VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

July 15, 1991

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. 91-389
NO/RPC:vlh
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
MONTHLY OPERATING REPORT

Enclosed is the Monthly Operating Report for Surry Power Station Units 1 and 2 for the month of June 1991.

Very truly yours,

W. L. Stewart
Senior Vice President - Nuclear

Enclosure

cc: U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

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VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION
MONTHLY OPERATING REPORT
REPORT NO. 91-06

Approved:

Station Manager

7/9/91
Date

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OPERATING DATA REPORT

Docket No.: 50-280
Date: 07-05-91
Completed By: M. A. Negron
Telephone: 804-365-2795

1. Unit Name: ..... Surry Unit 1
2. Reporting Period: ..... June 1991
3. Licensed Thermal Power (MWt): 2441
4. Nameplate Rating (Gross MWe): 847.5
5. Design Electrical Rating (Net MWe): 788
6. Maximum Dependable Capacity (Gross MWe): ... 820
7. Maximum Dependable Capacity (Net MWe): 781

8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

9. Power Level To Which Restricted, If Any (Net MWe): _____

10. Reasons For Restrictions, If Any: _____

		This Month	YTD	Cumulative
11.	Hours In Reporting Period	720.0	4343.0	162359.0
12.	Number of Hours Reactor Was Critical	720.0	4343.0	103817.2
13.	Reactor Reserve Shutdown Hours	0.0	0.0	3774.5
14.	Hours Generator On-Line	720.0	4343.0	101823.2
15.	Unit Reserve Shutdown Hours	0.0	0.0	3736.2
16.	Gross Thermal Energy Generated (MWH)	1731622.0	10524266.4	236744176.7
17.	Gross Electrical Energy Generated (MWH)	573005.0	3532295.0	77109118.0
18.	Net Electrical Energy Generated (MWH)	542758.0	3357800.0	73140929.0
19.	Unit Service Factor	100.0%	100.0%	62.7%
20.	Unit Availability Factor	100.0%	100.0%	65.0%
21.	Unit Capacity Factor (Using MDC Net)	96.5%	99.0%	58.1%
22.	Unit Capacity Factor (Using DER Net)	95.7%	98.1%	57.2%
23.	Unit Forced Outage Rate	0.0%	0.0%	19.9%

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):

25. If Shut Down at End of Report Period Estimated Date of Start-up: _____

26. Unit In Test Status (Prior to Commercial Operation):

	FORECAST	ACHIEVED
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

OPERATING DATA REPORT

Docket No.: 50-281
Date: 07-05-91
Completed By: M. A. Negron
Telephone: 804-365-2795

1. Unit Name: ..... Surry Unit 2
2. Reporting Period: ..... June 1991
3. Licensed Thermal Power (MWt): 2441
4. Nameplate Rating (Gross MWe): 847.5
5. Design Electrical Rating (Net MWe): 788
6. Maximum Dependable Capacity (Gross MWe): ... 820
7. Maximum Dependable Capacity (Net MWe): 781

8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

9. Power Level To Which Restricted, If Any (Net MWe):

10. Reasons For Restrictions, If Any:

		This Month	YTD	Cumulative
11.	Hours In Reporting Period	720.0	4343.0	159239.0
12.	Number of Hours Reactor Was Critical	127.7	2263.7	101436.0
13.	Reactor Reserve Shutdown Hours	0.0	0.0	328.1
14.	Hours Generator On-Line.....	70.5	2185.3	99755.9
15.	Unit Reserve Shutdown Hours.....	0.0	0.0	0.0
16.	Gross Thermal Energy Generated (MWH).....	44230.9	4538174.1	232673642.4
17.	Gross Electrical Energy Generated (MWH)....	14075.0	1524390.0	75754504.0
18.	Net Electrical Energy Generated (MWH).....	11960.0	1444820.0	71823545.0
19.	Unit Service Factor.....	9.8%	50.3%	62.6%
20.	Unit Availability Factor.....	9.8%	50.3%	62.6%
21.	Unit Capacity Factor (Using MDC Net).....	2.1%	42.6%	57.9%
22.	Unit Capacity Factor (Using DER Net).....	2.1%	42.2%	57.2%
23.	Unit Forced Outage Rate.....	87.2%	18.0%	15.3%

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
Forced outage expected to end 7-4-91.

