METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 206

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY: JJBroughton
Vice President and Director, TMI-1

Sworn and subscribed before me this
24th day of June, 1992.

Erin M. Flowers
Notary Public

Notarial Seal
Erin M. Flowers, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Sept. 11, 1993
Member, Pennsylvania Association of Notaries

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF
GPU NUCLEAR CORPORATION

DOCKET NO. 50-289
LICENSE NO. DPR-50

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 206 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Jay H. Kopp, Chairman
Board of Supervisors of
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Mr. Russell L. Sheaffer, Chairman
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GPU NUCLEAR CORPORATION

BY: J G Broughton
Vice President & Director, TMI-1

DATE: June 24, 1992

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 206

GPUN requests that the following changed replacement pages be inserted into the existing Technical Specification:

Revised pages: 1-2, 1-7, 1-8, 4-3, 4-4, 4-5, 4-5a, 4 6, 4-7, 4-7a, 4-8,4-10a, 4-38, 4-55, 4-55b, 4-60

New Page: 4-10b

These pages are attached to this change request.

II. REASON FOR CHANGE

The purpose of this Technical Specification Change Request (TSCR) is to revise the Technical Specifications to implement a 24 month plant cycle by changing the surveillance interval for Technical Specification surveillance requirements that are generally performed during a refueling outage. Technical Specification surveillance frequency requirements which have not yet been evaluated for extension are also being revised generically to limit the frequency to the existing refueling interval definition. The following Technical Specification changes are proposed:

Technical Specification Definition 1.2.8, REFUELING INTERVAL, is revised to clarify that the refueling interval is the time between normal refuelings of the reactor, or at least once per 24 months.

Technical Specification Table 1.2 is revised to clarify that the Refueling Interval (R) is defined as "once per 24 months." This table is also revised to add Notation "F" defined as "Not to exceed 24 months," which is the current refueling interval definition, for those surveillance requirements which have not yet been evaluated for extension. Technical Specification definition 1.25 is administratively revised to indicate that the 25% interval extension is not applicable to "F" designated intervals.

Technical Specification Table 4.1-1, Items 7, 10, 13, 19.a, 19.d, 19.e, 21.a, 25.a, 25.b, 27, 29, 30, 31.a, 31.b, 32.a, 32.b, 33, 35, 40, 41, 46, 47.a, 50, 51.a.1, 51.a.3, 53, 54.a, and 54.b are revised to specify that the calibration or test requirements shall be performed at intervals not to exceed 24 months since justification for extension is not provided.

Technical Specification Table 4.1-2, Item 10 is revised to specify that the visual inspection of the Intake Pump House Floor shall be performed at intervals not to exceed 24 months since justification for extension is not provided. Table 4.1-2, Item 4 is revised to specify approximately 50% of valves are to be tested.

Technical Specification Table 4.1-4, Items 1, 3, 5, 6, and 11 are revised to specify that the calibration requirements shall be performed at intervals not to exceed 24 months since justification for extension is not provided.

Technical Specification 4.4.4.1.b.1 is revised to clarify that the calibration requirements for Hydrogen Recombiner instrumentation and control circuits shall be performed at intervals not to exceed 24 months since justification for extension is not provided.

Technical Specification Sections 4.12.1.1 and 4.12.1.3 are revised to extend the Emergency Control Room Air Treatment System testing from once every 18 months to at least every refueling interval consistent with the 24 month plant cycle.

Technical Specification Sections 4.12.2.1 and 4.12.2.2 are revised to extend the Reactor Building Purge Air Treatment System ΔP testing and sample analysis to at least once per refueling interval consistent with the 24 month plant cycle.

Technical Specification 4.17.1.b is revised to extend subsequent visual inspections of safety related snubbers from 18 months $\pm$ 25% to 24 months $\pm$ 25% with zero (0) inoperable snubbers per inspection period, and from 12 months $\pm$ 25% to 24 months $\pm$ 25% with one (1) inoperable snubber per inspection period consistent with the 24 month plant cycle.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The following discussion supports the Technical Specification changes identified above. The discussion also provides the supporting justification for the component or system surveillance requirements which are specified on a refueling interval basis and are therefore, being extended to at least once per 24 months by the proposed change to the Technical Specification refueling interval definition in Section 1.2.8. The guidance contained in NRC Generic Letter 91-04, Enclosure 2, has been addressed in the evaluation of increased instrument calibration intervals. Included in this discussion are the following Technical Specification surveillance history evaluations completed to date:

Table 4.1-1	Item 8	High Reactor Coolant Pressure Channel Calibration
Table 4.1-1	Item 9	Low Reactor Coolant Pressure Channel Calibration
Table 4.1-1	Item 12	Pump Flux Comparator Calibration
Table 4.1-1	Item 15.a	High Pressure Injection Analog Channel - Reactor Coolant Pressure Channel Calibration
Table 4.1-1	Item 17.a	Low Pressure Injection Analog Channel - Reactor Coolant Pressure Channel Calibration
Table 4.1-1	Item 19.f	Reactor Building Emergency Cooling and Isolation System Analog Channels - Line Break Isolation Signal (ICCW & NSCCW) Calibration

Table 4.1-1	Item 22	Pressurizer Temperature Channel Calibration
Table 4.1-1	Item 23	Control Rod Absolute Position Calibration
Table 4.1-1	Item 24	Control Rod Relative Position Calibration
Table 4.1-1	Item 26	Pressurizer Level Channel Calibration
Table 4.1-1	Item 37	Reactor Building Sump Level Calibration
Table 4.1-1	Item 38	OTSG Full Range Level Calibration
Table 4.1-1	Item 39	Turbine Overspeed Test
Table 4.1-1	Item 42	Diesel Generator Protective Relaying Calibration
Table 4.1-1	Item 43.a & Item 43.b	4KV ES Bus Undervoltage Relays - Degraded Grid Calibration and Loss of Voltage Calibration
Table 4.1-1	Item 44	Reactor Coolant Pressure DH Valve Interlock Bistable Calibration
Table 4.1-1	Item 45	Loss of Feedwater Reactor Trip Calibration
Table 4.1-1	Item 47.b	PORV-Acoustic Flow Calibration
Table 4.1-1	Item 48	PORV Setpoints Calibration
Table 4.1-1	Item 49	Saturation Margin Monitor Calibration
Table 4.1-1	Item 51.a.2	Heat Sink Protection System, EFW Auto Initiation Channel - Loss of All RC Pumps Calibration
Table 4.1-1	Item 51.a.4	Heat Sink Protection System, EFW Auto Initiation Channel - OTSG Low Level Calibration
Table 4.1-1	Item 51.b	Heat Sink Protection System - MFW Isolation OTSG Low Pressure Calibration
Table 4.1-1	Item 51.c.1	Heat Sink Protection System, EFW Control Valve Control System - OTSG Level Loop Calibration
Table 4.1-1	Item 51.c.2	Heat Sink Protection System EFW Control Valve Control System - Controllers Calibration
Table 4.1-1	Item 51.d	Heat Sink Protection System - HSPS Train Actuation Logic Calibration

Table 4.1-1	Item 52	Backup Incore Thermocouple Display - Calibration
Table 4.1-2	Item 1	Control Rod Drop Time Test
Table 4.1-2	Item 3	Press rizer Safety Valves Setpoint Test
Table 4.1-2	Item 4	Main Steam Safety Valves Setpoint Test
Table 4.1-4	Item 2	Containment High Range Radiation - Calibration
Table 4.1-4	Item 4	Containment Water Level - Calibration
Table 4.1-4	Item 7&8	RCS Cold Leg and Hot Leg Water Temperatures
Table 4.1-4	Item 9	RCS Pressure
Table 4.1-4	Item 10	Steam Generator Pressure - Calibration
4.4.1.3		Reactor Building Isolation Valve Functional Tests
4.4.1.7		Reactor Building Purge Valve Seat Inspection/Test
4.4.4.1.b.2		Hydrogen Recombiner System Visual Examination
4.4.4.1.b.3		Reaction Chamber Gas Temperature Test
4.4.4.1.b.4		Hydrogen Recombiner System Heater Electrical Test
4.5.1.1.a		Emergency Loading Sequence and Power Transfer Test
4.5.2.1.a		High Pressure Injection System Test
4.5.2.2.a		Low Pressure Injection System Test
4.5.2.3		Core Flooding System Test
4.5.3.1.a.1		Reactor Building Spray System Test
4.5.3.1.b.1		Reactor Building Cooling and Isolation System Test
4.5.4.2		Decay Heat Removal System Tests
4.6.1.b		Emergency Diesel Generator Auto Start and Load Test
4.6.2.d		Station Batteries Load Test

