UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

March 2, 2015

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000259/2015301, 05000260/2015301, 05000296/2015301

Dear Mr. Shea:

During the period January 19 – 22, 2015, the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Browns Ferry Nuclear Plant. 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 January 28, 2015.

All applicants passed both the operating test and written examination. There were four post-administration comments concerning the operating test. 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.

The initial examination submittal was within the range of acceptability expected for a proposed examination. All examination changes agreed upon between the NRC and your staff were made according to NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1.

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 the NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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) 997- 4662

Sincerely,

/RA/

Eugene F. Guthrie, Chief
Operations Branch 2
Division of Reactor Safety

Docket Nos: 50-259, 50-260, 50-296
License Nos: DPR-33, DPR-52, DPR-68

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

cc: Distribution via Listserv

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ADAMS: X ☐ Yes ACCESSION NUMBER: _ML15062A501_______ X☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS			
SIGNATURE	**DRL2**	**JXV3**	**BLC2**	**EFG**			
NAME	LANYI	VIERA	CABALLERO	GUTHRIE			
DATE	2/24/2015	2/25/2015	2/24/2015	3/2/2015	3/ /2015	3/ /2015	3/ /2015
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\OLEXAMS\BROWNS FERRY EXAMINATIONS\BF 2015-301 INITIAL EXAM (BRUNO)\CORRESPONDENCE\BROWNS FERRY 2015-301 EXAMINATION REPORT FINAL.DOCX

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-259, 50-260, 50-296
License No.:	DPR-33, DPR-52, DPR-68
Report No.:	05000259/2015301, 05000260/2015301, 05000296/2015301
Licensee:	Tennessee Valley Authority (TVA), LLC
Facility:	Browns Ferry Nuclear Plant, Units 1, 2, and 3
Location:	Athens, AL 35611
Dates:	Operating Test – January 19-22, 2015 Written Examination – January 28, 2015
Examiners:	Bruno Caballero, Chief Examiner, Senior Operations Engineer David Lanyi, Senior Operations Engineer Joe Viera, Operations Engineer
Approved by:	Eugene F. Guthrie, Chief Operations Branch 2 Division of Reactor Safety

SUMMARY

ER 05000259/2015301, 05000260/2015301, 05000296/2015301; operating test January 19 – 22, 2015 & written exam January 28, 2015; Browns Ferry Nuclear Plant; 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 Browns Ferry Nuclear Plant staff developed both the operating tests and the written examination. The initial operating test, written RO examination, and written SRO examination submittals met the quality guidelines contained in NUREG-1021.

The NRC administered the operating tests during the period January 19 – 22, 2015. Members of the Browns Ferry Nuclear Plant training staff administered the written examination on January 28, 2015. All six Reactor Operator (RO) and three Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. All nine applicants were issued licenses commensurate with the level of examination administered.

There were four post-examination comments related to the operating examination.

No findings were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Operator Licensing Examinations

a. Inspection Scope

The NRC evaluated the submitted operating test by combining the scenario events and JPMs in order to determine the percentage of submitted test items that required replacement or significant modification. The NRC also evaluated the submitted written examination questions (RO and SRO questions considered separately) in order to determine the percentage of submitted questions that required replacement or significant modification, or that clearly did not conform with the intent of the approved knowledge and ability (K/A) statement. Any questions that were deleted during the grading process, or for which the answer key had to be changed, were also included in the count of unacceptable questions. The percentage of submitted test items that were unacceptable was compared to the acceptance criteria of NUREG-1021, "Operator Licensing Standards for Power Reactors."

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 administered the operating tests during the period January 19 – 22, 2015. The NRC examiners evaluated six Reactor Operator (RO) and three Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. Members of the Browns Ferry Nuclear Plant training staff administered the written examination on January 28, 2015. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the [plant name], met the requirements specified in 10 CFR Part 55, "Operators' Licenses."

The NRC evaluated the performance or fidelity of the simulation facility during the preparation and conduct of the operating tests.

b. Findings

No findings were identified.

