UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

June 25, 2010

Mr. Peter T. Dietrich
Site Vice President
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC EXAMINATION REPORT 05000333/2010301

Dear Mr. Dietrich:

On June 11, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an examination at your James A. FitzPatrick Nuclear Power Plant (FitzPatrick). The enclosed report documents the examination findings, which were discussed on June 11, 2010, with Mr. Joseph Barnes, Training Director, and other members of your staff.

The examination included the evaluation of two applicants for reactor operator licenses, four applicants for instant senior operator licenses, and three applicants for upgrade senior operator licenses. The written and operating examinations were developed by the NRC using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The license examiners determined that eight of the nine applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses were issued on June 11, 2010. In addition, one applicant for an instant senior operator license passed his exam but his license is being held as explained in paragraph D.3.c of Examination Standard (ES) 501 in NUREG-1021 until the facility has certified in writing that the applicant has completed all of his experience requirements.

No findings of significance were identified during this examination.

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).

Sincerely,

Samuel L. Hansell, Jr., Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-333
License No. DPR-59

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Sincerely,
/RA/

Samuel L. Hansell, Jr., Chief
Operations Branch
Division of Reactor Safety

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NRC Examination Report 05000333/2010301

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DRS File

EXAMINATION REPORT

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Dockets:	50-333
Licenses:	DPR-59
Report :	05000333/2010301
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Dates:	May 3 - 7, 2010 (Operating Test Administration) May 25, 2010 (Written Examination Administration) May 27, 2010 (Licensee Submitted Post Exam Package) May 10 - June 10, 2010 (NRC Examination Grading) June 11, 2010 (Licenses Issued)
Inspectors:	John Caruso, Chief Examiner, Senior Operations Engineer Peter Presby, Operations Engineer Tom Hedigan, Operations Engineer Bernard Litkett, Operations Engineer (Certification Exam)
Approved By:	Samuel L. Hansell, Jr., Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000333/2010301; May 3-25, 2010; James A. FitzPatrick Nuclear Power Plant; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of two applicants for reactor operator licenses, four applicants for instant senior operator licenses, and three applicants for upgrade senior operator licenses at James A. FitzPatrick Nuclear Power Plant. 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 facility on May 25, 2010. NRC examiners administered the operating tests on May 3-7, 2010. The license examiners determined that eight of the nine applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued. In addition, one applicant for an instant senior operator license passed his exam but his license is being held as explained in paragraph D.3.c of Examination Standard (ES) 501 in NUREG-1021 until the facility has certified in writing that the applicant has completed all of his experience requirements.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

The examiners reviewed all nine license applications submitted by the licensee to ensure the applications reflected that each applicant satisfied relevant license eligibility requirements. The applications were submitted on NRC Form 398, "Personal Qualification Statement," and NRC Form 396, "Certification of Medical Examination by Facility Licensee." The examiners also audited six 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 of significance were identified.

.2 Operator Knowledge and Performance

a. Examination Scope

On May 25, 2010, the licensee proctored the administration of the written examinations to all nine applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on May 27, 2010.

The NRC examination team administered the various portions of the operating examination to all nine applicants on May 3-7, 2010. The two applicants for reactor operator licenses participated in two dynamic simulator scenarios, in a control room and facilities walkthrough test consisting of 11 system tasks, and an administrative test consisting of four administrative tasks. The four applicants seeking an instant senior operator license participated in two or three dynamic simulator scenarios, a control room and facilities walkthrough test consisting of 10 system tasks, and an administrative test consisting of five administrative tasks. The three applicants for upgrade senior operator licenses participated in one or two dynamic simulator scenarios, a control room and facilities walkthrough test consisting of 10 system tasks, and an administrative test consisting of five administrative tasks.

b. Findings

All nine of the applicants passed all parts of the operating test. For the written examinations, both of the reactor operator applicants' scores averaged 81.33 percent, the senior operator applicants' average score was 89.42 percent and ranged from 85 to 93 percent. The overall written examination average was 87.62 percent. The text of the examination questions and the licensee's examination analysis may be accessed in the ADAMS system under the accession numbers noted in the attachment.

Chapter ES-403 and Form ES-403-1 of NUREG 1021 require the licensee to analyze the validity of any written examination questions that were missed by half or more of the applicants. The licensee conducted this performance analysis for four questions that met these criteria and submitted the analysis to the chief examiner. The analysis concluded that all four of these questions were technically accurate as written and no post exam comments were submitted.

