TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

J. L. Wilson
Vice President, Sequoyah Nuclear Plant

April 15, 1992

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - MARCH 1992 MONTHLY OPERATING REPORT

Enclosed is the March 1992 Monthly Operating Report as required by SQN Technical Specification 6.9.1.10.

If you have any questions concerning this matter, please call M. A. Cooper at (615) 843-8924.

Sincerely,

R. J. Beecken for
J. L. Wilson

Enclosure
cc: See page 2

IE24

cc (Enclosure):

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TENNESSEE VALLEY AUTHORITY

NUCLEAR POWER GROUP
SEQUOYAH NUCLEAR PLANT

MONTHLY OPERATING REPORT

TO THE

NUCLEAR REGULATORY COMMISSION

<u>MARCH 1992</u>

UNIT 1

DOCKET NUMBER 50-327

LICENSE NUMBER DPR-77

UNIT 2

DOCKET NUMBER 50-328

LICENSE NUMBER DPR-79

OPERATIONAL SUMMARY
MARCH 1992

UNIT 1

Unit 1 generated 499,580 megawatthours (MWh) (gross) electrical power during March, with a capacity factor of 57.79 percent. On March 6 at 1622 Eastern standard time (EST), a power level decrease was initiated because of delta T/T_{avg} problems associated with cable-induced spiking of the overpower and overtemperature setpoints. At 1822 EST on March 6, Unit 1 reactor power was at approximately 81 percent. Unit 1 reactor power remained at 81 percent until 1030 EST on March 8, when a power level increase was initiated. Unit 1 was again operating at 100 percent reactor power level on March 8 at 1725 EST.

On March 18 at 2048 EST, Limited Condition for Operation 3.6.5.3 was entered after 11 of 48 ice condenser lower inlet doors were found to require excessive force to open. On March 18 at 2210 EST, shutdown of Unit 1 was initiated. Unit 1 entered Mode 3 at 0247 EST.

On March 19 at 1347 EST, with Unit 1 still in Mode 3, it was determined that there was a feedwater leak on the steam generator (SG) No. 3 line that could not be repaired at this temperature. Unit 1 was taken to Mode 5 at 0205 EST on March 20 to allow further investigation, testing, and maintenance. After Unit 1 was taken to cold shutdown, an inspection of the feedwater line identified a circumferential crack at the transition piece between the SG's nozzle and feedwater line. Radiographic examination also revealed transition piece cracking on Unit 1 SG No. 4 and Unit 2 SG Nos. 1, 3, and 4. A decision was made to replace the transition pieces on all eight Units 1 and 2 SGs. Unit 1 remained in Mode 5 at the end of March.

UNIT 2

Unit 2 generated 271,110 MWh (gross) electrical power during March, with a capacity factor of 31.36 percent. Unit 2 was at approximately 79 percent reactor power level at the beginning of March and was in coastdown to Unit 2 Cycle 5 refueling outage. On March 13 at 1700 EST, a power level decrease was initiated to bring the unit offline for the outage. Unit 2 entered Mode 2 at 2125 EST and Mode 3 at 2130 EST on March 13. Mode 4 was entered on March 14 at 1101 EST, and Mode 5 was entered at 0902 EST on March 15.

Unit 2 entered Mode 6 on March 20 at 2158 EST, when the first reactor vessel head bolt was detensioned. On March 23 at 0550 EST, the reactor vessel head was lifted and inspected. Core offload began on March 26 at 0710 EST; core offload was completed on March 29 at 0410 EST. Unit 2 remained defueled in "no mode" at the end of March.

POWER-OPERATED RELIEF VALVES (PORV) AND SAFETY VALVES SUMMARY

There were no challenges to PORVs or safety valves in March.

