VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 15, 1997

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	97-521
NL/JDK/MAE	R0
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

The Monthly Operating Report for Surry Power Station Units 1 and 2 for the month of August 1997 is provided in the attachment.

If you have any questions or require additional information, please contact us.

Very truly yours,

J. H. McCarthy, Manager
Nuclear Licensing and Operations Support

Attachment

Commitments made by this letter: None

IE24 /,

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, S. W.
Suite 23T85
Atlanta, Georgia 30303

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION
MONTHLY OPERATING REPORT
REPORT NO. 97-08

Approved: 9-9-97

Station Manager | Date

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OPERATING DATA REPORT

Docket No.: 50-280
Date: 09/1/97
Completed By: D. K. Mason
Telephone: (757) 365-2459

1. Unit Name:.. Surry Unit 1
2. Reporting Period:... August, 1997
3. Licensed Thermal Power (MWt): 2546
4. Nameplate Rating (Gross MWe):......................... 847.5
5. Design Electrical Rating (Net MWe): 788
6. Maximum Dependable Capacity (Gross MWe):.... 840
7. Maximum Dependable Capacity (Net MWe):........ 801

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	Year-To-Date	Cumulative
11.	Hours in Reporting Period	744.0	5831.0	216455.0
12.	Number of Hours Reactor Was Critical	744.0	4262.1	151096.8
13.	Reactor Reserve Shutdown Hours	0.0	0.0	3774.5
14.	Hours Generator On-Line	744.0	4139.5	148670.5
15.	Unit Reserve Shutdown Hours	0.0	0.0	3736.2
16.	Gross Thermal Energy Generated (MWH)	1877848.2	10164497.5	348799941.3
17.	Gross Electrical Energy Generated (MWH)	617542.0	3364745.0	114337563.0
18.	Net Electrical Energy Generated (MWH)	595807.0	3245381.0	108839130.0
19.	Unit Service Factor	100.0%	71.0%	68.7%
20.	Unit Availability Factor	100.0%	71.0%	70.4%
21.	Unit Capacity Factor (Using MDC Net)	100.0%	69.5%	62.8%
22.	Unit Capacity Factor (Using DER Net)	101.6%	70.6%	63.8%
23.	Unit Forced Outage Rate	0.0%	6.5%	15.0%

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: 09/01/97
Completed By: D. K. Mason
Telephone: (757) 365-2459

1. Unit Name:.. Surry Unit 2
2. Reporting Period:... August, 1997
3. Licensed Thermal Power (MWt): 2546
4. Nameplate Rating (Gross MWe):.......................... 847.5
5. Design Electrical Rating (Net MWe): 788
6. Maximum Dependable Capacity (Gross MWe):.... 840
7. Maximum Dependable Capacity (Net MWe):........ 801

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	Year-To-Date	Cumulative
11.	Hours in Reporting Period	744.0	5831.0	213335.0
12.	Number of Hours Reactor Was Critical	744.0	5767.0	148842.6
13.	Reactor Reserve Shutdown Hours	0.0	0.0	328.1
14.	Hours Generator On-Line	744.0	5760.1	146857.9
15.	Unit Reserve Shutdown Hours	0.0	0.0	0.0
16.	Gross Thermal Energy Generated (MWH)	1890618.9	14496295.2	345970552.0
17.	Gross Electrical Energy Generated (MWH)	621329.0	4818019.0	113268818.0
18.	Net Electrical Energy Generated (MWH)	599580.0	4656416.0	107848295.0
19.	Unit Service Factor	100.0%	98.8%	68.8%
20.	Unit Availability Factor	100.0%	98.8%	68.8%
21.	Unit Capacity Factor (Using MDC Net)	100.6%	99.7%	63.1%
22.	Unit Capacity Factor (Using DER Net)	102.3%	101.3%	64.2%
23.	Unit Forced Outage Rate	0.0%	1.2%	12.1%

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
Refueling, October 6, 1997, 30 Days

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		

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

REPORT MONTH: August, 1997

Docket No.: 50-280
Unit Name: Surry Unit 1
Date: 09-03-97
Completed by: M. J. Fanguy
Telephone: (757) 365-2155