4.6.3.a	Pressurizer Heaters Tests
4.7.1.1	Control Rod Trip Insertion Time Test
4.8.2	Main Steam Isolation Valves Closure Time Test
4.9.1.4 4.9.1.5	EFW Pump and Control Valve Tests
4.11.1	Reactor Coolant System Vent Valve Test
4.12.1.1	Emergency Control Room Air Treatment System Δ P Flow Test
4.12.1.3	Control Building Isolation and Recirculation Damper Test
4.12.2.1 & 4.12.2.2.e	Reactor Building Purge Air Treatment System Δ P and Flow Test
4.12.3.1	Auxiliary and Fuel Handling Building Air Treatment System Δ P Flow Test and Sample Analysis
4.16.1	Reactor Internals Vent Valves Inspection/Test
4.17.1.b & e	Snubber Visual Inspection and Functional Test

CHANGE NO.1 - TECHNICAL SPECIFICATION DEFINITION

Technical Specification Definition 1.2.8, REFUELING INTERVAL, is revised to specify that the refueling interval is the time between normal refuelings of the reactor, or at least once per 24 months. This change is needed to provide consistency between the Technical Specification definition of refueling interval and the 24 month plant cycles. This change results in extending all refueling interval based Technical Specification surveillances, which have been appropriately evaluated, from the current restriction of not to exceed 24 months without prior approval of the NRC to at least once per 24 months.

This revision to Definition 1.2.8 would allow the existing 25% surveillance extension to be periodically applied to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a fuel cycle length surveillance interval. The 25% allowable surveillance extension is not intended to be used repeatedly and is provided to facilitate surveillance scheduling. As stated above, all affected surveillance extensions are supported herein or are being revised to indicate the existing interval restriction until evaluations of these components/systems is completed. The 25% allowable extension is not applicable to intervals designated "F."

CHANGE NO. 2 - TABLE 1.2 - FREQUENCY NOTATION

Technical Specification Table 1.2 is revised to clarify Refueling Interval notation "R" as once per 24 months consistent with Technical Specification Definition 1.2.8 revision described above and the technical justifications for each individual surveillance extension contained herein.

Several Technical Specification Sections which currently specify surveillance requirements on a refueling interval basis have not been completely evaluated for extension. These Technical Specification surveillance intervals are revised to designate notation "F", which is being added to Table 1.2 and restricts the interval to the existing definition of a refueling interval. This change allows the Technical Specification definition of a refueling interval (Technical Specification Section 1.2.8) to be revised, thereby extending the interval only for these systems and components evaluated and addressed herein. The following is a listing of the Technical Specification surveillance intervals which remain on the existing refueling interval basis and are being revised accordingly:

1.	Table 4.1-1, Item 7	Reactor Coolant Temperature Channel Calibration
2.	Table 4.1-1, Item 10	Flux/Reactor Coolant Flow Comparator Calibration
3.	Table 4.1-1, Item 13	High Reactor Building Pressure Channel Calibration
4.	Table 4.1-1, Item 19	Reactor Building Emergency Cooling and Isolation System, Analog Channels a. Reactor Building 4 psig channel calibration d. Reactor Building 30 psig channel calibration e. Reactor Building Purge Line High Radiation calibration
5.	Table 4.1-1, Item 21.a	Reactor Building Spray System Analog Channel, Reactor Building 30 psig channel calibration
6.	Table 4.1-1, Item 25	Core Flooding Tanks a. Pressure Channel Calibration b. Level Channel Calibration
7.	Table 4.1-1, Item 27	Makeup Tank Level Channel Calibrations
8.	Table 4.1-1, Item 29	High and Low Pressure Injection Systems Flow Channels Calibration
9.	Table 4.1-1, Item 30	Borated Water Storage Tank Level Indicator Calibration

10. Table 4.1-1, Item 31	Boric Acid Mix Tank a. Level Channel Calibration b. Temperature Channel Calibration
11. Table 4.1-1, Item 32	Reclaimed Boric Acid Storage Tank a. Level Channel Calibration b. Temperature Channel Calibration
12. Table 4.1-1, Item 33	Containment Temperature Calibration
13. Table 4.1-1, Item 35	Emergency Plant Radiation Instrument Calibration
14. Table 4.1-1, Item 40	BWST/NaOH Differential Pressure Indicator Calibration
15. Table 4.1-1, Item 41	Sodium Hydroxide Tank Level Indicator Calibration
16. Table 4.1-1, Item 46	Turbine Trip/Reactor Trip Calibration
17. Table 4.1-1, Item 47.a	Pressurizer Code Safety Valve and PORV Tailpipe Flow Monitors Calibration
18. Table 4.1-1, Item 50	EFW Flow Instrumentation Calibration
19. Table 4.1-1, Item 51	Heat Sink Protection System a. EFW Auto Initiation Instrument Channels 1. Loss of Both Feedwater Pumps Calibration 3. Reactor Building Pressure Calibration
20. Table 4.1-1, Item 53	Chlorine Detection System Instrumentation Calibration
21. Table 4.1-1, Item 54	RCS Inventory Trending System a. Level Calibration b. Void Fraction Calibration
22. Table 4.1-2, Item 10(a)	Silt Accumulation Visual Inspection of Intake Pump House Floor
23. Table 4.1-4, Item 1.a-f	Noble Gas Effluent Radiation Monitor Calibrations
24. Table 4.1-4, Item 3	Containment Pressure Calibration
25. Table 4.1-4, Item 5	Containment Hydrogen Calibration
26. Table 4.1-4, Item 6	Wide Range Neutron Flux Calibration
27. Table 4.1-4, Item 11	Condensate Storage Tank Water Level Calibration

28. 4.4.4.1.b.1 Hydrogen Recombiner Instrumentation and Control Circuit Channel Calibration

The proposed change will revise the surveillance interval designations for each of the above items to maintain the current refueling interval definition restrictions. The proposed change is editorial in nature and has no effect on th. safety function of the subject systems and components.

CHANGE NO. 3 - PROTECTIVE INSTRUMENTATION

1. Technical Specification Table 4.1-1, Item 8 and 9, High/Low Reactor Coolant Pressure Channel Calibration is currently specified to be performed on a refueling interval basis. The Reactor Protection System (RPS) trips the reactor on high pressure to prevent over-pressurization of the reactor coolant system and to limit safety valve and PORV lift, and on low pressure to prevent DNBR limits from being challenged. The purpose of this surveillance is to verify the high/low pressure trip actuation setpoint. The proposed change will extend the interval between successive calibrations to at least once every 24 months. Evaluation of surveillance data from 1982 to 1990 indicates that variations in the data were within the loop error calculations performed in support of the RPS trip points. As-found setpoint drift data recorded in March 1987 was not considered valid for the evaluation since it is believed to be attributed to "zero-shift in the negative direction" caused by the instrument sitting at 0 psig for several weeks in the shutdown mode prior to the surveillance check.

   Generic Letter 91-04 Instrument Calibration Issues:

   Issue 1:

   Statistical analysis of the surveillance data established a 95/95% historical error figure of ± 1.2% which is well within the as-found tolerances/calculated loop error for the transmitter of ± 2.48%

   Issue 2:

   Review of the data from surveillance to surveillance indicates random variations as opposed to drift. However, for conservatism, the entire value is treated as "drift" in establishing a drift rate.

   Issue 3:

   The "drift" described in Issue 2 above was increased by the ratio 30/18 to conservatively account for a longer operating cycle resulting in a projected error figure of ± 2.0%. This value is well within the loop error analysis assumed drift value of ± 2.48%, which is the basis for RCS high and low pressure trip setpoints.