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-327 UNIT No. One DATE: 04-03-92

COMPLETED BY: T. J. Hollomon TELEPHONE: (615) 843-7528

MONTH: MARCH 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	1142	17	1141
2	1143	18	1122
3	1143	19	20
4	1105	20	-7
5	1124	21	-9
6	1086	22	-7
7	939	23	-7
8	1028	24	-5
9	1142	25	-7
10	1143	26	-14
11	1143	27	-7
12	1144	28	-14
13	1144	29	-5
14	1142	30	-5
15	1139	31	-7
16	1137		

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-328 UNIT No. Two DATE: 04-03-92

COMPLETED BY: T. J. Hollomon TELEPHONE: (615) 843-7528

MONTH: MARCH 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	892	17	-7
2	885	18	-9
3	872	19	-16
4	868	20	-14
5	861	21	-7
6	856	22	-9
7	850	23	-9
8	841	24	-9
9	834	25	-7
10	829	26	-9
11	821	27	-12
12	814	28	-16
13	656	29	-9
14	-19	30	-7
15	-16	31	-9
16	-9		

OPERATING DATA REPORT

DOCKET NO. 50-327
DATE Apr. 3, 1992
COMPLETED BY T. J. Hollomon
TELEPHONE (615) 843-7528

OPERATING STATUS

Notes

1. Unit Name: Sequoyah Unit One
2. Reporting Period: March 1992
3. Licensed Thermal Power (MWt): 3411.0
4. Nameplate Rating (Gross MWe): 1220.6
5. Design Electrical Rating (Net MWe): 1148.0
6. Maximum Dependable Capacity (Gross MWe): 1162.0
7. Maximum Dependable Capacity (Net MWe): 1122.0
8. If Changes Occur in Capacity Ratings (Item Numbers 3 Through 7) Since Last Report, Give Reasons:

9. Power Level To Which Restricted, if Any (Net MWe): N/A
10. Reasons For Restrictions, If Any: N/A

	This Month	Yr-to-Date	Cumulative
11. Hours in Reporting Period	744	2,184	94,249
12. Number of Hours Reactor Was Critical	434.3	1,874.3	48,828.0
13. Reactor Reserve Shutdown Hours	0	0	0
14. Hours Generator On-Line	434.2	1,874.2	47,745.3
15. Unit Reserve Shutdown Hours	0.0	0	0
16. Gross Thermal Energy Generated (MWH)	1,440,860.1	5,974,236.8	155,586,770
17. Gross Electrical Energy Generated (MWH)	499,580	2,064,768	52,732,264
18. Net Electrical Energy Generated (MWH)	479,972	1,987,605	50,522,339
19. Unit Service Factor	58.4	85.8	50.7
20. Unit Availability Factor	58.4	85.8	50.7
21. Unit Capacity Factor (Using MDC Net)	57.5	81.1	47.8
22. Unit Capacity Factor (Using DER Net)	56.2	79.3	46.7
23. Unit Forced Outage Rate	41.6	14.2	40.8

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 Startup: April 18, 1992

OPERATING DATA REPORT

DOCKET NO. 50-328
DATE Apr. 3, 1992
COMPLETED BY T. J. Hollomon
TELEPHONE (615) 843-7528

OPERATING STATUS

1. Unit Name: Sequoyah Unit Two
2. Reporting Period: March 1992
3. Licensed Thermal Power (MWt): 3411.0
4. Nameplate Rating (Gross MWe): 1220.6
5. Design Electrical Rating (Net MWe): 1148.0
6. Maximum Dependable Capacity (Gross MWe): 1162.0
7. Maximum Dependable Capacity (Net MWe): 1122.0
8. If Changes Occur in Capacity Ratings (Item Numbers 3 Through 7) Since Last Report, Give Reasons:

Notes

9. Power Level To Which Restricted, If Any (Net MWe): N/A
10. Reasons For Restrictions, If Any: N/A

	This Month	Yr-to-Date	Cumulative
11. Hours in Reporting Period	744	2,184	86,209
12. Number of Hours Reactor Was Critical	309.5	1,717.5	50,726
13. Reactor Reserve Shutdown Hours	0	0	0
14. Hours Generator On-Line	309.3	1,704.5	49,747.7
15. Unit Reserve Shutdown Hours	0.0	0	0
16. Gross Thermal Energy Generated (MWH)	793,075.1	5,160,732.5	156,088,045
17. Gross Electrical Energy Generated (MWH)	271,110	1,762,266	52,920,557
18. Net Electrical Energy Generated (MWH)	254,850	1,691,711	50,636,67[illegible]
19. Unit Service Factor	41.6	78.0	57.7
20. Unit Availability Factor	41.6	78.0	57.7
21. Unit Capacity Factor (Using MDC Net)	30.5	69.0	52.4
22. Unit Capacity Factor (Using DER Net)	29.8	67.5	51.2
23. Unit Forced Outage Rate	0.0	2.6	35.3

