kankus, Joint Owner Affairs
J. J. Keenan, Esquire
Consumer Advocate, Office of Consumer Advocate
F. Pompper, Chief of Police and Emergency Management Coordinator
M. Wetterhahn, Esquire
N. Cohen, Coordinator - Unplug Salem Campaign
E. Gbur, Coordinator - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance
State of New Jersey
State of Delaware

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S. Barber, DRP
T. McGinty, OEDO
J. Clifford, NRR
G. Wunder, PM, NRR
R. Fretz, Backup PM, NRR
J. White, INPO whitejl@inpo.org
W. Lanning, DRS
R. Crlenjak, DRS
R. Conte, DRS
A. Blamey, Chief Examiner, DRS
C. Buracker, DRS (Master Exam File)
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No(s):	50-354
License No:	NPF-57
Report No:	50-354/03-301
Licensee:	PSEG LLC
Facility:	Hope Creek Nuclear Generating Station
Dates:	April 7, 2002 (Written Examination Administration) April 1 - 3, 2002 (Operating Test Administration) April 11 - 18, 2002 (Examination Grading)
Examiners:	A. Blamey, Senior Operations Engineer (Chief Examiner) G. Johnson, Operations Engineer (under instruction)
Approved by:	Richard J. Conte, Chief Operational Safety Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000354/03-301; On April 1 - 3, 2003 and April 7, 2003; Hope Creek Nuclear Generating Station; Initial Operator Licensing Examination. One of three LSRO applicants passed all portions of the examination.

The written examination was administered by the facility and the operating examination was administered by two NRC region-based examiners, one was under instruction. There were no inspection findings of significance associated with the examination.

A. Inspector Identified Findings

No findings of significance were identified.

B. License Identified Findings

No findings of significance were identified.

Report Details

1. REACTOR SAFETY

Mitigating Systems - Reactor Operator (RO), Senior Reactor Operator (SRO) Initial License Examination

a. Scope of Review

The NRC examination team reviewed the written and operating initial examinations and post examination comments to verify or ensure, as applicable, the following:

- The examination was prepared and developed in accordance with the guidelines of Revision 8, Supplement 1 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." A review was conducted both in the Region I office and at the Hope Creek plant and training facility. Final resolution of comments and incorporation of test revisions were conducted during and following the onsite preparation week.

- The operation of the Refueling Bridge was proper.

- A test item analysis was completed on the written examination for feedback into the systems approach to training program.

- Examination security requirements were met.

The NRC examiners administered the operating portion of the examination to all applicants from April 1 - 3, 2003**.** The written examination was subsequently administered by the Hope Creek training staff on April 7, 2003**.** The exam was originally scheduled for the week of March 17, 2003, but was rescheduled to April 1, 2003, to allow for completion of a refueling bridge modification. This was documented in PSEG notification 20137321.

b. Findings

Grading and Results

There were a total of three Senior Reactor Operators Limited to Fuel Handling (LSRO) applicants who took the initial licensing examination. One of the three applicants passed all portions of the examination. Two LSRO applicants did not pass the discussion scenario portion of the operating examination and one of these applicants also failed the written examination. Therefore, these two individuals are denied a license at this time**.**

Examination Preparation and Quality

The submitted examination was within the acceptable range.

There was one post-written examination comment submitted by PSEG. The NRC disagreed with the facility's comment. The facility's comment is listed in attachment 2 and the NRC resolution of the comment is provided in attachment 3.

Examination Administration and Performance

No findings of significance were identified.

4OA6 Exit Meeting Summary

On May 5, 2003, the NRC provided conclusions and examination results to Hope Creek management representatives via telephone. The LSRO license for the one applicant who passed all portions of the examination will be withheld pending completion of his required six months of site specific experience. Hope Creek was informed that when the NRC is notified, in writing, that the applicant has completed the required six months of site specific experience, his license will be issued. The other two applicants did not pass all portions of the examination and therefore are denied a license at this time.

The NRC expressed appreciation for the cooperation and assistance that was provided during the preparation and administration of the examination by the licensee's training staff.

ATTACHMENT 1

KEY POINTS OF CONTACT

LICENSEE

Archie Faulkner	Exam Development Supervisor
Jim Reid	Acting - Manager, Nuclear Training
Kurt Krueger	Operations Manager

NRC

Alan Blamey	Senior Operations Engineer
Gilbert Johnson	Operations Engineer (under instruction)

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

ITEM NUMBER	TYPE	DESCRIPTION
NONE		

LIST OF ACRONYMS

CRD	Control Rod Drive
CST	Condensate Storage Tank
LSRO	Senior Reactor Operator Limited to Fuel Handling
RACS	Reactor Auxiliary Cooling System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Clean up
SACS	Safety Auxiliary Cooling System

ATTACHMENT 2

LICENSEE COMMENTS ON THE WRITTEN EXAM

Question 10

Given the following conditions:

- The plant is in Operational Condition 4 following a forced shutdown 16 hours ago.
- RHR Loop "A" operating in Shutdown Cooling.
- The "B" RHR pump is Cleared & Tagged for motor replacement.
- The "A" RHR pump develops a high vibration and trips on overcurrent.
- HC.OP-AB.RPV-0009, Shutdown Cooling, is entered.