.3 Initial Licensing Examination Development

a. Examination Scope

The NRC developed the examinations in accordance with NUREG-1021, Revision 9, Supplement 1. All licensee facility training and operations staff involved in examination preparation and validation were listed on a security agreement. The NRC conducted an onsite validation of the operating examinations the week of April 5, 2010. The NRC satisfactorily completed comment resolution on April 23, 2010, for the Operating examination and on May 19, 2010, for the written examination.

b. Findings

No findings of significance were identified.

.4 Simulation Facility Performance

a. Examination Scope

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

b. Findings

No findings of significance were identified.

.5 Examination Security

a. Examination Scope

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

b. Findings

A potential compromise occurred for the Reactor Operator (RO) and Senior Reactor Operator (SRO) “draft” administrative Job Performance Measures (JPMs) topics during examination development. The potential compromise occurred when an NRC examiner transmitted the RO and SRO “draft” administrative JPM Outlines over the internet in a non-password protected email. Replacement administrative JPM topics were selected and replacement JPMs developed in order to ensure a valid examination.

4OA6 Meetings, Including Exit

The chief examiner presented the examination results to Mr. Joseph Barnes, Training Director, and other members of your staff on June 11, 2010. The licensee acknowledged the findings presented.

The licensee did not identify any information or materials used during the examination as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Barnes, Training Director
C. Adner, Operations Manager
D.J. Russell, Operations Training Superintendent for Initial Training
J. Burton, Requalification Program Supervisor
M. Emrich, Nuclear Training Instructor
M. Needles, Nuclear Training Instructor

NRC Personnel

G. Hunegs, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Closed

NONE

Discussed

NONE

ADAMS DOCUMENTS REFERENCED

Accession No. ML101740446 – FINAL-Written Exam
Accession No. ML101740486 – FINAL-Operating Exam

Facility Licensee: James A. Fitzpatrick Station
Facility Docket No.: 50-333
Operating Test Administered on: May 3-7, 2010

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.

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

Item	Description
1	Unable to run Turbine Building Ventilation Isolation/Reset in the scenarios to test EOP-6 flow path. This feature is necessary to exercise Emergency Operating Procedures (EOPs).
2	During exam validation, a licensed Senior Reactor Operator (SRO) indicated that he could not retrieve multiple Emergency and Plant Information Computer (EPIC) points at the same time in the simulator but could perform this task in the plant. This is a potential simulator fidelity issue.
3	No simulator malfunction exists for main steam flow instrumentation. The override used to trip relay 16A-K3A did not work.
4	Observed repeated EPIC system terminal lockups during exam scenarios, requiring soft reboots on the machines. Need to determine if this is an equipment problem, training issue, or simulator fidelity issue.
5	No simulator malfunction exists for any Primary Containment Isolation System (PCIS) instrumentation.

Item	Procedural Issues Identified	Attachment 3
1	• EAP-4.1, Release Rate Determination, Rev 19 • IAP-2, Classification of Emergency Conditions, Rev 28 Senior Reactor Operator (SRO) administrative JPM A-4, Determine Protective Action Recommendations (PAR) and Complete Event Notification Form. During exam validaton, the licensed SROs could not complete the PAR within 15 minutes. Most validators stated that making the Emergency Action Level (EAL) classification and PAR was the job of the Shift Manager. 55.59(a)(2)(ii) states the operating test will require the senior operator to demonstrate an understanding of and the ability to perform a comprehensive sample of items specified in 10 CFR 55.45(a)(2) through (13). 10 CFR 55.45(a)(2)(11) states, " Demonstrate knowledge of the emergency plan for the facility, including, as appropriate, the operator's or senior operator's responsibility to decide whether the plan should be executed and the duties under the plan assigned." EAP-4.1, Section 4.1.2.D, Page 8 of 26. Also, Attachment 1, Page 14 of 26. Procedure for conversion of dose meter readings into release rates provides generic instruction for determining noble gas and equivalent iodine release rates. It does not specify which conversions are required for PAR assessment. Facility feedback indicates noble gas rates should be used as the release rates for determining PARs. PAR chart in EAP-4 (Attachment 1) bases PAR on release rate (without identifying whether noble gas or iodine). Facility normally uses plant computer display release rate for PAR, which happens to be noble gas – although the computer also does not identify that it is noble gas. Note: four of the seven SRO applicants missed this job performance measure (JPM).	
2	• EOP-2; RPV Control The implemention of TERMINATE AND PREVENT guidance in the EOPs for a Loss of Coolant Accident (LOCA) by both the licensed operators during validation week and the applicants during the exam was incorrect. In the emergency depressurization leg of the EOP's there is a step that states, "Prevent injection from those residual heat removal (RHR) and Core Spray pumps not required to assure adequate core cooling (EP-5)." The performance of this step requires an evaluation of the size of the LOCA and what pumps are needed to assure adequate core cooling. The applicants' order to the balance of plant (BOP) operator to terminate all RHR and Core Spray pumps was incorrect because the pumps were needed to assure adequate core cooling.	