Issue 4/5:

Review of setpoints was not required as the calculated loop error is bounding.

Issue 6:

The allowable surveillance tolerances maintain instrument drift and errors within the analytical value of ± 2.48%. Therefore, the surveillance acceptance criteria satisfies the setpoint safety analysis assumptions.

Issue 7:

Repeated surveillance failures would be identified and corrective actions taken, if appropriate.

The proposed refueling interval change will have no effect on component availability since the instrumentation has demonstrated reliable operation and expected instrument drift over the extended calibration interval is acceptable. Additionally, Technical Specifications also require a shiftly channel check and a monthly channel test which will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of the High/Low Reactor Coolant Pressure channels.

2. Technical Specification Table 4.1-1, Item 12, Pump Flux Comparator calibration, and Item 51.a.2, Heat Sink Protection System - Emergency Feedwater Auto Initiation on Loss of All RC Pumps channel calibrations, are currently specified to be performed on a refueling interval basis. The purpose of this surveillance is to determine the reactor coolant pump (RCP) power level which will correlate to an RCP which is de-energized or unloaded. This instrumentation, which is part of the reactor protection system, trips the reactor if power measured by neutron flux is too high relative to the number and location of operating RCP's and also provides anticipatory initiation of emergency feedwater. The proposed change will extend the interval between successive calibrations to once every 24 months. Evaluation of surveillance data from 1982 to 1990 indicates that variations in the calibration check were random about the setpoint, and using that variation at the 95/95% level, the setpoint/error assumption has been satisfactorily met with no recalibrations required. Statistical analysis of the trip time delay also demonstrates that without recalibrations, the average of the results and the 95/95% variation is also within the specified requirement. In addition, Technical Specifications also require a shiftly channel check and a monthly channel test which provides additional assurance of component operability.

Generic Letter 91-04 Instrument Calibration Issues:

Issue 1:

Statistical analysis of surveillance calibration checks determined that at the 95/95% level, the surveillance results were consistent with As-Found tolerances. As-Found tolerances are the errors associated with operational experience (setpoint for RCP power level) and safety analysis related to 4RCP coastdown (trip time delay).

During the February, 1986 surveillance, three (3) of eight (8) monitors exceeded the trip time delay requirements. These were component failures. Peplacement of the failed components restored these instruments to compliance.

Issue 2:

Review of the data from surveillance to surveillance indicates random variations about an average value. Instrument drift per ISA definition is not identifiable. Re-calibrations were not performed - only calibration checks and replacement of failed components.

Issue 3:

A new error figure appropriate for 24 month cycles is not required, based on Item 2 above.

Issue 4/5:

As projected uncertainty in response error did not change because of the proposed 24 month cycle, relationships to safety analysis are not changed.

Issue 6:

The tolerances provided in the surveillance are more conservative than required As-Found tolerances.

Issue 7:

As these are Reactor Protection System related, associated with FSAR Chapter 14 safety analyses, tolerances related to acceptable As-Found tolerances are provided in the surveillance procedures.

The proposed refueling interval change will have no effect on component availability since this instrumentation has demonstrated reliable operation as cited above, and Technical Specification required shiftly channel checks and monthly tests will allow the operators to verify instrument channel performance. Therefore, the proposed change has no effect on the safety function of the Pump Flux Comparator instrument channel.

3. Technical Specification Table 4.1-1, Items 15.a and 17.a, Reactor Coolant Pressure Channel calibration for High Pressure Injection Analog Channels and Low Pressure Injection Analog Channels, respectively, are currently specified to be performed on a refueling interval basis. The analog channels consist of the following: 1) field transmitters located in the Reactor Building which are surveilled once every refueling outage, and 2) electronics located in the Control Building which are surveilled once every month. The purpose of the surveillance is to verify pressure transmitter (RC3A-PT3, RC3A-PT4 and RC3B-PT3) and associated electronics response corresponding to low reactor coolant system pressure for both High Pressure Injection (HPI) and Low Pressure Injection (LPI) initiation. The transmitters feed signal into respective electronics. An assessment has been performed of pressure transmitter responses to test signals for four surveillance intervals (1985 to 1990) to predict transmitter response at 30 months, and to quantify instrument drift and random uncertainty in transmitter response. A similar assessment of the electronics response to test signals equivalent to HPI/LPI for 17 monthly surveillances was also performed. A procedural surveillance tolerance limit for the transmitters and electronics is $\pm$0.5% and $\pm$0.4% FS respectively. Although, at times, the transmitters and electronics were found to be outside the surveillance acceptance criteria, statistical analysis of the surveillance data indicates that the transmitter and electronics response predicted at 30 months does not affect the protective function for HPI and LPI actuation. During three surveillances, RC3A-PT3 was out of calibration once, RC3A-PT4 was out of calibration once, and RC3-PT3 was out of calibration three times. All surveillance data points were included in the assessment of projected instrument response.

The low reactor coolant pressure setpoint for high and low pressure injection initiation is established based on a value such that protection is provided for the entire spectrum of break sizes and spurious initiation is avoided. The proposed change will extend the interval between successive transmitter calibrations to once every 24 months. Technical Specifications require setpoints of greater than or equal to 1600 psig and 500 psig for both HPI and LPI. The actual setpoints are consequently set at 1640 psig and 540 psig. An evaluation of historical calibration data using a linear regression model predicts daily drift rates as follows:

RC3A-PT3	=	2.7648 E-7 psi
RC3A-PT4	=	- 3.976 E-6 psi
RC3B-PT3	=	1.2971 E-5 psi

Accounting for drift and considering uncertainty associated with the linear regression model, as well as random uncertainty associated with the transmitter, the expected responses at 30 months are as follows:

RC3A-PT3	+ 1.16%	1659.24 psig	546.26 psig
	- 0.66%	1629.20 psig	536.44 psig
RC3A-PT4	+ 1.07%	1657.55 psig	545.78 psig
	- 1.12%	1621.62 psig	533.95 psig

RC3B-PT3	+ 3.30%	1694.12 psig	557.82 psig
	- 0.61%	1630.0 psig	536.71 psig

A similar expected response at 30 months for the electronics is as follows:

Channel (XMTR)	
RC1 (RC3A-PT3)	+0.22% -0%
RC2 (RC3A-PT4)	+0% -0.89%
RC3 (RC3B-PT3)	+1.44% -0%

A predicted error range encompassing all three transmitters projected to 30 months is -1.12% to 3.3% FS and that for three sets of electronics is -0.89% to 1.44% FS. The B&W analysis for the FSAR Chapter 14 accident analysis uses an HPI setpoint of 1480 psig while the actual plant setpoint is 1640 psig which affords a margin of 160 psig (1640-1480), or 6.4% of 2500 psig full scale. The negative instrument error will cause HPI actuation earlier than the 1640 psig setpoint. This is in the conservative direction, hence not a concern. The highest positive errors for electronics and transmitters are +1.44% and +3.3% in the same loop. These errors are in the nonconservative direction. The worst case positive error by summation method instead of the square root of the sum of the squares method would be +4.74% (1.44+3.3) which is still less than 6.4% margin. The same error by the square root of the sum of the squares method would be 3.6% which is acceptable. Thus, extension of the calibration interval to 30 months does not adversely impact nuclear safety.

Thus in all cases, transmitter response after 30 months will be at a pressure greater than that specified by Technical Specifications. Transmitter response within the range predicted at 30 months does not impact the protective function for HPI and LPI actuation. The FSAR Chapter 14 analyses assume protective function actuation prior to 1480 psig. This limit is not approached. HPI and LPI initiation at or before 500 psig has no basis in SAR accident analyses.

The proposed refueling interval change will have no effect on component availability since the instrumentation has demonstrated reliable operation and expected instrument drift over the extended calibration interval is acceptable. Additionally, Technical Specification required shiftly channel checks and monthly tests will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of the HPI and LPI Analog channels.