25. If Shut Down at End of Report Period Estimated Date of Start-up: 7-4-91

26. Unit In Test Status (Prior to Commercial Operation):

	FORECAST	ACHIEVED
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWN AND POWER REDUCTION
(EQUAL TO OR GREATER THAN 20%)

REPORT MONTH: June 1991

Docket No.: 50-280
Unit Name: Surry Unit 1
Date: 07-05-91
Completed by: M. A. Negron
Telephone: 804-365-2795

Date	(1) Type	Duration Hours	(2) Reason	(3) Method of Shutting Down Rx	LER No.	(4) System Code	(5) Component Code	Cause & Corrective Action to Prevent Recurrence
6-4-91	S	0	B	4	N/A	TA	V	Unit was ramped from 100% to 78.5% for 1-PT-29.1 (Turbine Valve Freedom Test)
6-18-91	S	0	B	4	N/A	SH	HX	Unit load was reduced to maintain condenser vacuum while cleaning waterboxes.
6-28-91	S	0	B	4	N/A	SH	HX	Unit load was reduced to maintain condenser vacuum while cleaning waterboxes.

(1)
F: Forced
S: Scheduled

(2)
REASON:
A - Equipment Failure (Explain)
B - Maintenance or Test
C - Refueling
D - Regulatory Restriction
E - Operator Training & Licensing Examination
F - Administrative
G - Operational Error (Explain)

(3)
METHOD:
1 - Manual
2 - Manual Scram.
3 - Automatic Scram.
4 - Other (Explain)

(4)
Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG 0161)

(5)
Exhibit 1 - Same Source.

UNIT SHUTDOWN AND POWER REDUCTION
(EQUAL TO OR GREATER THAN 20%)

REPORT MONTH: June 1991

Docket No.: 50-281
Unit Name: Surry Unit 2
Date: 07-05-91
Completed by: M. A. Negron
Telephone: 804-365-2795

Date	(1) Type	Duration Hours	(2) Reason	(3) Method of Shutting Down Rx	LER No.	(4) System Code	(5) Component Code	Cause & Corrective Action to Prevent Recurrence
6-7-91	F	16.3	A	4	N/A	FK	BKR	Unit lost load (~80 MWe) when disconnect G-202 was opened after arcing occurred on a breaker in the switchyard.
6-11-91	F	479.5	A	4	N/A	AA	DRIV	Unit was shutdown, in accordance with applicable procedures, to the refueling shutdown mode (RSD) after it was determined that the control rod at core position F-6 was not latched.

(1)
F: Forced
S: Scheduled

(2)
REASON:
A - Equipment Failure (Explain)
B - Maintenance or Test
C - Refueling
D - Regulatory Restriction
E - Operator Training & Licensing Examination
F - Administrative
G - Operational Error (Explain)

(3)
METHOD:
1 - Manual
2 - Manual Scram.
3 - Automatic Scram.
4 - Other (Explain)

(4)
Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG 0161)

(5)
Exhibit 1 - Same Source.

AVERAGE DAILY UNIT POWER LEVEL

Docket No.: 50-280
Unit Name: Surry Unit 1
Date: 07-05-91
Completed by: M. A. Negron
Telephone: 804-365-2795

Month: June 1991

Day	Average Daily Power Level (MWe - Net)	Day	Average Daily Power Level (MWe - Net)
1	751.3	17	765.5
2	742.2	18	742.5
3	737.9	19	755.6
4	732.6	20	769.3
5	764.3	21	764.5
6	742.4	22	763.4
7	762.1	23	765.4
8	771.3	24	766.4
9	773.0	25	761.8
10	772.1	26	759.1
11	774.0	27	758.2
12	773.2	28	743.2
13	771.3	29	574.6
14	771.2	30	751.1
15	769.7	31	
16	765.8		

INSTRUCTIONS

On this format, list the average daily unit power level in MWe - Net for each day in the reporting month. Compute to the nearest whole megawatt.

