UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
TENNESSEE VALLEY AUTHORITY	)	Docket Nos. 50-259
	)	50-260
(Browns Ferry Nuclear Plant units 1 and 2)	)	
	)	

AFFIDAVIT OF JACK R. CALHOUN

Jack R. Calhoun, being duly sworn, deposes and says: My business address is Tennessee Valley Authority, 702 Edney Building, Chattanooga, Tennessee. I am employed by the Tennessee Valley Authority as the Chief of the Nuclear Generation Branch, Division of Power Production. I am familiar with these proceedings and have personal knowledge of the matters contained herein.

Qualifications

I have been continuously employed by the Tennessee Valley Authority since 1949. Prior to that time I served for eight years in the United States Navy. Part of this time I was an Electrical Officer on the light cruiser USS Oklahoma City and the aircraft carrier USS Saratoga and was qualified as Engineering Officer-of-the Watch at sea on both ships.

I have received the Bachelor of Science degree in electrical engineering from Tennessee Technological University in 1949. During this period I was the Executive Officer and Electronics Officer of the U.S. Naval Reserve Electronics Warfare Company located at Cookeville, Tennessee.

I began my employment with TVA in 1949 as a student in the steam generating plant operator training program at the Watts Bar Steam Plant and later became an instructor in that program. I was transferred to the Johnsonville Steam Plant in 1952 as a unit operator and later assumed the position of an electrical engineer. In 1954 I was placed in charge of all electrical maintenance at the Johnsonville plant.

In 1958 I became assistant plant superintendent at the 1,500-MW Shawnee Steam Plant at Paducah, Kentucky.

In 1960 I became superintendent of the Experimental Gas-Cooled Reactor (EGCR) at Oak Ridge, Tennessee. During this period I attended the Oak Ridge School of Reactor Technology. In 1961 I spent five months at the Berkeley Nuclear Power Station in Bristol, England, assisting in the startup of that reactor. While at Berkeley I completed the reactor operator training course on a nuclear plant simulator used to train all reactor operators for the Central Electricity Generating Board.

In 1963 I was appointed assistant Project Manager of the Experimental Gas-Cooled Reactor and was responsible for assisting the project manager in all phases of technical and operational work.

From 1963 to 1966 I was a member (for reactor operation) of a panel created by an agreement between the United Kingdom Atomic Energy Authority and the United States Atomic Energy Commission to exchange information on gas-cooled reactors. As a member of this panel, I twice traveled to England to investigate and to observe the operation of the British Advanced Gas-Cooled Reactor in preparation for the startup of EGCR.

From February 1966 to February 1968, I held the position of Assistant to the Chief, Power Plant Maintenance Branch, Division of Power Production in TVA. I assisted in the engineering and coordination of the electrical and mechanical maintenance of all TVA steam and hydro plants. I was also responsible for the operation and maintenance planning relating to future TVA nuclear power plants.

From February 1968 to July 1971, I held the position of Plant Superintendent of the Browns Ferry Nuclear Plant in Athens, Alabama.

From July 1971 to April 1974, I was nuclear operations coordinator; and in April 1974 my title was changed to Chief, Nuclear Generation Branch. In this position I am responsible for and in charge of staffing, startup testing, and operations of all TVA nuclear power plants, including the Browns Ferry Nuclear Plant, units 1 and 2. I am also responsible for the coordination of the restoration and modifications activities, including fire protection improvements, of the Browns Ferry Nuclear Plant, units 1 and 2, following the March 22, 1975, fire.

I am presently a member of the Advisory Council at Pennsylvania State University (advisor to the Nuclear Engineering Department) and serve as Vice Chairman, Reactor Operations Division, American Nuclear Society.

I am familiar with this proceeding and have personal knowledge of the matters stated herein.

Statement

The modification and restoration of units 1 and 2 in accordance with TVA's "Plan for Evaluation, Repair and Return to Service of Browns Ferry, Units 1 and 2, (March 22, 1975, Fire)" have been substantially completed. Permission has been granted to load fuel in units 1 and 2.