4. Technical Specification Table 4.1-1, Item 19.f, Line Break Isolation Signal channel calibration for Reactor Building Emergency Cooling and Isolation System Analog channels is currently specified to be performed on a refueling interval basis. The purpose of this surveillance is to verify level transmitter (NS-LT-800, 801; IC-LT-802, 803) setpoints. The ICCW and NSCCW Line Break Isolation Signal by itself does not actuate ESAS. A low level in the ICCW or NSCCW surge tank concurrent with an ES actuation will result in the isolation of the affected system from the primary containment. The proposed change will extend the interval between successive transmitter calibrations to once every 24 months. An evaluation of historical calibration data using a linear regression model predicts daily drift rates as follows:

IC-LT-802	=	-4.845 E-5 in.
IC-LT-803	=	-5.445 E-5 in.
NS-LT-800	=	2.5662 E-5 in.
NS-LT-801	=	2.8652 E-5 in.

Accounting for drift and considering uncertainty associated with the linear regression model, as well as random uncertainty associated with the transmitter, expected actuation setpoints at 30 months are as follows:

IC-LT-802	8 in.	
	+ 0.31%	8.0248 in.
	- 8.24%	7.341 in.
IC-LT-803	8 in.	
	+ 0.24%	8.0192 in.
	- 9.44%	7.245 in.
NS-LT-800	19.2 in.	
	+12.93%	21.68 in.
	- 9.95%	17.29 in.
NS-LT-801	19.2 in.	
	+12.94%	21.68 in.
	- 9.25%	17.42 in.

Delayed actuation of transmitters IC-LT-802 and 803 at 7.341 in. and 7.245 in., respectively, results in no adverse impact on ICCW operation. Premature actuation at the higher setpoint values results in delayed actuation of line break isolation; however, these values are within the existing tolerance of +0.50 in. Thus, extension of the calibration interval to 30 months does not adversely impact ICCW system operation or reactor safety. Premature or delayed actuation of NSCCW surge tank low level transmitters does not support a safety function. Therefore, this uncertainty band is acceptable. Additionally, Technical Specification required weekly channel checks and monthly tests will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of Line Break Isolation Signal (ICCW and NSCCW) Analog Channels.

5. Technical Specification Table 4.1-1, Items 22 and 26, Pressurizer Temperature and Level Channel Calibrations, respectively, are currently specified to be performed on a refueling interval basis. Pressurizer temperature transmitters provide input for pressurizer level temperature compensation. Pressurizer level transmitters provide input for pressurizer level temperature compensation, alarms, deenergizing the pressurizer electric heaters on low level, and for pressurizer level control by makeup and letdown. The purpose of this surveillance is to verify the actuation setpoints associated with these safety functions. The proposed change will extend the interval between successive calibrations to once every 24 months. Results of the evaluation of surveillance data from 1985 to 1990 for each component of the Pressurizer Temperature and Level Channels to predict drift and overall uncertainty at 30 months is tabulated below:

Total Drift and Instrument Loops	Allowable Surveillance Uncertainty at 30 months	Tolerance
NNI		
1. Pressurizer Temperature		
RC2-TTI	+ 1.4°F - 0.0105°F	± 12.6°F
2. Temperature Compensation		
RC1-LR	+ 6.55 in. - 7.26 in.	± 6 in.
RC2-TI	+ 8.75°F - 6.91°F	± 8.73°F
Non-NNI		
1. Pressurizer Temperature		
RC-TY-2C	+ 1.12°F - 9.4°F	± 7°F
2. Level Transmitter		
RC-LT-777	+ 0.158 ma - 0.100 ma	± 0.04 ma
RC-LY-777A	+ 0.114 v - 0.047 v	± 0.071 v
3. Temperature Compensation		
RC-LI-777	+ 3.53 in. - 3.76 in.	± 7.6 in.
LI-777A	+ 4.04 in. - 4.0 in.	± 7.6 in.

With respect to the NNI temperature loops, the total uncertainty, including drift, as predicted at 30 months is within the existing allowable tolerance and thus, a 30 month interval does not result in a change. For the NNI temperature compensation circuit the prediction for RC2-TI is +8.75F, which is an insignificant increase above the existing tolerance of ±8.73F. For RC1-LR, the prediction of +6.55 in., -7.26 in. exceeds the existing tolerance of ±6.0 in. but is insignificant in comparison with the margin available of +82.2 in., -17.68 in.

The NNI temperature compensated level circuit transmitters provide input for level recorder LR1, control for RCS makeup, and to the limit switch for high and low level alarms and the heater cutoff interlock. A potential premature alarm on low pressurizer level or a slight delay in heater cutoff has no impact on the protective function of the components.

For the non-NNI control room indication loop (LI-777A), extension to 30 months translates to a maximum calculated drift contribution of ±1.43%, or ±5.72 inches, which exceeds the existing tolerance but is within the margin available. For the remote shutdown panel indication (LI777), the maximum calculated drift contribution is ±1.44%, or ±5.75 inches which is within the margin available. No automatic controls are provided from the non-NNI loop.

The overall instrument uncertainty at 30 months, including the effect of drift, is sufficiently small that functions are assured. Additionally, Technical Specification required shiftly channel checks will allow operator verification of instrument channel performance. The criteria of Generic Letter 91-04 is not specifically addressed since this instrumentation is not related to any safety analysis assumptions.

Therefore, the proposed change has no effect on the safety function of the Pressurizer Temperature and Level Channels.

6. Technical Specification Table 4.1-1, Items 23 and 24, Control Rod Absolute Position (API) and Control Rod Relative Position (RPI) Channel calibrations, respectively, are currently specified to be performed on a refueling interval basis. These instrument channels provide two (2) separate position indication signals to the main control room. Indicator lights status each rod as: fully inserted, fully withdrawn, under control, and whether a fault is present. The purpose of this surveillance is to verify proper indication of equipment response. The proposed change will extend the interval between successive calibrations to once every 24 months. Results of the evaluation of surveillance data from the last five (5) calibration surveillances indicates only (1) rod indication out of tolerance in one (1) surveillance during API testing and only one (1) rod indication out of tolerance in (1) surveillance during RPI testing. API surveillances are performed on each of 69 rods at 0%, 50% and 100%. RPI surveillances are performed on each of 69 rods at 0% and 100%

An evaluation of the historical calibration data using a linear regression model, accounting for drift, and considering uncertainty associated with the linear regression model as well as random uncertainty associated with detector response, predicted instrument uncertainties at 30 months as follows:

RPI	0% Calibration	+1.718% -0.983%
	100% Calibration	+2.070% -1.622%
API	0% Calibration	+1.303% -0.780%
	50% Calibration	+0.687% -0.764%
	100% Calibration	+1.141% -0.972%
Group Average	0% Calibration	+1.730% -0.966%
	50% Calibration	+1.898% -0.839%
	100% Calibration	+1.584% -0.797%

The acceptable tolerance identified in the surveillance procedure is as follows:

RPI	±2.5%
API	±1.5%
Group Average	±2.0%

The criteria of Generic Letter 91-04 is not specifically addressed since these instrument indications are not related to any safety analysis assumptions. Since the uncertainties predicted at 30 months fall within the existing allowable tolerances, extension of the calibration interval to 30 months results in no change to instrument functional capability. Additionally, Technical Specification required shiftly channel checks will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of the API and RPI instrumentation.

7. Technical Specification Table 4.1-1, Item 37 and Table 4.1-4, Item 4 Reactor Building Sump Level and Containment Water Level Channel Calibrations, respectively, are currently specified to be performed on a refueling interval basis. Level transmitters (LT-804/LT-805) provide redundant control room alarm on high level, low-low level, and a low level interlock to close the reactor building sump drain valve to maintain a loop seal between the reactor building sump and the auxiliary building sump. Level transmitters (LT-806/LT-807) provide redundant level indication and a high level alarm in the control room for Regulatory Guide 1.97 post-accident monitoring requirements. The purpose of this surveillance is to verify acceptable setpoints for these functions. The proposed change will extend the interval between

successive calibrations to once every 24 months.