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):
Unit 2 Cycle 5 refueling outage began March 13, 1992.

25. If Shut Down At End Of Report Period, Estimated Date of Startup: May 17, 1992

DOCKET NO: 50-32/
UNIT NAME: One
DATE: 04/03/92
COMPLETED BY: T. J. Hollomon
TELEPHONE: (615) 843-7528

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH: March 1992

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report No.	System Code[4]	Component Code[5]	Cause and Corrective Action to Prevent Recurrence
3	3/6/92	F	N/A	B	5	N/A	N/A	N/A	On 3/6/92 at 1622 EST, Unit 1 power level was reduced to approximately 81 percent because of delta T/T_{avg} problems. A power level increase was initiated at 1030 EST on 3/8/92. Unit 1 was at 100 percent on 3/8/92 at 1725 EST.
4	3/19/92	F	309.8	B	1	327/92007	BC	DR	On 3/19/92 at 0210 EST, Unit 1 was taken offline after inspection of ice condenser lower inlet doors showed 11 of 48 doors required excessive force to open. Unit 1 entered Mode 3 on 3/19/92 at 0247 EST. On 3/19/92 at 1347 EST, with Unit 1 in Mode 3, a feedwater leak was identified on the SG No. 3 feedwater line. Unit 1 was taken to Mode 5 at 0205 EST on 3/20/92 to allow further investigation, testing, and maintenance. A decision was made to replace the transition pieces.

[1]F: Forced
S: Scheduled

[2]Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restruction
E-Operator Training and License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

[3]Method:
1-Manual
2-Manual Scram
3-Automatic Scram
4-Continuation of Existing Outage
5-Reduction
9-Other

[4]Exhibit G-Instructions for Preparation of Data Entry sheets for Licensee Event Report (LER) File (NUREG-1022)

[5]Exhibit I-Same Source

DOCKET NO: 50-328
UNIT NAME: Two
DATE: 04/03/92
COMPLETED BY: T. J. Hollomon
TELEPHONE: (615) 843-7528

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH: March 1992

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report No.	System Code[4]	Component Code[5]	Cause and Corrective Action to Prevent Recurrence
2	3/13/92	S	434.7	C	1	N/A	N/A	N/A	Unit 2 entered Cycle 5 refueling outage at 2120 EST on 3/13/92 when the unit was removed from the grid. Unit 2 Cycle 5 continues.

[1] F: Forced
S: Scheduled

[2] Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restruction
E-Operator Training and License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

[3] Method:
1-Manual
2-Manual Scram
3-Automatic Scram
4-Continuation of Existing Outage
5-Reduction
9-Other

[4] Exhibit G-Instructions for Preparation of Data Entry sheets for Licensee Event Report (LER) File (NUREG-1022)

[5] Exhibit I-Same Source

PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Hope Creek Generating Station

April 15, 1992

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

MONTHLY OPERATING REPORT
HOPE CREEK GENERATION STATION UNIT 1
DOCKET NO. 50-354

In compliance with Section 6.9, Reporting Requirements for the Hope Creek Technical Specifications, the operating statistics for March are being forwarded to you along with the summary of changes, tests, and experiments for March 1992 persuant to the requirements of 10CFR50.59(b).