The NRC developed the written examination sample plan outline. Members of the Browns Ferry Nuclear Plant training 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. 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 determined, using NUREG-1021, that the licensee's initial examination submittal was within the range of acceptability expected for a proposed examination.

No issues related to examination security were identified during preparation and administration of the examination.

All applicants passed both the operating test and written examination and were issued licenses. Six RO applicants and three SRO applicants passed both the operating test and written examination.

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 four post-examination comments concerning the operating test. A copy of the final written examination and answer key, with all changes incorporated, may be accessed not earlier than February 6, 2017, and a copy of the licensee's post-examination comments may be accessed in the ADAMS system (ADAMS Accession Number(s) ML15040A589, ML15040A581, and ML15040A593.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 22, 2015, the NRC examination team discussed generic issues associated with the operating test with Keith Polson, Site Vice President, and members of the Browns Ferry Nuclear Plant staff. The examiners asked the licensee if any of the examination material was proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee personnel

Keith Polson, Site Vice President
Daniel L. Hughes, Director of Operations
Aaron S. Bergeron, Training Director
Chris L. Vaughn, Operations Training Manager
Donald C. Binkley, Initial License Training Supervisor
Michael Barton, Exam Developer
Keith Nichols, Operations Exam Representative
Tommy Albright, Simulator Manager
Russell Joplin, Corporate Exam Program Manager
Jim Stone, Site Licensing
Jamie Paul, Site Licensing
Todd Anderson, Quality Assurance

NRC personnel

David Dumbacher, Senior Resident Inspector

FACILITY POST-EXAMINATION COMMENTS AND NRC RESOLUTIONS

A complete text of the licensee's post-examination comments can be found in ADAMS under Accession Number ML15040A593.

Item #1: Scenario 2, Critical Task #2, After all RPV Level instruments flash (level unknown), inject into the RPV with all available sources until one of the conditions in C-4, Note 7 is met.

Post-Examination Comment

The licensee contended that "any or all" of the systems listed in C4, Reactor Flooding, were allowed to be used to flood the reactor, that is, the applicants were not required to simultaneously use ALL of the systems at one time. The licensee contended that the critical task should be re-worded as: "*After all RPV level instruments flash (level unknown), flood the reactor with any or all of the listed methods to the elevation of the main steam lines as indicated by C-4, Note 7.*"

NRC Resolution

The licensee's recommendation was accepted.

Once the SRVs were opened to depressurize the RPV, injection systems were required to be aligned to flood the RPV to the elevation of the main steam lines in accordance with Step C4-25 in EOI-C-4, Reactor Flooding, which stated:

FLOOD the RPV to the elevation of the main steam lines with the following:

INJ SOURCE	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available	5C	1200 psig
HPCI with CST suction if available	5D	1200 psig
CNDS	6A	480 psig
LPCI	6B, 6C	320 psig
CS	6D, 6E	330 psig
Stby coolant	7D	160 psig
RHR crosstie to other units	7C	320 psig
Fire Protection system	7K	150 psig
CNDS transfer pumps to RHR and CS	7A	110 psig
RHR drain pumps	7E, 7F	50 psig
PSC head tank pumps	7G	30 psig
RCIC (aux boiler steam) with CST suction if available	7H	1200 psig
RCIC manual start	EDMG-24 APPX B	1200 psig
HPCI (aux boiler steam) with CST suction if available	7J	780 psig
SLC (boron tank)	7B	1450 psig
SLC (test tank)	7B	1450 psig
Portable pump	EDMG-24 APPX E	150 psig

C4-25

The licensee's bases document (EOIPM 0-V-J, Contingency #4, RPV Flooding Bases) discussion for Step C4-25 stated:

> *"Injection sources are aligned to flood the RPV and establish core cooling by submergence. The list of flooding methods includes all sources capable of injecting into the RPV, including the 'alternate injection subsystems' defined in EOIPM Section 0-I-C. Any or all of the listed methods may be used as necessary to flood the RPV to the elevation of the main steam lines."* (Emphasis added.)