All control rods have been fully inserted and electrically disarmed throughout fuel loading which is now complete for unit 2. Unit 1 currently has 659 fuel assemblies loaded. After refueling is complete on each unit, TVA proposes to conduct the following subcritical testing, which is a part of the startup retest program: Control Rod Drive System tests (Startup Test No. 5) scheduled at zero reactor pressure. The control rod drive tests proposed are position indication, insert/withdraw time, coupling, friction, and scram testing at zero reactor pressure. The proposed testing is a portion of the startup test program previously approved by the Nuclear Regulatory Commission, and will be conducted as described in the Browns Ferry Final Safety Analysis Report, Section 13.5 (pages 13.5-18 and 13.5-19) and Table 13.5-5 (Attachment 1), except for those changes discussed in Part XI, Section D, of TVA's "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2, (March 22, 1975, Fire)" (Attachment 2).

The purpose of conducting the Control Rod Drive System tests will be to determine initial operating characteristics of the Control Rod Drive System and to ensure that no control rod interference exists in the fully loaded core. On successful completion of these tests, TVA will install the reactor vessel head which will reduce the startup retesting period by approximately ten days when permission is granted to operate units 1 and 2.

This testing will require the operability and use of each control rod, one at a time. At all times during this testing, the remaining rods not in use will be fully inserted, valved out, and electrically disarmed; the RHR system will be aligned to cool the core in the reactor vessel; and all valves in lines which could drain the reactor vessel and the RHR system in this mode will be disabled in the position that will not drain the reactor. These conditions are in accordance with the conditions stipulated when permission was granted to load fuel in units 1 and 2.

Justification is presented below for TVA's position that this testing can be performed with reasonable assurance that the health and safety of the public will not be endangered, with no reliance placed on fire damaged equipment to maintain the reactor subcritical or to mitigate the consequences of an accident.

Prior to commencing the proposed testing, a verification that the fuel in each core is loaded correctly in the position it occupied prior to the fire is made by comparing videotapes of the loaded core with core maps generated following initial fuel loading of units 1 and 2. An affidavit by R. G. Cockrell (attachment 4) shows that this verification has been completed on the unit 2 core and that it will be conducted prior to the testing of unit 1.

Permission to load fuel in units 1 and 2 included the requirements that all control rods be fully inserted, valved out, and electrically disarmed; the RHR system be aligned to cool the core in the reactor vessel;

and all valves in lines which could drain the reactor vessel and the RHR system in this mode be disabled in the position which will not drain the reactor. In an affidavit of Thomas V. Wambach, of the Nuclear Regulatory Commission, dated May 5, 1976, Mr. Wambach made an evaluation of the safety of units 1 and 2 in the above described conditions, and concluded "that the core could be kept adequately cooled without reliance on any system which has been restored after fire damage or whose design has been modified in the restoration work." For the proposed testing the only plant condition that differs from the conditions required for fuel loading is that each control rod will be made operable, one at a time. At all times while a control rod is in use, all 184 rods not in use will be fully inserted, valved out, and electrically disarmed. During the proposed testing, the RHR system will be aligned to cool the core in the reactor vessel, and all valves in lines which could drain the reactor vessel and the RHR system will be disabled in the position that will not drain the reactor. A safety evaluation for the worst accident that could occur with one control rod operational and the resulting effects on the margin to criticality and ability to cool the core is presented below.

Evaluation of Effects of Placing One Control Rod in Operation

NRC has previously approved TVA's loading of fuel into the unit 1 and unit 2 cores provided all control rods are fully inserted, valved out, and electrically disarmed. Upon completion of this operation, all fuel will be in its correct location in the unit 1 and 2 cores, and all control rods will be fully inserted, valved out, and electrically disarmed. The only change in plant conditions between those described above and those

that would exist during the performance of the proposed control rod tests is that one control rod will be operational. All other control rods (184) will be fully inserted, valved out, and electrically disarmed. This provides positive assurance that these 184 rods will not be withdrawn from the core. In order to adversely affect the ability to cool the core and the ability to keep it covered, as described in Mr. Wambach's affidavit dated May 5, 1976, a malfunction of the fire damaged equipment placed in operation would have to be of such severity to make the reactor critical. The evaluation of the failure of the fire damaged systems used to conduct the tests proposed in this affidavit for each unit has been made and is presented below.

Systems needed to conduct the proposed testing are as follows: (1) Source Range Monitoring System, (2) Control Rod Drive Hydraulic System, and (3) Reactor Manual Control System.