Which of the following will be adequate to maintain Operational Condition 4?

a. Crosstie "C" or "D" RHR pump for heat removal.
b. Maximize RWCU bottom head drain flow.
c. Raise level to +80 inches using natural circulation for heat removal.
d. Inject with Core Spray from the CST to the RPV.

Correct Answer: a. Crosstie "C" or "D" RHR pump for heat removal.

Licensee Comment: Recommended accepting answer choices "a" or "c" as correct answers.

Answer choice "c" states "Raise level to +80 inches using natural circulation for heat removal."

Natural circulation removes decay heat from the fuel bundles in the core to the bulk coolant. RWCU, C RHR, D RHR, or Condensate Transfer can be used to remove decay heat from the bulk coolant to the Main Condenser, Reactor Auxiliary Cooling System (RACS), or Safety Auxiliary Cooling System (SACS).

Abnormal procedure HC.OP-AB.RPV-0009 Condition E, action step E.2 states "Maintain Reactor Pressure Vessel (RPV) level greater than or equal to 80 inches but less than or equal to 90 inches." This step is performed if forced circulation cannot be established using preferred Residual Heat Removal (RHR) loops (A or B) or reactor recirculation. This step is performed when Reactor Water Clean Up (RWCU), C RHR, D RHR, or condensate transfer is required for alternate decay heat removal. The conditions of the stem require alternate decay heat removal methods to be used.

Condition E, action E.5 states "Evaluate the following systems for alternate decay heat removal:

- RWCU (subsequent F)
- C RHR (Attachment 1)
- D RHR (Attachment 2)
- Condensate Transfer (Subsequent G)"

The stem does not provide core exposure history other than shutdown 16 hours ago. The applicants could assume Beginning of Core Life, End of Core Life or anywhere in between. Since the stem does not rule out RWCU or condensate transfer operation, RWCU can be assumed in service and can be used in conjunction with natural circulation once reactor level has been raised to 80 to 90 inches. The heat removal means is natural circulation removing heat from the fuel bundles to the bulk reactor coolant, then to RACS and the main condenser. Under normal operation RWCU is rejecting 69 gpm from Control Rod Drive (CRD) injection, with some heat removed through RACS and some removed by replacement water from CRD. Based on stem conditions, RWCU is required for alternate decay heat removal. RWCU is realigned in accordance with subsequent action step F, which opens the cooling water supply valve ED-V035 full open and bypasses the non regenerative heat exchanger.

Heat removal capability in alternate decay heat removal mode is approximately 15 to 16 Million BTU's per hour. (Per System Engineering). Reactor decay heat load at the Beginning of Core Life during the initial startup from a typical 30 day refueling outage is approximately 13 Million BTUs per hour and rises with full power operation history. If a reactor automatic shutdown was assumed to occur during a startup from a refueling outage, before the reactor had any significant full power operation, the decay heat load 16 hours after the automatic shutdown would be well within RWCU Alternate Decay Heat Removal capability. Therefore, answer choice "c" would also be correct.

The applicants are not required to know the value of BTU's per hour removal rate, or the BTU generation rate of the core at a particular time of core life. From a procedure user point of view, answer choice "c" is also correct when applied to Subsequent Action step E.

Recommended action is to accept answer choices "a" or "c" as correct answers.

ATTACHMENT 3

NRC RESOLUTION OF LICENSEE COMMENTS

Written Question: 10

Comment: The question provides a condition in which the plant has been shutdown for 16 hours when shutdown cooling is lost. The applicant must determine which condition will be adequate to maintain the plant in operational condition 4, in accordance with HC.OP-AB.RPV-0009, "Shutdown Cooling." The correct answer was "a" to crosstie "C" or "D" residual heat removal (RHR) pump for core decay heat removal. Answer "c" Raise level to + 80 inches using natural circulation for heat removal was recommended to also be accepted as a correct answer. The basis for accepting answer "c" is that the reactor water clean up system would be in service with a normal line up and rejecting water to the condenser at 69 gpm following a reactor shutdown. If the plant has just started up following a 30 day refueling outage (typical length of time) then the decay heat load would be approximately 13 million BTU per hour. The reactor water clean up (RWCU) heat exchanger, in the alternate decay heat removal lineup, will have a capacity of 15 to 16 million BTUs per hour. Therefore, the plant will be able to be maintained in operational condition 4, after increasing level to + 80 inches and reconfiguring the RWCU system to the alternate decay heat removal mode of operation in accordance with the procedure.

NRC Resolution: The only correct answer is "a," crosstie "C" or "D" RHR pump for heat removal. This is based on the stem of the question which states that "Which of the following will be adequate to maintain Operational Condition 4." Answer "a" is the only answer that is sufficient to have enough heat removal capacity to maintain the plant in operational condition 4 and allowed by procedure HC.OP-AB.RPV-0009, Shutdown Cooling. Answer "c" states raise level to +80 inches using natural circulation for heat removal. This action is allowed by procedure HC.OP-AB.RPV-0009, Shutdown Cooling, and may initially keep the plant in operational condition 4 do to the additional heat capacity of the water used to increase level to + 80 inches. However, without removing the decay heat from the reactor pressure vessel the water will heat up and the plant will not be able to be maintained in operational condition 4. Raising level to + 80 inches only enhances natural circulation and heat transfer from the fuel to the coolant, but it will not remove the decay heat from the rector pressure vessel. Other system(s) will be required in addition to raising level, to remove the decay heat from the reactor pressure vessel. Therefore, "a" is the only correct answer.