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
8/21/97	S	NA	B	NA	NA	TA	V	Power reduction for turbine valve freedom test

(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: August, 1997

Docket No.: 50-281
Unit Name: Surry Unit 2
Date: 09-03-97
Completed by: M. J. Fanguy
Telephone: (757) 365-2155

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

None During the Reporting Period

(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: 09-04-97
Completed by: J. D. Kilmer
Telephone: (757) 365-2792

MONTH: August, 1997

Day	Average Daily Power Level (MWe - Net)	Day	Average Daily Power Level (MWe - Net)
1	811	17	804
2	802	18	804
3	805	19	805
4	812	20	810
5	813	21	572
6	813	22	805
7	813	23	809
8	812	24	808
9	812	25	806
10	813	26	808
11	811	27	809
12	811	28	807
13	810	29	805
14	809	30	805
15	808	31	806
16	806		

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: 09-04-97
Completed by: John D. Kilmer
Telephone: (757) 365-2792

MONTH: August, 1997

Day	Average Daily Power Level (MWe - Net)	Day	Average Daily Power Level (MWe - Net)
1	812	17	796
2	815	18	798
3	812	19	799
4	809	20	785
5	810	21	805
6	813	22	806
7	813	23	785
8	812	24	762
9	812	25	810
10	813	26	816
11	810	27	814
12	808	28	814
13	806	29	813
14	805	30	814
15	801	31	815
16	799		

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: August, 1997

The following 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:

8/1/97	0000	Unit 1 begins the month at 100% / 840 MWe.
8/21/97	0655	Start power decrease from 100% / 840 MWe to perform turbine valve freedom test
	1105	Stop power decrease at 73%.
	1122	Start power decrease to 50% to repair "C" intercept valve due to it failing to reopen following testing.
	1401	Stopped power decrease at 51%.
	2121	Start power increase after repair of "C" intercept valve
8/22/97	0142	Stop power increase at 100% / 837 MWe.
8/31/97	2400	Unit 1 finishes the month at 100% /835 MWe.

UNIT TWO:

8/1/97	0000	Unit 2 starts the month at 100% / 845 MWe.
8/23/97	0711	Adjusting turbine load to maintain condenser vacuum with "C" waterbox being removed from service for cleaning.
	1845	Start power increase from 96.5% / 790 MWe after "C" waterbox is returned to service.
	1945	Stop power increase at 100% / 845 MWe.
8/24/97	0438	Adjusting turbine load to maintain condenser vacuum with "A" waterbox being removed from service for cleaning.
	1620	Start power increase from 90% / 745 MWe after "A" waterbox is returned to service.
	1756	Stop power increase at 100% / 845 MWe.
8/31/97	2400	Unit 2 finishes the month at 100% / 845 MWe.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: August, 1997

TM S-97-013

Temporary Modification 8-4-97
(Safety Evaluation No. 97-096)

This Temporary Modification (TM) installs tygon tubing in place of flexible stainless steel tubing for the Emergency Service Water Pump (ESWP) flow indicators. This will allow troubleshooting of the oscillating flow indicator by visually inspecting the tubing to ensure the arrangement allows for proper venting of all air.

The currently installed tubing is 1/2 inch diameter Swagelok corrugated stainless tubing. The flow element and tubing are normally isolated from ESWP piping by a three inch ball valve. The flow element and instrumentation are only placed in service for pump testing. Normal operating pump discharge pressure is 13 psig with a system design pressure of 25 psig. The tygon tubing will be verified leak tight when placed in service. Should the tubing fail, the resultant leakage, a few gpm, is insignificant when compared to the pump design requirement of 14,020 gpm. Additionally the tygon tube can be isolated and will not affect the accuracy of the flow indication. Therefore an unreviewed safety question does not exist.

TM S2-97-07

Temporary Modification 8-4-97
(Safety Evaluation No. 97-100)

This Temporary Modification (TM) installs a temporary blower to provide additional ventilation to Main Steam Radiation Monitor 2-MS-RM-225. The monitor has had a history of failure that has been associated with the ambient temperature in which it operates. The manufacturer's recommended normal operating range for this device is 32 to 120 °F. The contact temperature on the radiation monitor shield box is about 150 °F.