Unit 2 Analysis

For unit 2, none of the components or equipment necessary for the conduct of the proposed testing were damaged by the fire or modified as a result of the fire. Therefore, the safety evaluation as described in Mr. Wambach's affidavit dated May 5, 1976, is valid and applicable, and this testing can be conducted with no reliance on fire damaged equipment. To prove that the reactor will always remain subcritical during the proposed testing, the following safety analysis was performed, in which the worst accident occurs which results in maximum reactivity insertion. The worst case accident is the failure of the one operational control rod such that it fails in the fully withdrawn position. For

this control rod system failure to result in the maximum increase in core reactivity, this safety analysis assumes that the failed rod is the analytically strongest rod (rod 26-07), as identified by the General Electric Company (see attachment 3). Unit 2 core average exposure is currently 2,165 MWd/t, and from attachment 3, the shutdown margin with the analytically strongest rod withdrawn is 3.15% Δk/k. This value was determined by General Electric Company's improved calculational methods recently reviewed by NRC. These methods result in calculations even more accurate than methods described in the Browns Ferry FSAR. The General Electric Company calculations are based on input data on the Browns Ferry core characteristics provided by TVA. In TVA's unit 2 technical specifications, a value of 0.38% Δk/k has been assigned to account for uncertainties in fuel content and uncertainties in calculating the analytically strongest rod. This value must be subtracted from any analytical determination of shutdown margin. After applying this 0.38% Δk/k uncertainty value, the core will still be subcritical by at least 2.77% Δk/k in the worst case accident, one in which the analytically strongest rod is fully withdrawn from the core. The withdrawal of any rod other than 26-07 will similarly not result in criticality. In the event that any accidental single rod movement occurs, there will still be no criticality in the reactor core and the test can be conducted without danger. Because the reactor will remain subcritical in the worst case accident, this testing can be conducted without endangering the safety of the public, with no reliance placed on fire damaged equipment or systems to maintain the reactor subcritical or to mitigate the consequences of an accident.

Unit 1 Analysis

For unit 1, as above for unit 2, the conditions that will exist during the proposed testing will be the same as those required for fuel loading with the exception that one control rod at a time will be operational. The ability to cool the core and to keep it covered will not be affected as long as the reactor remains subcritical throughout testing. Some of the systems to be used in the unit 1 testing were damaged by the fire and have been restored. For this analysis it will be assumed that this equipment fails and no credit will be taken for its ability to mitigate the consequences of the worst case accident which could occur during this testing. Effects of the failure of each of these systems on the margin to criticality during the worst accident, one in which the operational control rod fails in the fully withdrawn position, are analyzed below.

1. Source Range Neutron Monitoring System - With all rods not in use fully inserted, valved out, and electrically disabled, it is only possible to fully withdraw one rod from the core at a time, which would be the worst possible accident during this testing. From the attached Shutdown Margin Curve (Attachment 3) for 5,750 MWd/t exposure, the shutdown margin with the analytically strongest rod withdrawn is 2.10% $\Delta k/k$. Applying 0.38% $\Delta k/k$ for uncertainties in fuel content and uncertainties in calculating the analytically strongest rod, the reactor will reamin subcritical by at least 1.72% $\Delta k/k$. Since the reactor remains subcritical during the worst possible accident, the SRM system will not be needed to serve any safety actuation function during this testing. Therefore, this testing can be conducted without endangering the health and safety of the public, even if a complete failure of the SRM system occurs.

2. Control Rod Drive (CRD) Hydraulic System - Portions of the electrical cables for the CRD hydraulic system for unit 1 were damaged by the fire. The worst case accident that could occur if this system failed to function properly would be the full withdrawal of the one operational control rod. All other rods will remain inserted because of the fact that they will be valved out and electrically disabled. Assuming the worst case accident in which the analytically strongest rod fails in the fully withdrawn position, the reactor remains subcritical by at least 1.72% $\Delta k/k$. Therefore, it is concluded that this testing can be conducted without endangering the health and safety of the public, with no reliance placed on the proper operation of the CRD hydraulic system to maintain the reactor subcritical or to mitigate the consequences of an accident, provided all other rods are fully inserted, valved out, and electrically disarmed.

3. Reactor Manual Control System - With the mode switch in the refueling mode, the reactor manual control system prevents withdrawal of more than one rod at a time. However, portions of this system were damaged by the fire. By valving out and electrically disarming all rods other than the one in use in the fully inserted position, reliance on this system to ensure only one rod is withdrawn at a time is eliminated. Under this condition, the worst case accident that could occur if this system failed would be the full withdrawal of the one operational control rod. Assuming that this fully withdrawn rod is the analytically strongest rod, the reactor remains subcritical by at least 1.72% $\Delta k/k$. Therefore, it is concluded that this testing can be conducted without endangering the health

and safety of the public, without reliance on the proper operation of the Reactor Manual Control System to maintain the reactor subcritical or to mitigate the consequences of an accident, provided all other rods are fully inserted, valved out, and electrically disarmed.