AVERAGE DAILY UNIT POWER LEVEL

Docket No.: 50-281
Unit Name: Surry Unit 2
Date: 07-05-91
Completed by: M. A. Negron
Telephone: 804-365-2795

Month: June 1991

Day	Average Daily Power Level (MWe - Net)	Day	Average Daily Power Level (MWe - Net)
1	0.0	17	0.0
2	0.0	18	0.0
3	0.0	19	0.0
4	0.0	20	0.0
5	0.0	21	0.0
6	0.0	22	0.0
7	0.0	23	0.0
8	157.2	24	0.0
9	175.4	25	0.0
10	165.8	26	0.0
11	0.0	27	0.0
12	0.0	28	0.0
13	0.0	29	0.0
14	0.0	30	0.0
15	0.0	31	
16	0.0		

INSTRUCTIONS

On this format, list the average daily unit power level in MWe - Net for each day in the reporting month. Compute to the nearest whole megawatt.

SUMMARY OF OPERATING EXPERIENCE

MONTH/YEAR: June 1991

Listed below in chronological sequence by unit is a summary of operating experiences for this month which required load reductions or resulted in significant non-load related incidents.

UNIT ONE

06-01-91	0000	This report period started with the Unit operating at 100% power and 805 MWe.
06-02-91	2006	Started load reduction to maintain condenser vacuum while cleaning waterboxes; 100% power, 805 MWe.
06-03-91	0234	Stopped load reduction; 86.6% power, 675 MWe.
	0456	Started ramp up after waterboxes returned to service; 86.6% power, 675 MWe.
	0630	Stopped ramp; 100% power, 800 MWe.
06-04-91	0953	Started ramp down to perform 1-PT-29.1; 100% power, 810 MWe.
	1215	Stopped ramp; 78.5% power, 630 MWe.
	1354	Started ramp up after completion of test; 77.7% power, 630 MWe.
	1711	Stopped ramp; 100% power, 800 MWe.
06-06-91	0950	Unit power was reduced from 100% power, 810 MWe to 93% power, 770 MWe after the four Turbine Governor Valves received a full open signal.
	2050	Unit was returned to 100% power with the turbine in manual control.
06-18-91	2035	Started load reduction to maintain condenser vacuum while cleaning waterboxes; 100% power, 805 MWe.
	2137	Stopped load reduction; 80% power, 620 MWe.
06-19-91	0103	Started ramp up; 80% power, 620 MWe.
	0215	Stopped ramp; 100% power, 805 MWe.
06-28-91	2107	Started load reduction to maintain condenser vacuum while cleaning waterboxes; 100% power, 780 MWe.
06-29-91	0505	Stopped load reduction; 73.7% power, 580 MWe.
	0550	Started increasing load as condenser vacuum would allow; 74% power, 545 MWe.
	2250	Stopped load increase; 100% power, 800 MWe.
06-30-91	2345	Unit power was reduced from 100% power, 785 MWe to 90% power, 720 MWe to stabilize the unit following a transient induced by the "B" Main Feed Regulating Valve.
	2400	This report period ended with the Unit operating at 89% power and 710 MWe.

SUMMARY OF OPERATING EXPERIENCE

MONTH/YEAR: June 1991

[continued]

UNIT TWO

06-01-91	0000	This report period started with the Unit in a scheduled Refueling Outage at Intermediate Shutdown (ISD).
06-05-91	1730	Reactor was critical.
06-07-91	0920	Unit on line.
	0945	Unit off line; the G-202 disconnect was opened due to arcing on a breaker in the switchyard.
	1130	Reactor at 2% power.
06-08-91	0159	Unit on line and ramp up started.
	0316	Stopped ramp at 30% power, 220 MWe.
06-10-91	2305	Started ramp down to commence Unit shutdown to Refueling Shutdown (RSD) due to indications that Control Rod F-6 was not latched; 31% power, 195 MWe.
06-11-91	0031	Unit offline.
	0110	Reactor was manually tripped in accordance with 2-OP-3.1; 1.3E-7 amps on the Intermediate Range.
	2047	Unit at Cold Shutdown (CSD).
06-30-91	2400	This report period ended with the Unit at CSD.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

TM S2-91-19 **Temporary Modification** 05-25-91
(Safety Evaluation No. 91-136)

This Temporary Modification (TM) installs electrical jumpers from the 2S12 lighting panel, a non-emergency power source, to the 2J1-1 emergency bus 4160 volt to 480 volt transformer fans to provide an alternate power supply for the fans. The TM was necessary due to incompatibility between the fans' power supply configuration and a new transformer being installed following failure of the transformer.