The required performance characteristics of the radiation monitor will not be altered by improving the cooling of the shield enclosure. No air flow will be diverted away from safety related components. The 60 °F minimum temperature for the Aux Feedwater Pump lubricating oil will be met by securing the blower when the ambient temperature in the Main Steam Valve House reaches 70 °F. The blower will be secured to an existing structure to prevent movement during operation and a seismic event. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: August, 1997

JCO S1-97-002

Justification For Continued Operation 8-11-97
(Safety Evaluation No. 97-091 Rev. 1)

The Justification for Continued Operation (JCO) discusses continued Unit 1 operation with a containment hydrogen concentration limit of 0.8%. The evaluation supporting this change is documented in Engineering Transmittal (ET) NAF-970185, Rev. 0. The ET contains the following limiting conditions: 1) continued operation of the containment upper dome air fan and pressurizer cubicle ventilation while at power; 2) a hydrogen sampling program which confirms bulk concentrations do not exceed the 0.8% limit. These conditions will be tracked in accordance with JCO S1-97-002.

The results of Unit 1 containment sampling data have shown that the hydrogen concentration is well mixed from the containment basement to the operating floor. Adhering to the 0.8% hydrogen concentration limit provides high confidence that the accident analysis concentration limit will remain less than the 4% lower flammability limit. A sampling program will be maintained to confirm that hydrogen concentrations do not exceed the 0.8% value. The initial presence of 0.8% hydrogen in the containment will have a negligible impact on the containment peak pressure, depressurization and pump NPSH analyses. The self imposed limit of 0.8% is bounded by the conservatisms assumed in the post LOCA hydrogen analysis which include the one day hydrogen recombiner start time and the 50 scfm flowrate through the recombiner. Therefore an unreviewed safety question does not exist.

JCO S1-97-005

Justification For Continued Operation 8-11-97
(Safety Evaluation No. 97-099)

The Justification for Continued Operation (JCO) provides the justification of conditions and requirements for Letdown System operability. As previously identified, valve weights used in the piping seismic analyses were incorrect and non-conservative. The seismic pipe stress analysis and representative vibration analysis was performed with the revised valve weights. As documented by Engineering Transmittal, the piping stress was within code allowables, with a valve support exceeding allowables. Modifications have been made to the valve support to satisfy ASME III Appendix F criteria to allow continued operation until the next refueling outage in accordance with NRC Generic Letter 91-18.

This degraded and non-conforming condition has been reviewed and analyzed. The previous letdown operational restrictions and support modifications made provide reasonable assurance that the letdown piping system and supports meet the structural integrity requirements outlined in Generic Letter 91-18 Section 6.13. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: August, 1997

FS 95-034

Updated Final Safety Analysis Report Change 8-14-97
(Safety Evaluation 97-101)

Updated Final Safety Analysis Report (UFSAR) Change FS 95-034 revises UFSAR Section 14.3.1, Steam Generator Tube Rupture, because the current wording implies that operator action can accomplish the Steam Generator Tube Rupture (SGTR) procedure steps through the Safety Injection termination and initiation of post-SGTR cooldown within 30 minutes. In most cases this is impractical, as shown by simulator drills, and is not a requirement of the currently applicable analysis.

The UFSAR update clarifies the requirements imposed on operator action for coping with a SGTR accident. No plant hardware, systems, setpoints, procedures or operational methods are being impacted. Existing abnormal and emergency procedures remain unchanged. The SGTR analysis methodology, assumptions, results and conclusions are not changed. Therefore, an unreviewed safety question does not exist.

SE 97-104

Safety Evaluation 8-14-97

Safety Evaluation 97-104 was performed to evaluate equipment that has been Out of Service for greater than 30 days. Technical Specification related Heat Trace circuits 2A1-24, 26, 40, 43 and 2B1-28 have been out of service greater than 30 days. These circuits are on the boron injection flow path of record.

Redundant heat trace circuits maintain the associated boric acid solution temperature greater than Technical Specification limits to prevent solidification of boric acid. This is verified by Auxiliary Building logs. A Main Control Room annunciator alarms whenever the temperature of the heat trace circuits are not within required parameters. Therefore, an unreviewed safety question does not exist.