Sincerely yours,

J. J. Hagan
General Manager -
Hope Creek Operations

RAR:ld
Attachments

C Distribution

240059
The Energy People

9204240103 920331
PDR ADOCK 05000354
R PDR

INDEX

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-354
UNIT Hope Creek
DATE 4/15/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH March 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1.	1038	17.	61
2.	1062	18.	495
3.	1057	19.	857
4.	1054	20.	1045
5.	1054	21.	1068
6.	924	22.	1070
7.	0	23.	1069
8.	0	24.	1065
9.	0	25.	1063
10.	0	26.	1056
11.	0	27.	1059
12.	0	28.	1049
13.	0	29.	1059
14.	0	30.	1061
15.	0	31.	1060
16.	0		

OPERATING DATA REPORT

DOCKET NO. 50-354
UNIT Hope Creek
DATE 4/15/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

OPERATING STATUS

1. Reporting Period March 1992 Gross Hours in Report Period 744

2. Currently Authorized Power Level (MWt) 3293
Max. Depend. Capacity (MWe-Net) 1031
Design Electrical Rating (MWe-Net) 1067

3. Power Level to which restricted (if any) (MWe-Net) None

4. Reasons for restriction (if any)

	This Month	Yr To Date	Cumulative
5. No. of hours reactor was critical	522.8	1962.8	39,124.1
6. Reactor reserve shutdown hours	0.0	0.0	0.0
7. Hours generator on line	495.2	1935.2	38,509.8
8. Unit reserve shutdown hours	0.0	0.0	0.0
9. Gross thermal energy generated (MWH)	1,523,901	6,255,121	122,252,263
10. Gross electrical energy generated (MWH)	510,040	2,095,580	40,448,074
11. Net electrical energy generated	483,625	2,002,931	38,651,480
12. Reactor service factor	70.3	89.9	84.5
13. Reactor availability factor	70.3	89.9	84.5
14. Unit service factor	66.6	88.6	83.2
15. Unit availability factor	66.6	88.6	83.2
16. Unit capacity factor (using MDC)	63.0	89.0	81.0
17. Unit capacity factor (Using Design MWe)	60.9	86.0	78.3
18. Unit forced outage rate	0.0	0.0	4.9

19. Shutdowns scheduled over next 6 months (type, date, & duration):
Refueling outage, 9/12/92, 60 days

20. If shutdown at end of report period, estimated date of start-up:
N/A

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 4/15/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH March 1992

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
1	3/6	S	248.8	B	1	Scheduled Maintenance Outage

Summary

REFUELING INFORMATION

DOCKET NO. 50-354
UNIT Hope Creek
DATE 4/15/92
COMPLETED BY S. Hollingsworth
TELEPHONE (609) 339-1051

MONTH March 1992

1. Refueling information has changed from last month:

   Yes No X

2. Scheduled date for next refueling: 9/12/92

3. Scheduled date for restart following refueling: 11/11/92

4. A. Will Technical Specification changes or other license amendments be required?

   Yes No X

   B. Has the reload fuel design been reviewed by the Station Operating Review Committee?

   Yes No X

   If no, when is it scheduled? not scheduled (on or prior to 7/24/92)

5. Scheduled date(s) for submitting proposed licensing action: N/A

6. Important licensing considerations associated with refueling:

   - Same fresh fuel as current cycle: no new considerations

7. Number of Fuel Assemblies:

A.	Incore	764
B.	In Spent Fuel Storage (prior to refueling)	760
C.	In Spent Fuel Storage (after refueling)	1008

8. Present licensed spent fuel storage capacity: 4006

   Future spent fuel storage capacity: 4006

9. Date of last refueling that can be discharged to spent fuel pool assuming the present licensed capacity: (does not allow for full-core offload) 11/4, 2010 (EOC16)

HOPE CREEK GENERATING STATION

MONTHLY OPERATING SUMMARY

March 1992

Hope Creek entered the month of March at approximately 100% power. The unit operated until March 7, when it was manually shutdown for a scheduled maintenance outage. The plant completed its 300th day of continuous power operation prior to the shutdown. The reactor went critical on March 16 and the plant was brought back on-line at 1412 on March 17. The plant operated for the remainder of the month without experiencing any other shutdowns or reportable power reductions. On March 31, the plant completed its 14th day of continuous power operation.

SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS

FOR THE HOPE CREEK GENERATING STATION

MARCH 1992

The following items have been evaluated to determine:

1. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

2. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

3. If the margin of safety as defined in the basis for any technical specification is reduced.

The 10CFR50.59 Safety Evaluations showed that these items did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These items did not change the plant effluent releases and did not alter the existing environmental impact. The 10CFR50.59 Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

DCP	Description of Safety Evaluation
4EC-3112/13	This DCP replaced motor operated butterfly valves in the 'B' Service Water Pump and Strainer. They were replaced with upgraded butterfly valves that are designed with metal seats for extended wear and minimal maintenance. The internal control logic of the operators has been revised to incorporate torque seated valve operation instead of limit switch seated valve operation. No Unreviewed Safety Questions were involved because the valve replacement and modifications do not adversely affect the safety-related function of the Service Water piping. Additionally, the use of a metal seated valve will provide improved shutoff and performance resulting in minimizing the potential for strainer basket damage due to valve seat failure.
4EC-3114/06	This DCP replaced Service Water Strainer Backwash System spool pieces with 6% molybdenum stainless steel piping. Additionally, a butterfly valve will be replaced with an upgraded butterfly valve that is designed with metal seats for extended wear and minimal maintenance. The internal control logic of the operator has been revised to incorporate torque seated valve operation instead of limit switch seated valve operation. No Unreviewed Safety Questions were involved because the spool and valve replacement and modifications do not adversely affect the safety-related function of the Service Water piping. Additionally, the use of the new material piping and a metal seated valve will provide improved performance resulting in minimal maintenance.
4EC-3114/08	This DCP replaced Service Water Strainer Backwash System spool pieces with 6% molybdenum stainless steel piping. Additionally, a butterfly valve will be replaced with an upgraded butterfly valve that is designed with metal seats for extended wear and minimal maintenance. The internal control logic of the operator has been revised to incorporate torque seated valve operation instead of limit switch seated valve operation. No Unreviewed Safety Questions were involved because the spool and valve replacement and modifications do not adversely affect the safety-related function of the Service Water piping. Additionally, the use of the new material piping and a metal seated valve will provide improved performance resulting in minimal maintenance.

<u>DCP</u>	<u>Description of Safety Evaluation</u>
4EC-3226	This DCP modified the logic of the 'E' and 'F' Filtration, Recirculation, and Ventilation System Recirculation Fans. A permissive interlock was added to the low flow automatic start signal. The permissive is automatically activated by the automatic start signals for the Filtration, Recirculation, and Ventilation System train. The permissive is reset when the 'A', 'B', 'C', and 'D' fans are stopped and the HI RAD or LOCA signals are no longer present. No Unreviewed Safety Questions were involved because all of the emergency starting circuits remain as originally designed and the design does not change the number of Filtration, Recirculation, and Ventilation System Recirculation Fans available for normal or emergency operation.
4EC-3326	This DCP replaced the ring-lug termination of the servo-valve and position transducer wiring of valves in the Turbine Generator System. This provides additional support at the crimp point of the connections. This DCP did not change any of the operating parameters or the controlling functions of the equipment. The sole purpose was to provide a more reliable termination; therefore, no Unreviewed Safety Questions were involved with this DCP.
4HC-0212/05	This DCP installed a 600 gpm mechanical side-stream filtering unit with a 40 psid differential pressure monitor and all of the associated piping, fittings, valves, and instrumentation. The filter will cleanse the Turbine Building Chilled Water system of insoluble solid impurities and lubrication oil that hampers system performance and reduces the service life of the equipment. The Turbine Building Chilled Water system has no safety related functions with the exception of the isolation valves located at the Drywell penetrations. The proposed modification has no impact to the main flow to the pumps and the design includes a bypass back to the suction header of the pumps in the event that the filter is down. Therefore, no Unreviewed Safety Questions were involved with this DCP.