An engineering evaluation determined that transformer operability is not affected by the availability of fan cooling. Thus, an emergency power supply is not required for the fans and the lighting panel is an acceptable power supply. This TM will be isolated from the emergency bus and will have no effect on the operability of the transformer. In addition, operation of the fans will be verified and the lighting panel breakers will be checked each shift. Therefore, an unreviewed safety question does not exist.

TM S2-91-18 **Temporary Modification** 05-26-91
(Safety Evaluation No. 91-137)

This Temporary Modification (TM) installs mechanical jumpers (instrument air lines) around the outside blowdown trip valve solenoid valves, 2-BD-SOV-200B, 2-BD-SOV-200D, 2-BD-SOV-200F, to the valve actuators. This will allow testing of the ATWS Mitigating System Actuation Circuitry (AMSAC), Train "B", relative to Unit 2 modifications of the Anticipated Transient Without a SCRAM (ATWS) Mitigation system while maintaining the the outside blowdown trip valves open. This configuration will permit concurrent prestart-up operations evolutions associated with returning the steam generators to service.

The unit will be at Cold Shutdown with the steam generators depressurized and the Residual Heat Removal system operating while this TM is in place. The TM does not affect the pressure boundary and will be removed and the system restored to its previous configuration prior to Unit start-up. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

[continued]

SE 91-146 **Safety Evaluation** 06-03-91

This Safety Evaluation was performed to evaluate the use of freeze seals on the Reactor Coolant System (RCS) "C" loop Resistance Temperature Detector (RTD) manifold to isolate the return piping and allow the return isolation valve, 2-RC-95, to be repaired.

The Unit was at Cold Shutdown during the installation and special RCS pressure and temperature requirements were imposed. Limiting conditions were established to limit potential RCS leakage, as a result of freeze seal failure, to ensure the leakage would remain within the make-up capacity of the Safety Injection system. Contingency actions were also in place to minimize the consequences of a freeze seal failure. In addition, Nondestructive Examination of the affected piping was performed prior to and following the seal installation. Therefore, an unreviewed safety question does not exist.

EWR 91-076 **Engineering Work Request** 06-03-91
(Safety Evaluation No. 91-147)

This Engineering Work Request (EWR) removes the internals and bonnet from Reactor Coolant System (RCS) Resistance Temperature Device (RTD) manifold outlet isolation valve, 2-RC-95, and installs a blind flange.

The subject valve was installed to provide isolation for maintenance purposes, provides no nuclear safety function and remains open during normal plant operation. This modification does not adversely affect flow through the RTD manifold, RCS operating characteristics, or system pressure boundary integrity. Therefore, an unreviewed safety question does not exist.

SE 91-120 Revision 1 **Safety Evaluation** 06-04-91

This Safety Evaluation was performed to reevaluate the probability of occurrence of a broken fuel assembly spring clamp since this event was again observed during the Unit 2 Cycle 10 refueling outage. The broken spring clamps have only been associated with a particular spring clamp design configuration which has been modified. However, certain assemblies with this design are utilized in the Cycle 11 core designs for both Unit 1 and Unit 2.

The Safety Evaluation concludes that the probability of an accident is not increased since the spring clamp failure mechanism does not inhibit the ability of the control rod to perform its safety function (i.e., initial insertion into the reactor core during a reactor trip or for reactor shutdown). The evaluation reviews the increased probability of spring clamp failure and discusses the fact that such a failure, even if undetected, only renders the rod inoperable on withdrawal following initial insertion of the control rod into its host assembly during a trip or shutdown. In addition, current safety analyses already account for a single stuck rod, and are based on the assumption that the single most reactive rod is affected. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

[continued]

TM S2-91-22 **Temporary Modification** 06-04-91
(Safety Evaluation No. 91-148)

This Temporary Modification (TM) installs an electrical jumper to maintain power to certain Reactor Protection (RP) system circuit relays during the replacement of a failed relay, 2-RP-REL-SV3XA.

The Unit will be at Cold Shutdown during the installation of this TM and only one train of the RP system is affected. In addition, the "daisy chain" connection on either side of the subject relay will remain intact via the jumper. An evaluation, to determine the impact of disconnecting the output contacts on the failed relay, was performed and concluded that the action would be acceptable provided the reactor trip breakers are initially open and the turbine is not engaged. The electrical jumper will be replaced following relay replacement. Therefore, an unreviewed safety question does not exist.