Summary

Based on the above analyses, which show that the reactor will remain subcritical during the worst case accident, it is concluded that the proposed testing can be conducted without danger of uncovering the core or risk of serious accident, providing that (1) all control rods other than the one being tested are disabled in the fully inserted position, (2) the RHR system is aligned to cool the core in the reactor vessel, and (3) all valves in lines which could drain the reactor vessel and the RHR system in this mode are disabled in the position which will not drain the reactor. For this analysis, no credit was taken for fire damaged and restored equipment to maintain the reactor in the subcritical condition or to mitigate the consequences of the worst accident which could occur during the testing.

Jack R. Calhoun

Jack R. Calhoun

Subscribed and sworn to me

this 21st day of July, 1976

Carolyn B. Roberts

Notary Public

My commission expires 7-2-78

TEST NUMBER 5 – CONTROL ROD DRIVE SYSTEM

Purpose

The purposes of the Control Rod Drive System test are (a) to demonstrate that the Control Rod Drive (CRD) System operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and (b) to determine the initial operating characteristics of the entire CRD system.

Description

The CRD tests performed during Phases II through IV of the startup test program are designed as an extension of the tests performed during the preoperational CRD system tests. Thus, after it is verified that all control rod drives operate properly when installed, they are tested periodically during heatup to assure that there is no significant 58
binding caused by thermal expansion of the core components. A list of all control rod drive tests to be performed during startup testing is given below.

CONTROL ROD DRIVE SYSTEM TESTS

Test Description	Accumulator Pressure	Preop Tests	Reactor Pressure with Core Loaded psig (kg/cm²) 0	600 (42.2)	800 (56.2)	Rated
Position Indication		all	all			
56 Normal Times Insert/Withdrawn		all	all			4*
Coupling		all	all***			
Friction			all			4*
Scram	Normal	all	all	4*	4*	all
Scram	Minimum		4*			
Scram	Zero					4*
Scram (Scram Discharge Volume High Level)	Normal	4 (full core scram)				
Scram	Normal					4**

*Value refers to the four slowest CRD's as determined from the normal accumulator pressure scram test at ambient reactor pressure. Throughout the procedure, "the four slowest CRD's" implies the four slowest compatible with rod worth minimizer and CRD sequence requirements.

63 **Scram times of the four slowest CRD's will be determined at 25%, and 100% of rated power during planned reactor scrams.

***Establish initially that this check is normal operating procedure.

56 NOTE: Single CRD scrams should be performed with the charging valve closed (do not ride the charging pump head).

Criteria

Level 1

Each CRD must have a normal withdraw speed less than or equal to 3.6 inches per second (9.14 cm/sec), indicated by a full 12-foot stroke in greater than or equal to 40 seconds.

The mean scram time of all operable CRD's must not
54 exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

Percent Inserted	Scram Time (Seconds) Vessel Dome Pressure >950 psig (66.9 kg/cm²)	Scram Time (Seconds) Vessel Dome Pressure <950 psig (66.9 kg/sm²)
5	0.375	0.475
20	0.90	1.100
50	2.0	2.0
64 90	3.5	3.5

The mean scram time of the three fastest CRD's in a two-by-two array must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)
56

Percent Inserted	Scram Time (Seconds) Vessel Dome Pressure >950 psig (66.9 kg/cm²)	Scram Time (Seconds) Vessel Dome Pressure <950 psig (66.9 kg/sm²)
5	0.398	0.504
20	0.954	1.166
50	2.120	2.120
64 90	3.800	3.800

Level 2

Each CRD must have a normal insert or withdrawn speed of 3.0 ± 0.6 inches per second (7.62 sma,b 1.52 cm/sec), indicated by a full 12-foot stroke in 40 to 60 seconds.

With respect to the control rod drive friction tests, if
56 the differential pressure variation exceeds 15 psid (1 kg/cm²) for a continuous drive in, a settling test must be performed, in which case, the differential settling pressure should not be less than 30 psid (2.1 kg/cm²) nor should it vary by more than 10 psid (0.7 kg/cm²) over a full stroke.

Scram times with normal accumulator charge should fall within the time limits indicated on Figure 5.3-1 of the Startup Test Instructions.