SE 97-005I

Safety Evaluation 8-14-97

Safety Evaluation 97-005I was performed for the Surry ISFSI Safety Analysis Report. During fabrication of TN-32 spent fuel storage casks, carbon steel shims are inserted between the inner vessel flange and the outer gamma shield during fabrication of the casks. The shims are required because of a small circumferential gap between the flange and the gamma shield forms during the fabrication process. The use of the shims was not part of the cask design addressed in the TN-32 Topical Safety Analysis Report.

Transnuclear has evaluated the impact of the shims on the design of the TN-32 cask and found that they do not affect the strength of the weld, have no effect on the subcritical margin of stored fuel and have no affect on the TN-32 thermal analysis. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: August, 1997

FS 93-033

Updated Final Safety Analysis Report Change 8-14-97
(Safety Evaluation 97-107)

Updated Final Safety Analysis Report (UFSAR) Change FS 93-033 revises chapter 8.6 by deleting the paragraph which reads "A preventative maintenance program conducted by VEPCO includes the periodic testing of insulation values of circuits and equipment." Virginia Power Engineering has determined that there is no technical basis for requiring a preventative maintenance program that periodically tests the insulation resistance of all circuits and equipment. Insulation resistance testing is a potentially destructive test which could actually have some adverse effects on the life of a component. A Licensing or Engineering basis for a preventative maintenance program which performs periodic insulation resistance testing on all circuits/equipment other than the UFSAR was not found.

The UFSAR request will clarify the UFSAR, chapter 8.6 with no procedures or other documents being affected. Therefore, an unreviewed safety question does not exist.

TM S1-97-14

Temporary Modification 8-18-97
(Safety Evaluation No. 97-105)

This Temporary Modification (TM) installs a Passive Autocatalytic Recombiner to control the presence of hydrogen in the Unit 1 containment.

This activity will not affect the Updated Final Safety Analysis Report (UFSAR) hydrogen generation analysis. The post-LOCA DBA hydrogen generation analysis does not exceed the 4% flammability limit with an assumed 50 cfm electric recombiner operating. This activity does not alter the installed accident electric recombiner capacity. However, operation of a temporary recombiner could decrease the consequences of malfunctions associated with a failure of the installed electric recombiners. Therefore, an unreviewed safety question does not exist.

FS 96-061

Updated Final Safety Analysis Report Change 8-21-97
(Safety Evaluation 97-109)

Updated Final Safety Analysis Report (UFSAR) FS 96-061 reformats and revises Section 5.3, Containment Systems and 9.13, Auxiliary Ventilation System. The Ventilation sections of the Surry UFSAR require updating to reflect the equipment, systems and system operation changes. Where appropriate, the Ventilation Systems have been regrouped and clarification of equipment served added. The Post-TMI Technical Support Center and Local Emergency Offsite Facility systems have been added. The testing of charcoal from the safety related filters has been changed to agree with Technical Specification Change 325.

The UFSAR descriptions are being altered to reflect actual station operation and system design. Therefore, an unreviewed safety question does not exist.

FACILITY CHANGES THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: August, 1997

FS 97-035

Updated Final Safety Analysis Report Change 8-21-97
(Safety Evaluation 97-110)

Updated Final Safety Analysis Report (UFSAR) Change FS 97-035 revises section 11.2 Radioactive Waste Systems, because an implied 35 mph wind speed restriction on process vent downwash is in error. Downwash concerns on the process vent have been analyzed by Environmental Policy and Compliance. The assessment used an EPA recommended model for nobel gas effluent concentrations referenced in the UFSAR. The implied 35 mph wind speed limitation was verified to be unsubstantiated. Wind speeds to 67 mph were evaluated.

Dose rates from downwash intuitively rise as ambient wind speed approaches zero. The assessment evaluated atmospheric Stability Classes A thru F. No violations or compromising or encroaching of the limits could happen. The radioactive gaseous effluent concentrations used in the analysis are bounding because they are much larger than what is released. Nominal Waste Gas Decay Tank releases involve 100 curies or less. The dispersion modeled by the downwash assessment coupled with the low concentrations results in excess margin. Therefore, an unreviewed safety question does not exist.