Item	Procedural Issues Identified	Attachment 3
3	• OP-16; Nuclear Instrumentation, Section E.1 OP-16 guidance for bypassing a flow unit is not clear. Caused a 30 minute delay during exam pre-validation. During scenario #1 the BOP operators in two different scenarios responded to a failed recirc. flow unit, marked the steps in OP-16, Section E.1 for bypassing a flow unit as N/A (not applicable). Neither applicant informed the SRO that the steps had been marked N/A. The steps were required to be performed to bypass the flow unit to allow the scram to be reset.	
4	• ARP 09-5-2-3, ROD DRIFT, was updated with guidance that is different than the guidance in AOP-27, Control Rod Drift (CR-JAF-2010-02463 previously written). The two procedures should be consistent.	
5	• AOP-25, Uncoupled Control Rod does not state entry into technical specification (TS) 3.1.3 when the control rod is uncoupled.	
6	• EOP-2; RPV Control Guidance in EOP's when requirements for Emergency Depressurization are met causes a significant delay when it is determined that water level cannot be maintained above -19 inches (but also requires level to be less than 0 inches before taking action). When the decision to emergency depressurize is made, the operator is challenged to open 7 ADS valves before water level drops below -19 inches.	
7	• OP-20; Standby Gas Treatment; Sections F.1 and F.2 During an inadvertent initiation of high pressure coolant injection (HPCI), the operator is required to manually shutdown standby gas treatment (SBGT) system; this action prevents an automatic re-initiation of SBGT system if another auto signal for SBGT is later received. During the exam, this led to confusion with the applicants whether to leave one train running or to secure both trains of SBGT. The procedure does not provide clear direction in this situation. The TS 3.6.4.3 for SBGT is a seven day LCO for one SBGT subsystem inoperable and TS 3.0.3 entry is required if both SBGT subsystems are inoperable. The note does not state the requirements for entry into TS 3.6.4.3 and does not provide direction for resetting the initiation signal for SBGT.	

Item	Procedural Issues Identified	Attachment 3
8	• AOP-24; Stuck Control Rod During a JPM (S-3) for a stuck rod (AOP-24), the applicants attempted to insert the control rod before raising drive pressure. The procedure should be enhanced to provide clear direction when a control rod cannot be moved with normal drive water pressure.	
9	• ST-5H; SRM Signal to Noise Ratio During conduct of the JPM (S-7) for ST-5H, SOURCE RANGE MONITOR (SRM) SIGNAL TO NOISE RATIO DETERMINATION TEST**, Step 8.2.5 states: "Ensure SRM may be fully withdrawn." Several applicants misinterpreted this to mean that this was a functional test, and this is not the case. In addition, Surveillance Test, ST-5N does not mention expected alarms.	
10	• RAP-7.3.03; Core Thermal Power Evaluation During conduct of RAP-7.3.03, CORE THERMAL POWER EVALUATION, the procedure identifies the formula but does not specify what each variable represents. The reactor analyst procedure (RAP) uses the 'old' steam table nomenclature. This procedure does not reflect the nomenclature used in the 'new' steam tables. The procedure also does not specify how many decimals to carry forward. Heat balance procedure RAP-7.3.03 does not reference how to calculate the enthalpy of compressed water. Step 9.2.4 states to use the American Society of Mechanical Engineers (ASME) steam tables to calculate items 24 through 30 on Attachment 1. The steam tables that were used for the JPM and that are available in the control room do not have compressed water tables. Items 26 through 30 values need to be obtained from compressed water tables, since the feedwater, reactor water cleanup, and control rod drive (CRD) are subcooled liquids. The procedure does not have a table with approved ASME values in the possible range of temperature and pressure for feedwater, cleanup, and CRD.	