An evaluation of the historical calibration data using a linear regression model, accounting for drift, and considering uncertainty associated with the linear regression model as well as random uncertainty, the predicted instrument uncertainties at 30 months are as follows:

Indication	Error
LI-804	0.726%, -0.478%
LI-805	0.587%, -0.480%
LI-806	0.876%, -0.363%
LI-807	1.276%, -0.526%
Indication	
LYA-804	1.022%, -0.697%
LYA-805	1.374%, -0.742%
LYA-806	1.265%, -0.760%
LYA-807	1.937%, -0.724%
Alarm	
LYA-804/805	3.00%, -1.72%
LYA-806/807	1.77%, -2.36%

The calibration procedure specifies a tolerance of ± 2%. Since the instrument indication uncertainties predicted at 30 months fall within the existing allowable tolerances, extension of the calibration interval to 30 months results in no change to instrument functional capability. The predicted containment level alarm response of -2.3% is in a conservative direction (premature alarm). The sump level response of +3.0% would be a delayed alarm up to 2.7 inches above the 69 inch setpoint, or 71.7 inches. This still allows adequate warning for operators to drain the sump prior to overflow above 90 inches. Therefore, extension of the calibration interval to 30 months will not effect the alarm functions of these instrument loops.

Level Switches LS-804A/B had only one (1) surveillance data point available. The level switch predicted drift per the manufacturer is ±0.2% (3 sigma) for 12-18 months. This value was conservatively applied as a 12 month value, and extrapolation to 30 months results in a ±0.5% drift for the level switch. Converting to a 2 sigma value and using the square root of the sum of squares method the total loop uncertainty for LS-804A/B is +0.802%, -0.586%, which is within the existing allowable tolerance of ±2%. Therefore, extension of the calibration interval to 30 months results in no change to instrument functional capability.

Additionally, Technical Specification required weekly channel checks for Containment Water Level instrument loops will allow operator verification of instrument channel performance. Based on the above evaluations, it is concluded that the proposed change has no effect on the safety function of the Reactor Building Sump Level and Containment Water Level Channels.

The criteria of Generic Letter 91-04 are not specifically addressed since these instrument channel functions are not related to any safety analysis assumptions.

8. Technical Specification Table 4.1-1, Item 38, OTSG Full Range Level channel calibration is currently specified to be performed on a refueling interval basis. Full range level indication is utilized to meet the post-accident monitoring requirements of Regulatory Guide 1.97 and to provide level indication during normal heatups and cooldowns. The operator does not take action in a design basis event based upon full range level indication. Full range level indication does not interface with any control or protection system. The purpose of this surveillance is to monitor instrument loop accuracy. The proposed change will extend the interval between successive calibrations to once every 24 months.

   Results of the statistical evaluation of historical surveillance data from 1986 to 1991 determined that at the 95/95% level, the error at 30 months is larger than the current loop error calculations assume. However, there is no accuracy requirement for this indication for reasons as discussed above. Extension of the calibration interval to 30 months results in no change to instrument functional capability. Additionally, Technical Specification required weekly channel checks will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the function of the OTSG Full Range Level instrumentation channels.

   The criteria of Generic Letter 91-04 are not specifically addressed since these instrument channel functions are not related to any safety analysis assumptions.

9. Technical Specification Table 4.1-1, Item 39, Turbine Overspeed Trip channel test is currently specified to be performed on a refueling interval basis. The purpose of this surveillance is to test the mechanical trip device and electronic back-up trip device. The purpose of these trip devices is to prevent the turbine from acceleration to excessive overspeeds. The design trip setpoint of the mechanical trip device is 110% rated speed. The electronic overspeed trip device is a back-up overspeed trip to the mechanical device and the design trip setpoint is 112% rated speed. The proposed change will extend the interval between successive overspeed trip tests to once every 24 months.

   Evaluation of the three (3) surveillance results conducted since Cycle 5 startup, October 1985 was performed. At each data point, the turbine received two (2) actual overspeed tests to check the performance of the mechanical trip device and backup electronic trip device. The tests were performed with the rotor hot and the turbine-generator off line. The criteria for mechanical trip points is 1,980 rpm to 1,998 rpm (110% to 111%). The actual measured trip point for the three (3) data points varied from 1,950 rpm to 1,960 rpm. TMI-1 has elected to continue operating at this lower setpoint without re-adjusting. This is a conservative decision which means that the unit will likely trip on lower overspeed if full load is lost.

The criteria for electronic backup overspeed trip is 2,016 +0, -10 rpm (112%). The actual measured trip point for the three (3) data points varied from 2,006 rpm to 2,015 rpm which is within specification.

Therefore, since restart (Cycle 5) which is a duration of approximately 56 months to February 1990, all trip tests have been repeatable and reliable with no drift or change to the setpoint outside an acceptable range. Further, all steam valves tripped shut as required demonstrating high component reliability.

These test results provide conclusive evidence that the trip devices are reliable and can function for 24 months at a time without requiring maintenance or without decreasing their availability to function. In addition, to Technical Specification required surveillance testing at the refueling intervals, simulated circuit tests and routine functional tests of the mechanical and electrical devices are performed monthly while on line to verify operability. These tests consist of locking out the trip circuit and actuating the mechanical and backup overspeed trip devices at 1,800 rpm to verify their operability. These routine tests have always been successful, providing further evidence that the trip systems do not deteriorate with time. Therefore, the proposed change has no effect on the functional capability of these trip devices.

The criteria of Generic Letter 91-04 is not specifically addressed since these trip setpoints are not related to any safety analysis. The potential for generation of a turbine missile due to failure of the overspeed trip devices is bounded by the design basis aircraft impact scenario.

10. Technical Specification Table 4.1-1, Item 42, Diesel Generator Protective Relaying calibration, is currently specified to be performed on a refueling interval basis. These relays ensure the availability of the diesel generator during a sustained degraded grid or loss of offsite power by protecting the diesel generator against faults and initiates alarms on abnormal generator conditions. These relays also ensure that the diesel generator is synchronized, or that voltage and frequency are within allowables, prior to breaker closure. The purpose of this surveillance is to verify relay setpoint value associated with these functions. The proposed change will extend the interval between successive calibrations to once every 24 months. Evaluation of historical surveillance data is described for each relay as follows:

    a. Synchronism (Sync) Check Relay

    Historical data does not exist for the sync relays since these relays were installed with ABB Type ITE-25S in February 1990 and the January 1991 maintenance record showed that the relay settings have been changed. However, calibrating the ABB Type ITE-25S relays for a 24 month cycle will have no adverse impact on its safety function because the relays are not expected to drift ±1% of the set point. According to the manufacturer's specification, the maximum tolerance on repeatability for the voltage difference pick up and time delay setpoint of the relay is ±1%. Surveillance Procedure acceptance criteria limits the drift of the sync relays to ±8.3% for the voltage difference pick up setpoint and ±10% for the time delay setpoint.

b. Reverse Power Relay

Extending the surveillance of the reverse power relay has no adverse effect on its safety function because the maintenance records since 1984 show that during 5 out of 6 surveillance tests the components of the relay such as the timer and directional units stayed within the acceptable tolerance specified in the surveillance procedure. During this one occasion that the relay was out of tolerance, the safety function of the relay was not affected. The directional unit picked up at 1.88 va which is ahead of the minimum allowable setting of 2.28 va. Also, these maintenance records of TMI-1 diesel generators show that the diesel generator did not incorrectly trip on the reverse power relay.

c. Loss of Field Relay

Extending the surveillance of the loss of field relay has no adverse effect on its safety function because the maintenance records since 1984 show that during 4 out of 6 surveillance tests the components of the relay such as the impedance trip unit, directional unit, and telephone unit (X) stayed within their acceptable tolerance specified in the surveillance procedure. During the two (2) occasions that the relay went out of tolerance, the safety function of the relay was not affected. Past records of TMI-1 emergency diesel generators showed that the diesel generator did not inadvertently trip on the loss of field relay.