SE 97-113

Safety Evaluation 8-25-97

Safety Evaluation 97-113 was performed to determine the acceptability of the tree heights with respect to the meteorological tower and the resulting data. The UFSAR states the tree heights are 40 to 50 feet at the primary site and 10 to 15 feet at the secondary site. However, the tree heights are estimated to be 60 to 80 feet high. EPA guidance recommends that the measurement instruments should be placed on a tower in open terrain. Currently, Surry is not in compliance with this EPA document and the UFSAR.

The effect of different tree heights as stated in the UFSAR would result in wind speeds and atmospheric stability that are not representative of the actual conditions in the surrounding area. The calculated offsite doses from the met tower data would be conservative in the event of an accident. The met tower is not designated as a safety related component or system. Therefore, an unreviewed safety question does not exist.

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

MONTH/YEAR: August, 1997

0-FCA-1.00

Fire Contingency Action Procedure
(Safety Evaluation 97-093 Rev. 1) 8-11-97
(Safety Evaluation 97-093 Rev. 2) 8-14-97

Two separate areas of Appendix R non-compliance have been identified with respect to the 120 Volt Vital Bus System. The first issue deals with a fire in the main control room with no means to isolate the affected Vital Bus panel from the associated Uninterruptable Power Supplies (UPS). This postulated event could potentially affect the associated UPS and, therefore, affect the downstream Appendix R Panel. Revision 1 of the Safety Evaluation clarified the power supply for the excore flux monitor. Secondly, Chapter 9 of the Appendix R Report indicates that proper selective tripping is required for faults on Vital Bus branch circuits; a high fault current could result in the main breaker tripping simultaneously with, or in lieu of, the branch circuit breaker.

Compensatory measures implemented in the FCAs will a) disconnect the feeder conductors routed to the control room at UPSs 1A1, 1A2, 2A1, and 2A2 for a main control room fire to ensure the availability of distribution panels located in the ESGR 1 and 2 fire areas; Revision 1 included fuse replacements as necessary to ensure UPS restoration and b) restore Control Room or ESGR Vital Bus distribution panels lost as a result of fire induced hot shorts and mis-coordination between the main and branch circuit breakers as needed to accomplish safe shutdown. Revision 2 clarified that the compensatory measures can be accomplished within the Appendix R Report timeline. These compensatory measures do not adversely impact the Class 1E electrical distribution system. Therefore, an unreviewed safety question does not exist.

0-OP-HG-001

Operating Procedure 8/11/97
(Safety Evaluation 97-097)

Operating Procedure 0-OP-HG-001, "Unit 1 and 2 Main Generator Gas Sampling," is being revised to include the installation of procedurally controlled temporary modifications to facilitate calibration and analysis of the main generator hydrogen gas cooling system. This activity involves the use of temporary plastic tubing connections to facilitate safe collection of main generator hydrogen gas samples for collection and analysis.

The major consequence is the potential ignition of the hydrogen gas sample flow. The potential for uncontrolled ignition is minimized by utilizing the limited flow rate capability of the existing analyzer sample tubing, routing the sample purge flow to an external vent line, and stationing an operator by the sample valve for immediate isolation. Existing fire suppression controls in the area of the activity are designed for such an event. This activity does not affect the ability of the station to achieve and maintain safe shutdown under any conditions. Therefore, an unreviewed safety question does not exist.

**PROCEDURE OR METHOD OF OPERATION CHANGES
THAT DID NOT REQUIRE NRC APPROVAL**

MONTH/YEAR: August, 1997

2-NPT-RX-002
2-OPT-RX-006

Engineering Periodic Test Procedure
Operations Periodic Test Procedure
(Safety Evaluation 97-103)

8-14-97

Engineering Periodic Test Procedure 2-NPT-RX-002, "Reactor Core Flux Maps," and Operations Periodic Test Procedure 2-OPT-RX-006, "Rod Position Verification Using Incore Flux Mapping System," are being revised to provide a temporary modification which provides steps for jumpering out the safety limit switch on the "E" incore detector for Unit 2 which is not functioning properly. This switch prevents use of the "E" detector for the performance of a flux map.