TEST NUMBER 6 – SRM PERFORMANCE AND CONTROL ROD SEQUENCE

Purpose

The purpose of this test is to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

Description

The operational neutron sources will be installed and source range monitor count-rate data will be taken during rod withdrawals to critical and compared with stated criteria on signal and signal count-to-noise count ratio.

A withdrawal sequence has been calculated which completely specifies control rod withdrawals from the all-rods-in condition to the rated power configuration. Critical rod patterns will be recorded periodically as the reactor is heated to rated temperature.

Movement of rods in a prescribed sequence is monitored by the Rod Worth Minimizer and the Rod 56
Sequence Control System, which will prevent out-of-sequence withdrawal and insertions.

As the withdrawal of each rod group is completed during the power ascension, the electrical power, steam flow, control valve position, and APRM response will be recorded.

Data will be obtained to verify the relationship between core power and first stage turbine pressure to insure that the RSCS properly fulfills its intended function up to the required power level.

Criteria

Level 1

There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRM's or Fuel Loading Chambers.

There must be a minimum count rate of 3 counts/second on the required operable SRM's or Fuel Loading Chambers.

The IRM's must be on scale before the SRM's exceed the rod block set point.

The Rod Sequence Control System shall be operable as specified in the Technical Specifications.

Table 13.5-5

STARTUP TEST PROGRAM (UNIT 2)

		OPEN VESSEL OR COLD TEST	HEAT UP		50% Flow Control Line				75% Flow Control Line				100% Flow Control Line				
	Power, %[1]			15-35	30-50	40-60	~25	~53	37-54	50-70	65-85	37	55-75	75-90	95-100	~40	95-100
	Flow, %[2]			~47	~70	~104	NC	~116	~46	~70	~102	NC	~42	~70	~100	NC	~109
	T CONDITION (See Figure 2)			1	2B	2E	2A	2C	3C	3B	3E	3A	4C	4B	4E	4A	5
1	Chemical & Radiochemical	X	X	X		X					X				X		
2	Radiation Measurement	X	X	X		X					X				X		
3	Fuel Loading	X															
4	Full Core Shutdown Margin	X															
5	CRD	X	X	X		X					X				X		
6	SRM Perf. & Control Rod Seq.	X	X	X													
9	Water Level Measurement		X	X											X		
10	IRM Performance	X	X	X													
11	LPRM Calibration			X		X					X				X		
12	APRM Calibration		X	X		X					X				X		
13	Process Computer	X	X			X[6]									X		
14	RCIC		X	M													
15	HPCI		X			X	X										
16	Selected Process Temperatures		X				X					X				X	
17	System Expansion	X	X	X													
18	Core Power Distribution			X		X					X				X		
19	Core Performance			X	X	X	X	X	X	X	X	X	X	X	X	X	X
20	Steam Production														X		
21	Flux Response to Rods			M		M					M				M		
22	Press. Reg: Set Point Changes			M		M					M		M	M	M		
	Backup Regulator			M		M					M				M		
23	FW System: FW Pump Trip														M		
	Water Level Stpt Chg.			M,A		M,A					M,A		M,A	M,A	M,A		M
24	Bypass Valves			M,A		M,A	X				M,A		M,A	M,A	M,A	M	M
25	Main Steam Iso. Valves: Each Vlv		X[9]			M,SP								M,SP			
	Full Iso														M,SE		
26	Relief Valve: Capacity			M													
	: Actuation[3]		X[9]	M							M						
27	Turbine Trip and														M,SE[8]		
	Generator Load Rejection			M,SP											M,SP[8]		
28	Flow Control			A	A	A		A	A	A	A		A	M,A	M,A		
30	Recirc. Sys: Trip One Pump					M[4],M[6]					M[4],M[7]				M[4],M[7]		
	Trip Both Pumps					M					M[4]				M		
	Sys. Performance					X	X	X			X	X			X	X	X
	Non-Cavit. Verif.					X[12]											
31	Loss of T-G & Offsite Power			M,SE[7]													
32	Recirc MG Set Speed Control			L,M	M	M		M	M	M	M		M	M	M		
33	Turbine Stop Valve Surv. Test			M		M					M				M,SP[11]		
34	Vibration Measurements			X[13]	X	X	X	X	X	X	X	X	X	X	X	X	X
35	Recirc. System Flow Calibration					X					X				X		
70	Reactor Water Clean-up System		X														
71	Residual Heat Removal System			X													
72	Drywell Atmospheric Cooling Sys[10]		X												X		
73	Cooling Water Systems[13]		X												X		
74	Off Gas System*			X		X					X				X		