TM S2-91-23 **Temporary Modification** 06-10-91
(Safety Evaluation No. 91-150)

This Temporary Modification (TM) installs an electrical jumper to bypass the rod bottom bistable for Unit 2 control rod, F-6 (control bank "D"). The Rod Cluster Control Assembly (RCCA) located in control rod location F-6 has been determined to be uncoupled from its driveshaft. This change will permit the control rod driveshaft to be moved closer to the detached control rod without generating a turbine runback signal or a rod urgent failure alarm from the Individual Rod Position Indication (IRPI). This TM will also permit the driveshaft of the control rod to be placed down on top of the core during unit shutdown to minimize any potential movement of the drive shaft during operations to shutdown the unit to investigate and repair the unlatched rod.

This TM does not constitute a reactivity control or nuclear safety concern for bypassing the rod bottom bistable since the control rod is already fully inserted. Therefore, an unreviewed safety question does not exist.

TSR 91-065 **Temporary Shielding Requests** 06-11-91
(Safety Evaluation No. 91-153)

This Temporary Shielding Request (TSR) installs temporary lead shielding on Charging System (CH) and Safety Injection (SI) System piping in the Unit 1 "C" Charging Pump cubicle to reduce the radiation dose received by personnel while painting the Charging Pump cubicle.

The subject CH and SI lines will be considered operable while the shielding is installed. A seismic analysis was performed and the installation was determined to be acceptable. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

[continued]

TM S2-91-24 **Temporary Modification** 06-14-91
(Safety Evaluation No. 91-156)

This Temporary Modification (TM) installs a temporary spoolpiece in place of the suction strainer for the "B" High Level Intake Traveling Screen Screenwash System until a replacement strainer can be procured.

This TM will not affect the operability of the intake canal level instrumentation or the flowpaths to the Service Water system. In addition, the spoolpiece is fully capable of withstanding the anticipated stresses and is resistant to general corrosion in a saltwater environment. Therefore, an unreviewed safety question does not exist.

TM S2-91-25 **Temporary Modification** 06-19-91
(Safety Evaluation No. 91-160)

This Temporary Modification (TM) installs electrical jumpers to provide power to certain Reactor Protection (RP) system relays during the replacement of failed relays 2-RP-REL-P7XA and 2-RP-REL-P7YA.

The Unit will be at Cold Shutdown during the installation of this TM and only one train of the RP system is affected. In addition, the positive and negative "daisy chain" connection on either side of the subject relays will remain intact via the jumper. An evaluation, to determine the impact of disconnecting the output contacts on the failed relays, was performed and concluded that the action would be acceptable provided the reactor trip breakers are open. The electrical jumper will be removed following replacement of the relays. Therefore, an unreviewed safety question does not exist.

TM S2-91-26 **Temporary Modification** 06-21-91
JCO C-91-005 **Justification for Continued Operation**
(Safety Evaluation No. 91-161)

This Temporary Modification (TM) de-energizes the inlet damper actuator hydraulic motor for the Auxiliary building ventilation system fan, 1-VS-F-58B, by removing the fuses in the motor power leads and lifting the leads that provide the position signal to the actuator. This modification will be implemented to allow local manual control of the damper to be taken until the hydraulic motor can be repaired.

The flow rate will be set by manually adjusting the damper based on the flow gauge indications in the Control Room. Periodic Test PT-32.1, Auxiliary Ventilation Filter Train Test, will be performed following the manual damper setting and weekly thereafter until remote control is reestablished. This TM enables the affected train of the Auxiliary building ventilation system to remain fully operable in a condition which was anticipated in the original design by providing the manual actuator for the damper. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

[continued]

TSR 91-070 — **Temporary Shielding Requests** — 06-21-91
(Safety Evaluation No. 91-162)

This Temporary Shielding Request (TSR) installs temporary lead shielding on Reactor Coolant System (RCS) and Safety Injection (SI) System piping in the Unit 2 Containment building to reduce the radiation dose received by personnel during the replacement of SI valve 2-SI-79.