Technical Specifications require a flux map be performed on a monthly basis and when conditions, such as a dropped control rod, require a flux map to measure core power distribution. The incore instrumentation system does not perform any safety related functions. This activity directly affects the operation of the moveable incore instrumentation system, as it jumpers out the safety limit switch for the "E" incore detector. The operation of the incore instrumentation system is not adversely impacted as a result of this activity, as cautions will be placed in the procedure to warn the performer to monitor the "E" detector while withdrawal in case of an additional component failure. Therefore, an unreviewed safety question does not exist.

1/2-OPT-RX-001
1/2-OPT-RX-002
1/2-OPT-RX-003
1/2-OPT-RX-004

Operations Periodic Test Procedures
(Safety Evaluation 96-108)

8-20-97

These procedures perform Reactor Power Calorimetrics using various methods. This change modifies the specified tolerance between Nuclear Instrumentation System (NIS) and calorimetric power from +2%/-0% to +4%/-0%. This limits the potential for downward NIS Gain adjustments at reduced power preventing the unintentional decalibration and loss of accuracy for the NIS channels when calorimetric alignment is done at reduced power.

The Safety Analysis Report (SAR) does not describe the adjustment of the power range drawers to match the calorimetric, nor does it describe any tolerance associated with the adjustment. This activity will change the tolerance to +4%/-0% between the power range detectors and the power calorimetric. As such, the change in the tolerance ensures the NIS will continue to operate the way it is described in the SAR, as it will ensure continued overpower protection. Therefore, an unreviewed safety question does not exist.

2-MOP-DG-001

Maintenance Operating Procedure
(Safety Evaluation 97-111)

8-21-97

Maintenance Operating Procedure 2-MOP-DG-001, "Removal and Return to Service of the PDTT for Maintenance" is being revised to install an instrument air jumper by-passing the SOV for the Primary Drain Transfer Tank (PDTT) Pressure Control Valve, keeping it in the open position. The valve must be maintained open to provide a flowpath for the RCS loop drains and other drains to the PDTT.

The jumper will be in place while Unit 2 is at Cold Shutdown (CSD) or Refueling Shutdown (RSD) and the Reactor Coolant System and containment are at atmospheric pressure; therefore there will be little or no pressure differential to cause gas flow from the PDTT to containment. The installation of this jumper does not affect containment isolation capability nor compliance of Technical Specification required systems. Therefore, an unreviewed safety question does not exist.

**PROCEDURE OR METHOD OF OPERATION CHANGES
THAT DID NOT REQUIRE NRC APPROVAL**

MONTH/YEAR: August, 1997

0-ECM-1509-03	**Electrical Corrective Maintenance Procedure** (Safety Evaluation 97-114)	8-29-97

Electrical Corrective Maintenance Procedure 0-ECM-1509-03, "Votes MOV Testing for Quarter Turn Valves," incorporates a One Time Only change to administratively control Component Cooling Water Heat Exchanger (CCHX) Service Water (SW) Supply MOV, 1-SW-MOV-102A, to maintain the capability to isolate the non-essential Service Water flowpaths within one hour. The one hour time limit for isolating the CCHX SW Supply are assumed as part of the design basis calculations associated with minimum initial canal level required to maintain the capability to supply adequate SW flow to the Recirculating Spray Heat Exchangers (RSHXs).

The administrative controls invoked provide acceptable contingency actions to ensure the non-essential SW isolation function can be met within the required design basis time frames assumed in calculation ME-0166. The administrative controls will be established to locally close the SW MOV. In event the MOV fails to fully close, local manual isolation of the CCHX SW supply is directed by 0-AP-12.01. Therefore, an unreviewed safety question does not exist.