On these two (2) occasions that the relay went out of tolerance only the telephone unit (X) was off calibration. At one time the unit dropped out at 28 cycles; and, at the other time the unit dropped out at 8 cycles. The telephone unit is calibrated to provide 12 to 18 cycles (0.2 to 0.3 seconds) delay on the tripping circuit of the relay in order to prevent inadvertent tripping during recoverable swings and voltage transients. The safety function of the telephone unit is to provide continuity on the relay's trip circuit to allow the relay to perform its safety function before the diesel generator sustains damage due to overheating. It may be conservatively stated that synchronous machines can usually tolerate 5 seconds (300 cycles) rotor and station heating due to an open field and 10 seconds (600 cycles) of heating due to a shorted field. On both occasions, continuity on the trip circuit was provided before 5 seconds.

d. Negative Sequence Relay

Extending the surveillance of the negative sequence relay has no adverse effect on its safety function because the maintenance records since 1984 show that the relay stayed within the acceptable tolerance specified in the surveillance procedure.

e. Thermal Overload Relay, Field Ground Relay

These relays do not affect the safety function of the emergency diesel generator. These relays only initiate an alarm when the diesel generator is overloaded or during a ground fault on the generator field. Therefore, extending the relay surveillance/ calibration test has no adverse effect on the performance of the emergency diesel generators.

f. Up-to-Voltage Relays

The existing Westinghouse Type CV7 voltage relays are being replaced with ABB Type ITE-59N relays during the Diesel Generator annual overhaul. According to the manufacturer's specification, the expected tolerance of the ITE-59N relays for the pick-up and dropout setting is ±0.3%; i.e., ±0.1% tolerance for repeatability over "allowable" dc control power range and ±0.2% tolerance for repeatability over a temperature range of 0 to 40°C.

A 24 month calibration cycle for the ITE-59N relays will have no adverse impact on its safety function because of its expected ±0.3% drift. The surveillance procedure provides a ±3% tolerance for the relay pick-up setting.

g. Over Voltage Relay, Neutral Ground Relay, Field Overload Relay

Extending the surveillance of the over voltage and neutral ground relays from 18 months to 24 months has no adverse effect on its safety function because the maintenance records since 1984 show that these relays stayed within the acceptable tolerance specified in the surveillance procedure.

h. Up-to-Frequency Relay

These relays were replaced with ABB Type ITE-81 relays in November/ December 1989. The October 1990 maintenance record for these relays shows that the as-found calibration data is within the acceptable tolerance specified in the surveillance procedure. This excellent performance of the relays for a 10 month duration can be expected for a 24 month calibration cycle because the relays are not expected to drift ±0.008 hertz from its trip set point. According to the manufacturer's specification, the maximum trip point accuracy/repeatability at -20 to +55°C is ±0.008 hertz. Surveillance procedure acceptance criteria limits the drift of the up-to-frequency relays to ±0.3 hertz.

i. 30% Differential Relay

Extending the surveillance of the differential relay from 18 months to 24 months has no adverse effect on its safety function because the maintenance records since 1985 show that the relay stayed within the acceptable tolerance specified in the surveillance procedure.

The proposed refueling interval change will have no effect on component availability since the relays which perform a safety function have demonstrated reliable operation and expected drift over the extended calibration interval is acceptable. Therefore, the proposed change has no effect on the safety function of the Diesel Generator Protective Relay channels.

The criteria of Generic Letter 91-04 is not specifically addressed since these protective relay setpoints are not related to any safety analysis assumptions.

11. Technical Specification Table 4.1-1, Item 43, 4KV ES Bus Undervoltage Relay calibrations are currently specified to be performed on a refueling interval basis. Undervoltage protection is provided by the degraded grid and loss of voltage relays on the 4KV ES buses, which start the emergency diesel generators thus protecting the safety related equipment from loss of power during a sustained grid condition or loss of offsite power. The purpose of this surveillance is to verify relay setpoint values associated with these functions. The proposed changes will extend the interval between successive calibrations to once every 24 months.

    Evaluation of historical surveillance data is described for each relay as follows:

    a. Degraded Grid Relays

    The history from 1985 to 1990 for the degraded grid relays (ITE-27N) was reviewed. The "as found" pick-up and drop-out setting during this period remained within the acceptance criteria tolerance specified in the surveillance procedure.

    The worst case deviation of the pickup and dropout setting of the ITE-27N relays for two surveillance cycles (approximately 36 month interval) without calibration is 0.2 volts (0.33%) and 0.1 volt (0.17%), respectively. The maximum relay setting tolerance according to the manufacturer's specification is $\pm$0.3%; i.e., $\pm$0.1% tolerance for repeatability over "allowable" dc control power range plus $\pm$0.2% tolerance for repeatability over a temperature range of 0 to 40°C. The surveillance procedure acceptance criteria envelop the maximum tolerance specified by the relay manufacturer for repeatability.

    The "as found" setting of the AGASTAT 7012 relays (27XCTD and 27EXTD) associated with the degraded grid relays are within the acceptance criteria tolerance specified in the surveillance procedures.

    b. Loss of Voltage Relays

    The history from 1984 to 1990 for the loss of voltage relays (ITE-27H) was reviewed. The "as-found" pickup and drop out setting during this period stayed within the acceptable criteria tolerance specified in the surveillance procedure.

The worst case deviation of the pick-up and drop-out setting of the ITE-27H relays for two (2) surveillance cycles (approximately 36 month interval) without calibration is 0.7 volts and 0.5 volts respectively. Typical relay setting tolerance according to the manufacturer's specification is ±1.1 volts; i.e., ±0.6 volts repeatability for a 30 volt variation in control voltage and ±0.5 volt repeatability over 20-40°C. The terminal voltage on the battery varies from 105 to 136 volts.

The "as found" setting of the AGASTAT 7012 relays (27XDTD, 27XFTD, 27XGTD, AND 27XHTD) associated with the loss of voltage relays have always been within the acceptance criteria tolerance specified in the surveillance procedure.

The proposed refueling interval change will have no effect on component availability since the instrumentation has demonstrated reliable operation and expected drift over the extended calibration interval is acceptable. The degraded voltage logic and loss of voltage logic is designed for 2 out of 3 relay operation in series with the respective time delay relays to provide a high degree of instrument reliability. Additionally, Technical Specification required monthly operability tests will allow operator verification of relay performance. Therefore, the proposed change has no effect on the safety function of the 4KV ES Bus Undervoltage Relays.

The criteria of Generic Letter 91-04 is not specifically addressed since these protective relay setpoints are not related to any safety analysis assumptions.

12. Technical Specification Table 4.1-1, Item 44, Reactor Coolant Pressure Decay Heat Valve Interlock Bistable channel calibration, is currently specified to be performed on a refueling interval basis. The Decay Heat valve interlocks preclude valve opening at pressures greater than 400 psig to prevent ever-pressurization of the Decay Heat Removal System. The purpose of this surveillance is to verify interlock setpoint values. Technical Specifications require calibration on a refueling cycle basis, and testing on a monthly basis, as well as shiftly channel checks. TMI-1 Procedure 1303-4.19 requires that the interlock bistables be bypassed on a monthly basis and calibrated. TMI-1 Procedure 1302-5.8 requires that HPI and LPI analog channels for RCS pressure be calibrated once per refueling cycle with the Decay Heat Removal interlock bistables bypassed, and following these HPI and LPI Channel calibrations, the bistables are reset.

    Since calibration is in fact performed as part of the monthly test, extension of the Technical Speciication calibration interval to 24 months has no impact on the functional capability. Therefore, the proposed change has no effect on the functional capability of the Reactor Coolant Pressure Decay Heat Valve Interlock Bistable Channels.

13. Technical Specification Table 4.1-1, Item 45, Loss of Feedwater Reactor Trip Channel calibration, is currently specified to be performed on a refueling interval basis. These switches provide a faster reactor trip response to a loss of feedwater transient than can be provided by the high reactor coolant pressure trip. This anticipatory reactor trip on loss of feedwater pumps minimizes the number of challenges to the PORV by reducing the number of transient responses resulting in high reactor coolant pressure. The purpose of this surveillance is to verify switch setpoint response. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of historical surveillance data using a linear regression model predicts a drift for the pressure switches monitoring main feedwater pump control oil pressure of -0.3 psi for a 30 month period. Accounting for drift, and considering uncertainty associated with the linear regression model as well as random uncertainty, the predicted uncertainty in the 75 psig setpoint at 30 months is +1.89 psi, -4.18 psi. Surveillance procedure acceptance criteria specifies a tolerance of $\pm$2% of span, which is equivalent to $\pm$ 4.4 psi. Thus, the setpoint uncertainty at 30 months is within the tolerance criteria. Additionally, Technical Specification required shiftly channel checks and monthly channel tests will allow operator verification of channel performance. Therefore, the proposed change has no effect on the safety function of the Loss of Feedwater Reactor Trip instrument channel.