Item	Procedural Issues Identified	Attachment 3
11	• EDG T/S 3.8.1 and TRM 3.7.P • OP-22; Diesel Generator Emergency Power The emergency diesel generator (EDG) TS does not contain the requirement for minimum fuel oil (FO) storage tank level. TRM 3.7.P does not specify required volume. Per UFSAR, Chapter 8 (page 8.6-2), "The Fuel Oil System" for each of the emergency alternating current (AC) power sources has the capacity to supply fuel to its respective emergency AC power source to operate it continuously at full load for seven days. OP-22 does NOT contain any normal operating log-type checks of FO Storage Tank level.	
12	• TRM 3.4.B; Reactor Coolant System (RCS) Chemistry TRM 3.4.B, RCS Chemistry has implementation information hidden in the bases document. High conductivity by itself is not a reason to enter the limiting condition of operations (LCO). This misinterpretation also existed in Fitzpatrick 2002 NRC Exam, Q# S86 (SRO).	
13	• EN-OP-103; Reactivity Management Program EN-OP-103, Reactivity Management Program, Rev. 3, Page 24 has a conflicting threshold for initiation of single notching during startup. Attachment 9.3, sheet 3, third bullet states notch withdrawal should be used once any SRM count rate reaches 10 times or four doublings until criticality is achieved. Four doublings is an SRM increase of factor of 16 which is greater than 10 times. There also appears to be a conflict in OP-65, Startup and Shutdown procedure that mentions only four doublings.	
14	• ST-9BB' 'B' and 'D' EDG Full Load Test and ESW Pump Operability Test ST-9BB directs EDG shutdown for 'emergencies' per OP-22, Section G.6, single engine shutdown. Rules of usage would delay shutdown of the other EDG if both are affected. In addition, the first step in Section G.6 doesn't apply. This ST does not contain emergency shutdown instructions. Finally, the single EDG shutdown method secures emergency service water (ESW) pump with second EDG still in service.	
15	• AOP-72; 115 KV Grid Loss, Instability, or Degradation The loss of 115KV, AOP-72 does not address the failure of both EDG output breakers to close. It has an override that says if you lose offsite power and scram, then exit this procedure. It appears that no procedural guidance exists for a loss of offsite power.	

Item	Procedural Issues Identified	Attachment 3
16	• OP-3A; Feedwater Heaters • AOP-62; Loss of Feedwater Heating OP-3A was revised to provide new reactor power reduction guidance when isolating a feedwater heater string which was not incorporated into AOP-62, Loss of Feedwater Heating. The new guidance in OP-3A requires reducing reactor power to 25 percent prior to isolating a feedwater heater string.	
17	• OP-19; Reactor Core Isolation Cooling (RCIC) OP-19, Posted Attachment 7, step A.5.2. EOP protected step 14, states monitor torus water level, temperature, and pressure; however, no actual limits are listed on the attachment.	
18	• Thermal Hydraulic Instability (THI) Guidance Operator guidance for THI may need enhancement. During exam validation a crew of licensed operators incorrectly inserted a manual scram on an electrohydraulic control (EHC) pressure oscillation because they believed THI was occurring at 90 percent reactor power.	
19	• SP-01.05, Wastewater Sampling and Analysis, Rev. 10 Admin JPM A-1-1 SRO, A-3 RO the required procedure forms were difficult to read, resulting in errors being made by the validators.	
20	• OPG-13; Transient Mitigation OPG-13 guidance for implementing EOPs states, the first entry *SHOULD* be announced. Is it acceptable for the control room supervisor (CRS) to <u>not</u> announce first entry or subsequent entries? It appears like this is a *SHALL*.	
21	• OP-15; HPCI During the exam HPCI was restarted if needed during the anticipated transient without scram (ATWS) at full flow setting. Enhanced guidance may be needed to direct slow introduction of injection flow during an ATWS.	

List of Acronyms

AC	Alternating Current
ADS	Automatic Depressurization System
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
CRD	Control Rod Drive
CRS	Control Room Supervisor
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EHC	Electrohydraulic Control
EOP	Emergency Operating Procedure
EP-5	Emergency Procedure 5
EPIC	Emergency and Plant Information Computer
ES	Examination Standard
ESW	Emergency Service Water
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
FO	Fuel Oil
HPCI	High Pressure Coolant Injection
JPM	Job Performance Measure
LCO	Limiting Condition of Operations
LOCA	Loss of Coolant Accident
MSIVs	Main Steam Isolation Valves
NRC	U. S. Nuclear Regulatory Commission
PAR	Protective Action Recommendations
PARS	Publicly Available Documents
PCIS	Primary Containment Isolation System
RAP	Reactor Analyst Procedure
RCS	Reactor Coolant System
RHR	Residual Heat removal
RCIC	Reactor Core Isolation Cooling
RO	Reactor Operator
SBGT	Standby Gas Treatment
SRM	Source Range Monitor
SRO	Senior Reactor Operator
ST	Surveillance test
THI	Thermal Hydraulic Instability
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
TRM	Technical Requirement Manual