TESTS AND EXPERIMENTS THAT DID NOT REQUIRE NRC APPROVAL

MONTH/YEAR: August, 1997

None During the Reporting Period

CHEMISTRY REPORT

MONTH/YEAR: August, 1997

Primary Coolant Analysis	Unit No. 1 Max.	Unit No. 1 Min.	Unit No. 1 Avg.	Unit No. 2 Max.	Unit No. 2 Min.	Unit No. 2 Avg.
Gross Radioactivity, μCi/ml	3.31E-1	1.73E-1	2.38E-1	2.63E-1	1.62E-1	2.04E-1
Suspended Solids, ppm	–	–	–	–	–	–
Gross Tritium, μCi/ml	6.13E-1	5.82E-1	5.92E-1	2.05E-1	1.55E-1	1.77E-1
I^{131}, μCi/ml	3.30E-4	2.14E-4	2.87E-4	1.56E-4	7.25E-5	1.16E-4
I^{131}/I^{133}	0.08	0.06	0.07	0.14	0.07	0.10
Hydrogen, cc/kg	31.0	26.2	28.9	29.8	25.3	27.4
Lithium, ppm	2.36	2.07	2.21	2.23	1.93	2.09
Boron - 10, ppm*	239.9	223.6	232.4	75.8	56.1	66.2
Oxygen, (DO), ppm	≤0.005	≤0.005	≤0.005	≤0.005	≤0.005	≤0.005
Chloride, ppm	0.007	0.003	0.004	0.005	≤0.001	0.002
pH at 25 degree Celsius	6.24	5.63	6.01	6.96	6.44	6.76

* Boron - 10 = Total Boron x 0.196

Comments:

None

FUEL HANDLING
UNITS 1 & 2

MONTH/YEAR: August, 1997

New Fuel Shipment or Cask No.	Date Stored or Received	Number of Assemblies per Shipment	Assembly Number	ANSI Number	Initial Enrichment	New or Spent Fuel Shipping Cask Activity
Unit 2 Batch 17 Shipment 1	8/19/97	12	30L	LM15A7	3.8950	15.00Ci
			46L	LM15AP	4.0631	
			15L	LM159S	3.9195	
			47L	LM15AQ	4.0643	
			28L	LM15A5	3.9000	
			07L	LM159J	3.9057	
			36L	LM15AD	4.0545	
			14L	LM159R	3.8985	
			16L	LM159T	3.8985	
			04L	LM159F	3.9179	
			23L	LM15A0	3.9104	
			44L	LM15AM	4.0650	
Unit 2 Batch 17 Shipment 2	8/21/97	12	18L	LM159V	3.8948	14.88Ci
			31L	LM15A8	3.9006	
			01L	LM159C	3.9095	
			11L	LM159N	3.8992	
			19L	LM159W	3.9047	
			10L	LM159M	3.8989	
			13L	LM159Q	3.8996	
			17L	LM159U	3.8984	
			40L	LM15AH	3.0640	
			27L	LM15A4	3.9003	
			12L	LM159P	3.8954	
			43L	LM15AL	4.0657	

FUEL HANDLING
UNITS 1 & 2

MONTH/YEAR: August, 1997

New Fuel Shipment or Cask No.	Date Stored or Received	Number of Assemblies per Shipment	Assembly Number	ANSI Number	Initial Enrichment	New or Spent Fuel Shipping Cask Activity
Unit 2 Batch 17 Shipment 3	8/25/97	12	02L	LM159D	3.9170	15.10 Ci
			03L	LM159E	3.9156	
			55L	LM15AY	4.0556	
			51L	LM15AU	4.0652	
			08L	LM159K	3.9093	
			20L	LM159X	3.9043	
			29L	LM15A6	3.9006	
			32L	LM15A9	3.9002	
			49L	LM15AS	4.0638	
			54L	LM15AX	4.05880	
			50L	LM15AT	4.0622	
			52L	LM15AV	4.0591	
Unit 2 Batch 17 Shipment 4	8/26/97	12	37L	LM15AE	4.0602	15.14Ci
			34L	LM15AB	4.0473	
			26L	LM15A3	3.9031	
			42L	LM15AK	4.0714	
			38L	LM15AF	4.0600	
			39L	LM15AG	4.0621	
			22L	LM159Z	3.9156	
			53L	LM15AW	4.0653	
			35L	LM15AC	4.0551	
			41L	LM15AJ	4.0622	
			56L	LM15AZ	4.0593	
			48L	LM15AR	4.0651	

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

MONTH/YEAR: AUGUST, 1997

None During the Reporting Period