[1]Percent of rated power, 3293 MWt
[2]Percent of rated flow, 102.5 x 10^6 lb/hr
[3]Also obtain data with Tests 25, Full Iso & Test 27
[4]Obtain data with Test 30
[5]Perform Test 5, timing of 4 slowest control rods in conjunction with these scrams
[6]Perform the Dynamic System Test Case
[7]Included only to meet Test 34 Requirements
[8]Trip the Generator Field Breaker
[9]Heat up tests of MSIVs & Relief Valves are to check operation only
[10]Unit II only (cf. Page 36)
[11]Determine maximum power without scram
[12]From Test Condition 2E to 5
[13]Not required if 50% power testing will be done within about 2 months

L = Local Manual Flow Control Mode
M = Master Manual Flow Control Mode
A = Automatic Flow Control Mode
X = Test Independent of Flow Control Mode
SP = Scram Possibility
SE = Scram Expected
NC = Natural Circulation

*Applies to unit 3 and, subsequent to equipment installation, to unit 2.

TEST NUMBER 5 - CONTROL ROD DRIVE SYSTEM

1. Deviation from purpose, description, and criteria

   a. Purpose - The FSAR calls for demonstration of CRD system operation over the full range of primary coolant temperatures and pressures from ambient to operating. Determination of initial operating characteristics of the entire CRD system is also required. For the purposes of startup retesting it will be sufficient to determine initial operating characteristics by friction and scram testing at zero reactor pressure after fuel loading (the preop tests as listed will also be performed prior to fuel loading). Scram times will also be measured at rated temperature and pressure during heatup and/or low power retesting.

   b. Description - According to the FSAR the periodic CRD testing during heatup is done to assure that there is no significant binding caused by thermal expansion of the core components. Since the thermal expansion characteristics have already been proven, they will not require periodic testing during heatup for the retest program. The control rod drive tests which will be required during startup retesting are position, indication, insert/withdraw times, coupling friction, and scram testing at zero reactor pressure; and scram testing of all CRD's at rated temperature and pressure. Additional initial startup testing with various accumulator pressures and with repeated confirmatory tests for selected rods has demonstrated expected design response and expected repeatability; therefore an extended retesting is not needed. The testing program will adhere to all technical specification requirements.

   c. Criteria - No change.

TEST NUMBER 5 - CONTROL ROD DRIVE SYSTEM (Continued)

2. Deviation from table 13.5-5 frequency

STI 5 will only be performed during open vessel testing, heatup, and at 15-40% power as described in the purpose and description (see 1.a and 1.b above). The change of the upper limit at 15-35% to 15-40% is consistent with technical specification requirements and rod sequence control system limitations. Further testing at test conditions 2E, 3E, and 4E is not needed for reasons given in 1.b above.

Attachment 3

GENERAL ELECTRIC

POWER SYSTEMS

SALES OPERATION

GENERAL ELECTRIC COMPANY . . . . 832 GEORGIA AVENUE
CHATTANOOGA, TENNESSEE 37402, Phone (615) 894-2550

July 12, 1976

Mr. J. R. Calhoun
Tennessee Valley Authority
702 Edney Building
Chattanooga, TN 37401

Subject: Browns Ferry 1 and 2
Shutdown Margin
GE Letter No. CF-78

Dear Jack:

Attached you will find a copy of the shutdown margin curve for Browns Ferry 1 and 2. This curve assumes: strongest control rod withdrawn (Core Coordinates 26-07), cold (20°C) zenon-free condition.