The subject RCS and SI lines will be considered operable (but will not be pressurized) while the shielding is installed. Seismic and deadweight analyses were performed on the affected piping and the installation was determined to be acceptable. The shielding will be removed prior to pressurizing the systems. Therefore, an unreviewed safety question does not exist.

SE 91-088
Revision 1 — **Safety Evaluation** — 06-27-91

This Safety Evaluation was performed to evaluate the storage of a NAC-I28 spent fuel cask at the Independent Spent Fuel Storage Installation without an upper impact limiter.

The evaluation concluded that storage of the cask without an impact limiter is safe based on 1) analyses which demonstrate that a cask will not tipover during a Surry ISFSI Design Basis Earthquake (DBE), 2) a site specific tipover analysis performed by NAC demonstrating that the cask seal design configuration and fuel integrity are not compromised in the event that the NAC-I28 cask were to tipover without an upper impact limiter on the cask. Therefore, an unreviewed safety question does not exist.

TM S2-91-27 — **Temporary Modification** — 06-27-91
(Safety Evaluation No. 91-168)

This Temporary Modification (TM) mechanically fixes in the open position the fan outlet damper on the "A" train of the Control Rod Drive Mechanism (CRDM) shroud cooling ventilation fan, 2-VS-F-60A, until the damper can be repaired.

This TM will ensure that the "A" train will operate to provide maximum cooling of the CRDM area. The modification will not affect electrical circuity or increase the probability of a loss of power. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

[continued]

TM S2-91-28 **Temporary Modification** 06-28-91
(Safety Evaluation No. 91-169)

This Temporary Modification (TM) installs electrical jumpers to provide power to certain Reactor Protection (RP) system relays during the replacement of failed relay 2-RP-REL-P10YB.

This TM affects only one train of the RP system. The Unit will be at Cold Shutdown during the installation and both reactor trip breakers will be open and the jumper will be removed following replacement. Therefore, an unreviewed safety question does not exist.

PROCEDURES OR METHOD OF OPERATION CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

2-PT-6.0 **Periodic Test Procedure** 06-09-91
(Safety Evaluation No. 91-149)

A "one-time" change to Periodic Test Procedure 2-PT-6.0, Control Rod Assembly Partial Movement, was implemented to permit the driveshaft for Rod Cluster Control Assembly F-6 ("D" bank) to be inserted while the remaining "D" bank RCCAs remain partially withdrawn.

This change is being made to verify movement of the subject RCCA. It will be performed with the Unit 2 power level less than 50% to eliminate the potential for fuel damage resulting from a fully misaligned RCCA. In addition, a shutdown margin calculation was performed, assuming full insertion of the RCCA. Therefore, an unreviewed safety question does not exist.

2-PT-6.0 **Periodic Test Procedure** 06-10-91
(Safety Evaluation No. 91-152)

A change to Periodic Test Procedure 2-PT-6.0, Control Rod Assembly Partial Movement, was implemented to permit the driveshaft for inoperable control rod F-6 ("D" bank) to be fully inserted prior to shutting the Unit down. The purpose of this activity is to place the driveshaft of the control rod down on top of the core so the Unit may be shutdown to repair the inoperable control rod.

This TM does not affect reactivity control since the respective control rod is fully inserted. Lowering the driveshaft will not affect the ability of the other control rods to perform their safety function and will reduce the potential for equipment or fuel damage due to a free fall of the driveshaft. Therefore, an unreviewed safety question does not exist.

2-OP-4.1 **Periodic Test Procedure** 06-14-91
(Safety Evaluation No. 91-158)

A "one-time" change to Operations Procedure 2-OP-4.1, Refueling Operations, was implemented to provide direction for the trouble shooting and repair efforts needed to address failed Rod Cluster Control Assembly R172 (control rod location F-6).

The change allows flexibility in the refueling sequences and will be conducted under the direction of the Refueling Coordinator and the Refueling Senior Reactor Operator (licensed SRO). The Technical Specification requirements for refueling conditions will be adhered to during the performance of this procedure and abnormal procedures are in place to address the malfunction of fuel handling equipment or other fuel handling events. Therefore, an unreviewed safety question does not exist.