TMR	Description of Safety Evaluation
92-001	This TMR installed electrical jumpers across the Feedwater Heater's High High Level Trip Switches. These switches cause spurious high level trip signals during low power levels due to inleakage in the reference leg. The jumpers were removed after the level signals stabilized. The Feedwater system is not safety related and is not required to be operable following a LOCA, other than for containment isolation. Failure of the Feedwater system does not compromise any safety related system or components. This TMR has no impact on the containment isolation function of the Feedwater system.
92-003	This TMR removed the overload heaters from the breakers for the Reactor Water Cleanup Discharge to Condenser Valve and the Reactor Water Cleanup Discharge to Equipment Drain Valve. Removing the overload heaters from the breakers will prevent the valves from inadvertently opening during an Appendix R fire. Disabling these valves, along with the overhead annunciator, does not prevent their associated systems from performing their designed functions. Also, the UFSAR discusses the Appendix R requirement that the valves be disabled.

TMR	Description of Safety Evaluation
92-006	This TMR eliminated the possibility of a Residual Heat Removal Shutdown Cooling isolation due to a loss of the 'A' Reactor Protection System power during an outage with the plant in cold shutdown. The TMR eliminated all automatic isolation signals to the Shutdown Cooling Suction Inboard Isolation Valve powered by the 'A' Reactor Protection System. It also removed one of two Reactor High Pressure interlock signals (powered from the 'A' Reactor Protection System) from automatically closing the Shutdown Cooling Outboard Suction Valve, the two Shutdown Cooling Injection Valves, and the Head Spray Outboard Isolation Valve. The TMR was removed prior to plant startup. The isolation of the Residual Heat Removal Shutdown Cooling Suction and Injection Lines is provided by motor operated gate valves that are interlocked closed by a Reactor High Pressure signal during normal operation and are automatically closed during an accident by Low Water Level isolation signals. The removal of automatic controls from these valves did not create the possibility of the Shutdown Cooling lines not isolating. Although part of the automatic isolation capability is removed, manual capability remains. The isolation logic channels are not required to be operable in cold shutdown per the Technical Specifications. Therefore, no Unreviewed Safety Questions are associated with this TMR.

Procedure Revision	Description of Safety Evaluation
HC.IC-LC.AE-0005(Q) Rev 0	This procedure performs a loop calibration of the feedwater flow transmitters to the feedwater flow computer points. Individual components are adjusted as required if the loop is out of tolerance. This procedure installs jumpers to bypass the 20% total feedwater flow interlock to the recirculation pump speed limiter to preclude an actual recirculation runback from occurring during the transmitter calibration. The use of jumpers to bypass the recirculation pump speed runback is the only item in the procedure that is a change as described in the SAR. The recirculation runback logic does not make a significant contribution to the mitigation of the accident and it does not challenge the core thermal margins or vessel pressure boundary before the scram. Therefore, there are no Unreviewed Safety Questions associated with this new procedure.
HC.SS-IS.ZZ-0010(Q) Rev 0	This procedure provides methods to test valves to ensure pressure isolation valve leakage rates, primary containment leakage rates, and individual valve leakage rates are met. This procedure changes the facility by installing jumpers and lifting leads to defeat the High Pressure Coolant Injection and Reactor Core Isolation Cooling Low Steam Supply Pressure signals. The procedure also rotates a Breathing Air Spectacle flange, replaces the Primary Containment Instrument Gas Drywell Intake Screen with a test flange, and installs a test plug in the High Pressure Coolant Injection Suppression Pool. The use of this procedure does not increase the potential for draining the Reactor because the Prerequisites require that the Main Steam Line Plugs are installed or that the Reactor Vessel level is maintained below the Main Steam Line Nozzles. Also, adequate precautions are taken to prevent equipment malfunction during the performance of this procedure; therefore, no Unreviewed Safety Questions are involved.

UFSAR Section	Description of Safety Evaluation
Table 11.5-1 Table 12.3-9	During the Radiation Monitoring System Configuration Baseline Document review, several typographical and editorial deficiencies were identified. One of the resolutions involves updating the UFSAR to reflect the correct information. The extent of these changes are correcting editorial and typographical errors. Therefore, no Unreviewed Safety Questions are involved with this UFSAR change.