Very truly yours,

W. E. Buist

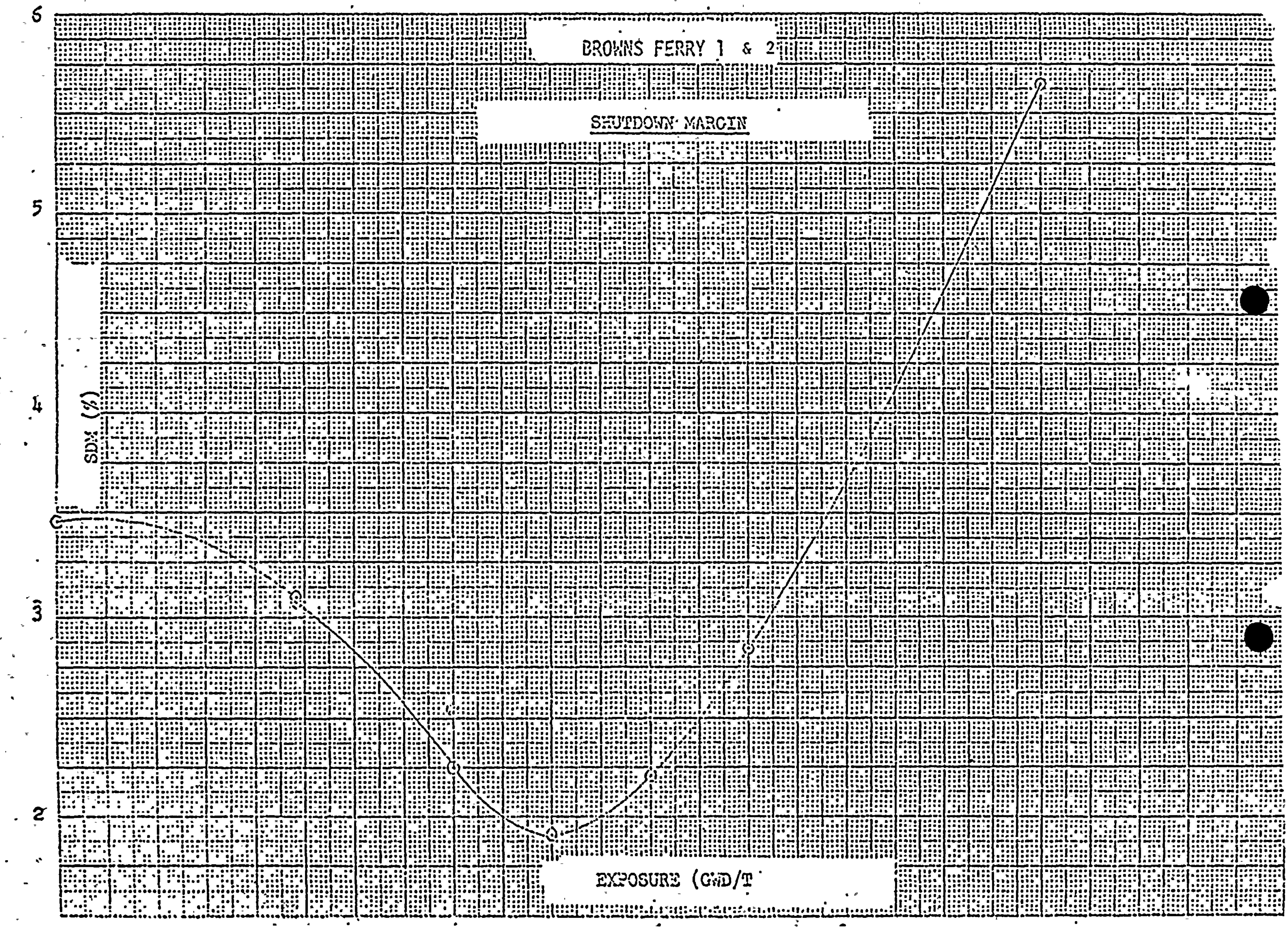
W. E. Buist
Generation Sales Engineer

WEB/lg

Attachment

RECEIVED
DPP - Nuclear
JUL 12 76

Referred To			
JRC			
JBS			


BROWNS FERRY 1 & 2
SHUTDOWN MARGIN
SDM (%)
EXPOSURE (GWD/T
6
5
4
3
2

GENERAL ELECTRIC

POWER SYSTEMS

SALES OPERATION

GENERAL ELECTRIC COMPANY . . . . 832 GEORGIA AVENUE
CHATTANOOGA, TENNESSEE 37402, Phone (615) 894-2550

July 19, 1976

Mr. J. R. Calhoun
Tennessee Valley Authority
702 Edney Building
Chattanooga, Tenn. 37401

Subject: Browns Ferry 1 and 2 Shutdown Margin
GE Letter No. CF-80

Reference: (1) GE Letter No. CF-78
(2) NEDE-20913-P, "Lattice Physics Methods"
(3) NEDO-20939, "Lattice Physics Methods Verification"
(4) NEDO-20953, "3 Dimensional BWR Core Simulator"
(5) NEDO-20946, "BWR Simulator Methods Verification"

Dear Jack:

The following should be added to our letter of July 12 (Reference No. 1):

> "Control rod 26-07 is the analytically strongest rod throughout Cycle 1. This curve was derived using methods documented in references 2-5."