    This anticipatory trip feature is provided to minimize challenges to the PORV over plant lifetime, and is not credited in any safety analysis assumptions. Therefore, the criteria of Generic Letter 91-04 is not specifically addressed.

14. Technical Specification Table 4.1-1, Item 47.b, PORV-Acoustic Flow Channel calibration, is currently specified to be performed on a refueling interval basis. This instrument monitors the vibration level at the PORV discharge pipe and provides indication of the vibration level. High vibration, indicative of flow through an open PORV, results in a local alarm and a control room alarm. The purpose of the surveillance is to verify proper acoustic monitor response. The proposed change will extend the interval between successive calibrations to once every 24 months.

    An analysis of historical calibration data using a linear regression model predicts daily drift rates and 30 month uncertainties for parameters of the PORV acoustic monitoring system. Evaluation of the results of the analysis has established that the system will perform its intended function at the 30 month surveillance interval considering the effects of drift and random variability. A summary of these results follows along with a comparison to the existing tolerance specified in the surveillance procedure. It was also noted that no component failures occurred during the period covered by the 5 surveillances.

    The acoustic monitor amplifier has a 30 month uncertainty of +1.047%, -0.847%. The existing tolerance is ±1%. This parameter is just outside the existing tolerance in a conservative direction (more likely to cause an alarm). Therefore, a calibration interval of 30 months does not adversely impact its intended function. The panel meter indication has a 30 month uncertainty of +1.108%, -1.762%. Since this is within the existing tolerance of ±3%, a calibration interval of 30 months does not adversely impact the functional capability. The High-Pass and Low-Pass filter projected performance at 30 months is adequate, since given the worst expected drift there will always be a band between 4.34 KHZ and 6.44 KHZ that is unattentuated. The expected uncertainty in valve monitor sensitivity at 30 months is +5.13%, -4.07%. Given a 9g alarm setpoint, the alarm setpoint uncertainty would be +0.462g, -0.366g. This is acceptable, since PORV lift tests have resulted in 40g readings at lower pressures. Therefore, the system would still perform its function. Additionally, Technical Specification required monthly tests will allow operator verification of channel performance. Therefore, the proposed change has no effect on the safety function of the PORV Acoustic Flow Monitor.

    The PORV Acoustic Flow Monitor is not credited in any safety analysis assumptions. Therefore, criteria of Generic Letter 91-04 is not specifically addressed.

15. Technical Specification Table 4.1-1, Item 48, PORV Setpoints channel calibration, is currently specified to be performed on a refueling interval basis. The PORV setpoints are specified with tolerances assumed in the bases for technical specification pressurization, heatup and cooldown limitations. The purpose of this surveillance is to verify proper setpoint values. The proposed change will extend the interval between successive calibrations to once every 24 months. Technical Specifications currently require monthly channel tests of PORV setpoints. These monthly tests require setpoint actuations in

response to input voltage signals for both high and low PORV setpoint and temperature interlocks values. The setpoints are calibrated and retested based on these monthly test results, if required. The only additional verification performed on a refueling cycle basis beyond that on a monthly basis is valve operability (open or closed). Thus, extending the refueling cycle interval has no impact on the PORV setpoint verification.

16. Technical Specification Table 4.1-1, Item 49, currently specifies calibration for the Saturation Margin Monitor to be performed on a refueling interval basis. The Saturation Margin Monitor does not perform any automatic functions, but informs the operator of the margin between the existing reactor coolant system temperature and the temperature at which the reactor coolant would saturate to steam. The reactor coolant saturation temperature is determined as a function of reactor coolant pressure as provided by inputs from pressure transmitters PT-949 and PT-963. The temperature input is provided by hot leg water temperature channels TE-958 and TE-960. The purpose of this surveillance is to verify proper setpoint values. The proposed change will extend the interval between successive calibrations to once every 24 months. Analyses of the Saturation Margin Monitor System surveillance data have been performed using linear regression techniques to predict daily drift rate of the instruments in this system, as well as to predict total uncertainty in instrument response at 30 months (24 months $\pm$ 25%).

    In the pressure transmitter loops, only the pressure transmitters themselves require analysis, because these instruments are located within the Reactor Building and are inaccessible for calibration during the operating cycle. For the temperature input to the Saturation Margin Monitor, only the temperature elements (TE-958 and TE-960) are within the Reactor Building. Because these components are passive devices, they do not require calibration and are acceptable for use in the extended operating cycle. The remaining components which are located in a mild environment outside of the Reactor Building (in the Control Building), are checked monthly via calibration procedures.

    A statistical evaluation of pressure transmitter historical calibration data was performed for PT949/PT963 using linear regression to predict drift and variability of response. Accounting for drift, and considering uncertainties associated with the linear regression model as well as uncertainty associated with transmitters due to random variation, the total uncertainty at 30 months is predicted to be + 0.17%, - 0.06% of span. This is within the existing allowable tolerance of $\pm$0.25%, and as such does not represent a change. On this basis, extension of the surveillance interval to 30 months for the Saturation Margin Monitor is justified. Additionally, Technical Specification required shiftly channel checks and monthly channel tests will allow operator verification of channel performance. Therefore, the proposed change has no effect on the safety function of the Saturation Margin Monitor System.

17. Technical Specification Table 4.1-1, Item 51.a.4, Heat Sink Protection System (HSPS) - Emergency Feedwater (EFW) Auto Initiation on OTSG Low Level instrument channel calibration, is currently specified to be performed on a refueling interval basis. Surveillance of the low level transmitters and a multiple point calibration check of the HSPS electronics is specified by Technical Specification Table 4.1-1, Item 51.c.1. Injection of test signals into the HSPS and testing of trip points and outputs to coincidence logic, and testing of the coincidence logic and outputs up to and including valve actuation and pump start are performed on a quarterly basis as specified by Technical Specification Table 4.1-1 Items 51 a,b,c and d. The complete function of EFW low level initiation is either performed on a quarterly basis or is addressed in Technical Specification Table 4.1-1, Item 51.c.1 below. Therefore, the proposed change has no effect on the safety function of the HSPS - EFW Auto initiation on OTSG Low Level instrumentation channel.

18. Technical Specification Table 4.1-1. item 51.b, HSPS-MFW Isolation OTSG Low Pressure instrument calibration, is currently specified to be performed on a refueling interval basis. The purpose of this surveillance is to verify proper MFW valve closure setpoint. The safety function of this signal input is to prevent additional energy being dumped to the containment in a steam line break. The proposed change will extend the interval between surveillance for certain portions of the HSPS MFW isolation on low OTSG pressure to once every 24 months. Transmitter calibration is addressed in Change No. 5, Item 4 below (Table 4.1-4, Item 10 Steam Generator Pressure Transmitters). Injection of test signals to HSPS setpoint checks up to input of logic matrices, and testing of each logic matrix are performed on a quarterly basis by Technical Specification

    Table 4.1-1, Item 51.b. Testing of the final "and" logic function and MFW valve closure is performed on a refueling basis. This surveillance procedure does not address variable signals or setpoints, but is on-off in nature. Review of the two refueling cycle surveillance data results conducted since the instrumentation was installed indicated no deficiencies or failures to function. No variation in performance was experienced. Therefore, the proposed change has no effect on the safety function of the on-off MFW isolation on low OTSG pressure surveillance. The issues of Generic Letter 91-04 are not specifically addressed as no drift or accuracy figures are involved.

19. Technical Specification Table 4.1-1, Item 51.c.1, HSPS - EFW Control Valve Control System, OTSG Level Loops instrument calibration, is currently specified to be performed on a refueling interval basis. The purpose of this surveillance is to verify proper setpoint values for the level control function associated with EFW control valve level control. The HSPS utilizes the OTSG levels as the control signals for manipulating EFW control valves. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of the existing HSPS electronics surveillance results from the last five surveillance periods calculated a statistical representation of the error, conservatively increased it to account for the longer interval and determined at a 95/95% confidence level an

error of ± 1.22% for the startup range instruments and -14.4%, +4.8% for the operating range instruments. This is within the assumed accuracy of ± 1.3% (startup range) and -20%, +10% (operating range) for the electronics portion of the overall loop accuracy. The derived error conservatively assumes that the entire figure is "drift" versus "inherent accuracy and drift".