The wording of Scenario 2, Critical Task #3 included the phrase "*inject into the RPV with all available sources*" which could potentially be misconstrued to imply that the applicants were required to simultaneously use all systems listed in Step C4-25. The licensee's proposed change to Critical Task #3 aligned with the BWR Owners' Group Emergency Procedure and Severe Accident Guidelines (February 2013, Revision 3), which allowed for any or all of the systems to be used to flood the reactor. Furthermore, the intent of the critical task was to evaluate the applicants' ability to recognize that all level instruments were unavailable, enter C-4, and flood the reactor to the elevation of the main steam lines while maintaining sufficient injection to maintain pressure on the reactor using any or all of the systems listed in Step C4-25. Therefore, the licensee's recommendation was accepted.

Item #2: Scenario 3, Critical Task #3, During an ATWS, with emergency depressurization required, terminate and prevent reactor pressure vessel (RPV) injection from ECCS and Feed Water until reactor pressure is below the minimum steam cooling pressure, as directed by the Unit Supervisor.

Post-Examination Comment

The licensee contended the following performance standard for Scenario 3, Critical Task #3, was only applicable when RPV level was > (-) 50 inches:

"*This Critical Task is not met if the Crew injects too fast and causes power oscillations or APRM downscale clear (>5% power).*"

The licensee recommended the performance standard for Critical Task #3 be re-worded to clarify this applicability as follows:

"*This Critical Task is not met if the crew fails to recognize and take action to secure injection to the reactor if the conditions of C5-5 or C5-15 are met.*"

NRC Resolution

The licensee's recommendation was accepted.

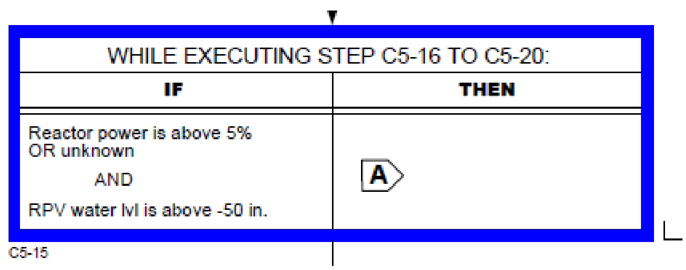
Scenario 3, Critical Task #3 evaluated the applicants' ability to emergency depressurize the reactor during a low power failure-to-scram (ATWS) event. Specifically, the applicants were expected to terminate and prevent injection and emergency depressurize the reactor until reactor pressure lowered to < 190 psig. As soon as reactor pressure lowered to < 190 psig, the applicants were expected to slowly recommence injection in accordance with the definition of "slowly" in EOIPM Section 0-V-K, Contingency #5, Level/Power Control Bases, and EOIPM Section 0-I-C, Glossary of Key Words and Terms:

Slowly: Perform a specified action in a careful, controlled manner, making incremental adjustments to avoid undesirable consequences that may result from rapid changes in

parameter values. Used specifically to identify the need for caution when restoring or increasing RPV injection under failure-to-scram conditions. If a potential for power excursions exists, RPV injection should be increased gradually while monitoring reactor power. If a sustained increase in reactor power is observed, the injection rate should be stabilized until reactor power is no longer increasing.

During the onsite preparatory week visit, the NRC exam team recommended the APRM downscale set point as the measurable performance indicator (See NUREG-1021, Appendix D, Section D.1.c) for Scenario 3, Critical Task #3, to evaluate the applicants' ability to slowly recommence injection. The BWR Owners' Group Emergency Procedure and Severe Accident Guidelines (February 2013, Revision 3) and the licensee's bases for Level/Power Control, used the plant APRM downscale set point as the threshold for a stabile margin with respect to core boiling boundary and reactor power. The licensee subsequently agreed and incorporated the recommendation into their final operating test submittal.