Very truly yours,

W. E. Buist

W. E. Buist
Generation Sales Engineer

/mja

Attachment 4

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
TENNESSEE VALLEY AUTHORITY	)	Docket Nos. 50-259
	)	50-260
(Browns Ferry Nuclear Plant units 1 and 2)	)	

AFFIDAVIT OF ROBERT G. COCKRELL

Robert G. Cockrell, being duly sworn, deposes and says: My business address is Tennessee Valley Authority, Division of Power Production, 727 Edney Building, Chattanooga, assigned to Browns Ferry Nuclear Plant, Athens, Alabama. I am employed by the Tennessee Valley Authority as a nuclear engineer by the Division of Power Production. I have personal knowledge of the matters stated herein.

Qualifications

B.S. Engineering Science - Tennessee Technological University

M.S. Nuclear Science and Engineering - Virginia Polytechnic Institute and State University

I was hired by the Tennessee Valley Authority in the spring of 1975 and spent two months in the Plant Engineering Branch central office in Chattanooga before being sent to the Browns Ferry Nuclear Plant. I worked for approximately five months in the Browns Ferry Nuclear Plant Quality Assurance Staff as a Quality Assurance Engineer. I was then

transferred to the Power Plant Results Section to prepare for fuel loading and startup shift coverage. During this period, I was designated cognizant engineer on several projects including fuel assembly upper tie plate replacements, fuel assembly lower tie plate drilling for bypass flow holes, and operational startup source installation. For the past month, I have been standing coverage as a shift nuclear engineer for fuel loading on Browns Ferry units 1, 2, and 3.

Fuel Loading Verification

In order to ensure the proper loading of fuel assemblies at Browns Ferry, the fully loaded cores are inspected to verify correct placement and orientation of all fuel assemblies.

The core of each Browns Ferry reactor contains 764 fuel assembly bundles. Each of these bundles is permanently identified by a number etched into the assembly bail (handle), which is visible from above the fuel assembly.

Fuel bundles must also be checked for the proper orientation with respect to the control rod within each control cell (2 x 2 bundle array around a control rod). There are several ways to verify this orientation, but two methods are normally used. Each fuel bundle has a channel fastener in one corner which should point toward the center of the control rod when properly oriented. The operators use this method during fuel loading since it is easy to see this fastener from a distance. Secondly, when properly oriented, the fuel bundle identification number on a fuel assembly handle (bail) is always upright when viewed from the center of the control cell.

Upon completion of fuel loading, a video camera attached to the refueling boom is lowered to slightly above the fuel bundles. A monitor and videotape machine are also connected to the video camera. The fuel bundles are then scanned row by row in a prescribed manner producing a videotape that shows the identification number and orientation of every fuel bundle.

Browns Ferry units 1 and 2 were originally loaded to a preplanned array specified in the startup program. (Browns Ferry Nuclear Plant FSAR Section 13.5.) Verification that the cores were originally loaded correctly was made, as required by the Browns Ferry FSAR Section 13.5.1.2. This was achieved by making a videotape of the serial numbers of the loaded fuel. A core loading map of the serial numbers was made from this tape. This core loading map was compared to the preplanned map and verified identical by a Tennessee Valley Authority employee and a General Electric startup engineer. The maps and tapes are available for inspection and have been audited by Nuclear Regulatory Commission and Tennessee Valley Authority Quality Assurance personnel for both units.

After Browns Ferry unit 2 was reloaded, videotapes of the core were made by Tennessee Valley Authority employees on July 19, 1976. To ensure independent review of the tapes, I was not involved in the taping process itself.

Subsequently, I reviewed the tapes made on July 19, 1976, on a television monitor and observed every fuel bundle, noting the bundle location, identification number, and orientation. Since the camera was fixed, fuel bundle orientation was determined by observing the alignment of the identification numbers. As I observed each fuel bundle, I copied the identification number and orientation on a blank core map. After the tapes were viewed in entirety and the core map completed, I compared my map against the core loading map, which was made at the completion of the original fuel loading. I compared the two maps on a bundle-by-bundle basis and found them to be identical. On this basis, I conclude that Browns Ferry unit 2 is loaded to the exact configuration established prior to the March 22, 1975, shutdown.

At the completion of fuel loading on Browns Ferry unit 1, I will follow the same core verification procedure.

Robert G. Cockrell
Robert G. Cockrell

Sworn and subscribed before me

this 21st day of July, 1976.

Carolyn B. Roberts
Notary Public

My Commission Expires 7-2-78