PROCEDURES OR METHOD OF OPERATION CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

[continued]

MMP-C-RC-038 **Mechanical Maintenance Procedure** 06-16-91
(Safety Evaluation No. 91-159)

Mechanical Maintenance Procedure MMP-C-RC-038, Removal and Reinstallation of Reactor Vessel Studs, was changed to reduce (from 3 to 2) the required number of guide studs to lift the reactor head. This change was made to prevent delaying a reactor head lift due to a bent guide stud.

The use of two guide studs for head lifts was reviewed and recommended by Westinghouse, the reactor designer. The process will not produce a condition that is not addressed by the UFSAR fuel handling accident analyses. Therefore, an unreviewed safety question does not exist.

0-TOP-4006 **Temporary Operating Procedure** 06-27-91
(Safety Evaluation No. 91-167)

This Temporary Operating Procedure was developed to provide instructions for removing a damaged part-length control rod from a fuel assembly to permit the assembly to be placed in a dry storage cask. This procedure is required since initial attempts to remove the part-length control rod using the Rod Cluster Control Assembly change tool have failed because of a damaged hub.

This activity has no effect on spent fuel pool reactivity or the probability of occurrence or consequences of accidents considered in the UFSAR. Therefore, an unreviewed safety question does not exist.

TESTS AND EXPERIMENTS THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: June 1991

None During This Reporting Period.

CHEMISTRY REPORT

MONTH/YEAR: June 1991

Primary Coolant Analysis	Unit No. 1			Unit No. 2		
	Max.	Min.	Avg.	Max.	Min.	Avg.
Gross Radioactivity., µCi/ml	7.78E-1	3.09E-1	5.13E-1	7.30E-2	4.72E-4	5.09E-3
Suspended Solids, ppm	0.0	0.0	0.0	0.0	0.0	0.0
Gross Tritium, µCi/ml	3.34E-1	3.18E-1	3.26E-1	-------	-------	-------
I^{131}, µCi/ml	1.00E-2	1.56E-3	3.25E-3	-------	-------	-------
I^{131}/I^{133}	0.14	0.08	0.10	-------	-------	-------
Hydrogen, cc/kg	35.2	17.9	28.7	29.9	2.3	16.6
Lithium, ppm	2.34	2.03	2.19	3.38	0.03	1.90
Boron - 10, ppm*	115	102	109	500	86.2	364
Oxygen, (DO), ppm	≤0.005	≤0.005	≤0.005	6.0	≤0.005	2.18
Chloride, ppm	0.005	≤0.001	0.003	0.006	≤0.001	0.002
pH at 25 degree Celsius	6.98	6.82	6.89	6.20	4.60	5.10

* Boron - 10 = Total Boron x 0.196

Comments:

Unit 1: Dissolved H_2 concentration (conc.) of 20.7cc/kg on 6-3 at 2030 was below the limit of 25cc/kg. Back in spec on 6-4 at 0640 with a reading of 27.8cc/kg. Total out of specification (spec) time was 10 hours 10 minutes.

Lithium (Li) conc. of 2.04 ppm at 0020 on 6-22 was below the limit of 2.05 ppm. This occurred while letdown was out of service. Li was back in spec on 6-22 at 0830 with a concentration of 2.10 ppm. Total out of spec time was 8 hours 10 minutes.

Unit 2: Dissolved H_2 conc. of 17.0 cc/kg at 1800 on 6-5 was below the limit of 25 cc/kg. H_2 was back in spec with a conc. of 25.2 cc/kg on 6-6 at 0210. Total out of spec time was 8 hours 10 minutes.

On 6-7, dissolved H_2 conc. was determined to be 23.9 cc/kg at 1805 and was back in spec on 6-8 at 0020 with a reading of 25.1 cc/kg. Total out of spec time was 6 hours 15 minutes.

FUEL HANDLING
UNITS 1 & 2

MONTH/YEAR: June 1991

Units One and Two

Cask	Stored	Number for Assemblies per Shipment	Assembly Number	ANSI Number	Nominal Initial Enrichment	New or Spent Fuel Shipping Cask Activity

None during this reporting period.

DESCRIPTION OF PERIODIC TEST(S) WHICH WERE NOT COMPLETED WITHIN THE TIME LIMITS SPECIFIED IN TECHNICAL SPECIFICATIONS

MONTH/YEAR: June 1991

None During This Reporting Period.