The performance standard wording for Critical Task #3 ["*This Critical Task is not met if the Crew injects too fast and causes power oscillations or APRM downscale clear (>5% power)*"] was only applicable when reactor water level was > (-) 50 inches because this was the point when power production challenged containment integrity. This is based on Step C5-15 (see below) which *allowed reactor power to be greater than 5% if reactor water level was ≤ (-) 50 inches*.



If reactor water level did not exceed (-) 50 inches, then no immediate threat to containment existed. If reactor water level exceeded (-) 50 inches, then C-5, Level/Power Control, required re-terminating injection and re-lowering reactor water level. The performance standard, as originally worded, did not encompass the difference between < or > than (-) 50 inches. Therefore, the licensee's recommendation was accepted.

Item #3: JPM 631, Restore Offsite Power to 4kV Shutdown Board at Unit 2 Panel 9-23

Post-Examination Comment

The licensee contended that JPM Step 16 [Procedure Step 17.1] was not a critical step even though it was designated as a critical step in the JPM. Specifically, the licensee contended that, since the Diesel Generator (DG) breaker was a Siemens 5KV breaker with a normal interrupting capability of 10,000 amps and a short circuit interrupting capability of 29,000 amps, JPM Step 16 was not a critical step.

NRC Resolution

The licensee's recommendation was accepted.

During the JPM the applicants were expected to reconnect offsite power to 4kV Shutdown Board “A”, while the DG was still providing power to the board. Afterwards, the applicants were expected to disconnect the DG from the Shutdown Board. JPM Step 16 [Procedure Step 17.1] was:

8.3 Restoring Offsite Power to 4-kV Shutdown Board at Panel 9-23 (continued)

[17] **WHEN** Parallel with System operation is no longer desired, **THEN**

UNLOAD the Diesel Generator as follows:

CAUTION

[II/C] When unloading the Diesel Generator, failure to slowly approach the 300 kW/250 kVAR limit may result in a reverse power trip of the Diesel Generator output breaker. [II-92-055]

[17.1] [II/C] **USE** the associated Diesel Generator's governor control switch and voltage regulator control switch to reduce generator load to approximately 300 kW and 250 kVAR: [II-92-055] ☐

The performance standard for JPM Step 16[Procedure Step 17.1] stated:

(Applicant) Unloads the DG to approximately 300 kW and 250 kVAR.

The licensee’s vendor manual (BFN-VTD-S106-0040) indicated that the normal interrupting capability was 10,000 amps, and the short circuit interrupting capability was 29,000 amps, which were above full load values, and far above the currents corresponding to the 300 kW and 250 kVAR values in JPM Step 16. Additionally, the DG output breaker is required to open when an accident signal is received following a loss-of-offsite-power (LOOP) event at BFN, which further demonstrates that the performance standard was not a critical step; no bases was found for an upper limit on kW or kVAR prior to opening the DG output breaker.

<u>Item #4</u>: JPM 631, Restore Offsite Power to 4kV Shutdown Board at Unit 2 Panel 9-23

<u>Post-Examination Comment</u>

The licensee contended that an unsafe plant condition was not created if the 4kV Shutdown <u>BUS</u> #1 Auto Transfer Lockout Relay was (incorrectly) placed in the MANUAL position instead of placing the 4kV Shutdown <u>Board</u> A Auto Transfer Lockout Relay to MANUAL because 1) there was no immediate consequence and 2) the risk analysis, using the equipment-out-of-service (EOOS) program, indicated that core damage frequency and large early release fraction both remained GREEN.

<u>NRC Resolution</u>

The licensee’s recommendation was not accepted.

During the JPM the applicants were expected to reconnect offsite power to 4kV Shutdown Board “A”, while the DG was still providing power to the board. Afterwards, the applicants were

expected to disconnect the DG from the Shutdown Board. During this process, the applicant was expected to perform JPM Step 2, which was NOT a critical step:

Step 2:

[2] **VERIFY** the associated 4kV shutdown board auto transfer lockout relay is tripped to MANUAL.

Diesel	Handswitch Name	Handswitch No.	Panel
A	4KV SD BD A AUTO/LOCKOUT RESET	0-211-3EA	0-9-23-7

Standard:

Verifies 4KV SD BD A AUTO/LOCKOUT RESET, HS-0-211-A, is tripped to manual.

JPM Step 2 was NOT a critical step because the 4kV Shutdown Board Auto Transfer Lockout Relay hand switch was already in the MANUAL position in the JPM initial conditions.

However, during administration of the operating exam, one applicant (incorrectly) placed the 4kV Shutdown BUS #1 Auto Transfer Lockout Relay Switch in the MANUAL position and never identified the mistake.

0-OI-57A, Switchyard and 4160V AC Electrical System, Illustration 6, Limiting Conditions, included a required compensatory action when the 4kV Shutdown BUS #1 automatic transfer feature was disabled.

LIMITING CONDITIONS ASSOCIATED WITH 4kV SHUTDOWN BOARDS AND BUSSES

Limiting Condition	AFFECTED BOARD OR BUSS				
	UB 1A	UB 1B	UB 2A	UB 2B	Shutdown Bd C
Shutdown Bd C on Alt feed (SD Bus 1)					Limit load to 340 amps May exceed 340 amps one time for max 100 hrs. (11)
SD Bus 1 on Alt feed				BK Auto Xfer to Start Bus (9)	
SD Bus 1 Auto Xfer Inop	BK Auto Xfer to Start Bus (8)				
SD Bus 2 on Alt feed		BK Auto Xfer to Start Bus (7)			
SD Bus 2 Auto Xfer Inop			BK Auto Xfer to Start Bus (10)		

Specifically, when the 4kV Shutdown BUS #1 automatic transfer feature was inoperable, the required compensatory action was to BLOCK the upstream Unit Board 1A automatic transfer feature to preclude overloading the 161 kV Common Station Service Transformer if a loss of the 500 kV Switchyard subsequently occurred. Additionally, when the 4kV Shutdown BUS #1 automatic transfer feature was disabled, the emergency diesel generators would be unnecessarily challenged if a subsequent loss of the 500 kV Switchyard occurred.

The licensee also contended that an unsafe plant condition was not created by the applicant because the equipment-out-of-service (EOOS) software program indicated that core damage frequency (CDF) and large early release fraction (LERF) both remained GREEN. NUREG 1021, Rev. 9, Supplement 1 does not provide guidance for identifying new critical steps in JPMs as they pertain to CDF or LERF. In accordance with NUREG 1021, every procedural step that the applicant must perform correctly to accomplish the task was identified as a “critical step” in the JPM; critical steps are identified based on procedural requirements, not risk. Likewise, incorrect or unexpected actions taken by an applicant during the performance of a JPM may

require further required actions (not previously identified in the JPM) to ensure the task is completed and/or the plant is left in a configuration that is not less safe than the initial conditions. The application of a risk analysis would require a much more in-depth analysis with respect to accident sequences that not only include the out-of-service equipment, but also other specific initiating events, equipment failures, and operator errors that transpire before, during, and after the operator task, to identify CDF and LERF. Nevertheless, the licensee's EOOS software risk analysis indicated that CDF and LERF were an order of magnitude higher (less safe) when the 4kV Shutdown BUS #1 automatic transfer feature was disabled because the further required action to BLOCK the upstream Unit Board was never implemented. Therefore, the licensee's recommendation was not accepted.

SIMULATOR FIDELITY REPORT

Facility Licensee: Browns Ferry Nuclear Plant

Facility Docket No.: 50-259, 260, and 296

Operating Test Administered: January 19 – 22, 2015

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
PER # 983938	During a scenario on the Unit 2 simulator, the crews were unable to establish CRD Cooling Water Header delta-P between 10-20 psid on PDI-85-18A.