The startup range level transmitters were replaced in 9R with Rosemount Model 1154's which have superior accuracy. The manufacturer's projected drift/accuracy for 30 months is within allowable tolerances.

The proposed refueling interval change will have no effect on component availability since the instrumentation has demonstrated reliable operation and expected instrument drift over the extended calibration interval is acceptable. Additionally, Technical Specification required weekly channel checks and quarterly channel test will allow operator verification of channel performance. Therefore, the proposed change has no effect on the safety function of the HSPS-EFW control valve control system, OTSG level instrumentation.

Generic Letter 91-04 Instrument Calibration Issues:

Issue 1: Statistical analysis of multi-point calibration surveillance data established 95/95% error figures which are well within calculated as-found tolerances.

Issue 2: The above figure is treated as a drift figure, although this string of electronic modules exhibits a result which is more akin to random variation about zero error, as the loop error analyses assume.

Issue 3: The "drift" described in Issue 2 was increased by the ratio 30/18 to conservatively account for a longer operating cycle resulting in a 95/95% figures which are well within the assumed drift values.

Issue 4: It was not necessary to revise the associated loop error analysis as the results from Issue 3 above were still within the calculated as-found tolerance for the electronics. The error analysis was revised to reflect the new figures for the start-up range transmitter.

Issue 5: It was necessary to review accuracy requirements only for operating range level control. These requirements are still met.

Issue 6: Surveillance procedure as-left tolerances are consistent with errors and assumptions of the safety and setpoint analyses.

Issue 7: Repeated surveillance failure would be identified and corrective actions taken, if appropriate.

20. Technical Specification Table 4.1-1, Item 51.c.2, HSPS - EFW Control Valve Control System Controllers calibration, is currently specified to be performed on a refueling interval basis. The safety function of the HSPS level controllers is to modulate the control valve in order to maintain the OTSG level signal at setpoint. The purpose of this surveillance is to verify proper setpoint values. The proposed change will extend the interval between successive calibration to once every 24 months. A statistical evaluation of the data from the two refueling interval surveillances conducted to date indicates with a 95/95% confidence level a controller error at 24 months of -17.3 inches, + 16.3 inches for the startup range which is well within the level control error limits of ± 25 inches; and -12.9%, + 3.28% for the operating range which is also well within the level control error limits of -15%, + 10%.

Generic Letter 91-04 Instrument Calibration Issues:

Issue 1: Portions of the overall level control function were not addressed in existing calculation or used to determine drift for the overall loop from as-found tolerances of individual components. However, surveillance data was available for the components in the loop, and an overall controller error was determined as described above.

Issue 2: Surveillance results were analyzed to determine a statistical 95/95% value.

Issue 3: The statistical errors obtained were increased by the ratio 30/18 to account for a 24 month operating cycle. This is an appropriate approach as surveillances determine "inherent accuracy + drift". If vendor specified accuracy was larger, it was utilized.

Issue 4/5: The error values obtained above were combined with the loop error for the remainder of the over-all control function. These new resulting loop errors, more inclusive than existing analysis, were compared to the same specified accuracy requirements in the existing analysis. These requirements were still maintained and thus the setpoints are still adequate.

Issue 6: As these setpoints do not relate to FSAR Chapter 14 analysis, the procedures do not contain As-Found tolerances. Tolerances provided in the surveillance procedures are vendor specified accuracy.

Issue 7: Patterns of failure or significant recalibrations are addressed on a case-by-case basis.

The proposed refueling interval change will have no effect on component availability since the instrumentation has demonstrated reliable operation and expected instrument drift over the extended calibration interval is acceptable. Additionally, Technical

Specification required weekly channel checks will allow operator verification of channel performance. Therefore, the proposed change has no effect on the safety function of the HSPS-EFW Control Valve Control System Controllers.

21. Technical Specification Table 4.1-1, Item 51.d, HSPS Train Actuation Logic calibration, is currently specified to be performed on a refueling interval basis. The EFW initiation portion of HSPS Train Actuation logic testing is performed in accordance with Technical Specification Table 4.1-1, Item 51. This is a functional on-off test which actuates pumps and valves. It is performed quarterly and therefore is not affected by a longer operating cycle.

    The MFW isolation portion of HSPS train actuation logic testing is performed in accordance with Technical Specification Table 4.1-1, Item 51. This is a functional on-off test which tests everything up to but not including closing MFW valves. It is performed quarterly and therefore is not affected by a longer operating cycle. Actual closing of MFW valves is addressed in Item 19 above, which concluded that extension of the operating cycle to 24 months does not impact the safety function of this part of the actuation logic.

    Therefore, the proposed change has no effect on the safety function of the HSPS Train Actuation Logic.

22. Technical Specification Table 4.1-1, Item 52, Backup Incore Thermocouple Display calibration, is currently specified to be performed on a refueling interval basis. The Backup Incore Readout (BIRO) System is a diverse readout system, redundant to the computer, for monitoring core exit temperature via thermocouples. The BIRO is used as a source for determining margin to saturation in the event of natural circulation cooling. The purpose of this surveillance is to verify proper thermocouple and RTD output converter indications. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of historical data from 1986 to 1990, using a linear regression model, accounting for drift and considering uncertainty associated with the linear regression model as well as random uncertainty, predicted uncertainties at 30 months as listed below:

Signal Converters (R/E)	Uncertainty at 30 Months	Tolerance
TY-952A	+0.02 VDC,-0.02 VDC	±0.05 VDC
TY-953A	+0.03 VDC,-0.005 VDC	±0.05 VDC
TY-954A	+0.02 VDC,-0.002 VDC	±0.05 VDC
TY-955A	+0.03 VDC,-0.009 VDC	±0.05 VDC
RTD Comparison Checks	+2.89F, -0.997F	±4F
Control Room Indication - TI-952		
Penetration Temp. 32-100F	+3.53F, -5.09F	±8F
Penetration Temp. 200-400F	+4.05F, -3.75F	±8F

Thermocouples - Simulated Input

100F	+4.65F, -0.26F	±8F
380F	+3.30F, -1.75F	±8F
540F	+3.10F, -1.25F	±8F
980F	+2.54F, -2.11F	±8F
1420F	+1.65F, -3.71F	±8F
1860F	+0.08F, -0.83F	±8F
2300F	+0.63F, -0.63F	±8F

In all cases, the predicted uncertainty at 30 months is within the allowable tolerance criteria. Additionally, Technical Specification required monthly channel checks will allow operator verification of channel performance. Therefore, the proposed change has no effect on the safety function of the Backup Incore Thermocouple Display instrumentation.

The BIRO System is not credited in any safety analysis assumptions. Therefore, the criteria of Generic Letter 91-04 is not specifically addressed.

CHANGE NO. 4 - EQUIPMENT TESTS

1. Technical Specification Table 4.1-2, Item 1, Control Rods, and Technical Specification Section 4.7.1.1, Control Rod Drive System Functional Tests, currently specify control rod drop time testing to be performed on a refueling interval basis prior to return to power. The control rod drive mechanisms provide for controlled withdrawal or insertion of the control rod assemblies out of or into the core and are capable of rapid insertion or trip. There are a total of sixty-nine (69) control rod drive mechanisms in the system, which are divided into eight (8) groups. The purpose of this surveillance is to verify that the safety and regulating rod drop times are below the assumed rod drop times in the analysis and that the APSRs do not drop on a trip. The proposed change will extend the interval between successive tests to once every 24 months. Test data recorded from actual tests performed during previous refueling outages indicates no significant trend of system degradation with respect to control rod trip insertion time. The available test data indicated a very slight upward trend in rod drop time which can be attributed to a number of factors, such as, equipment accuracies, different reactor coolant flow conditions, and normal wear and tear of all associated equipment. No effect has been identified that relates the small drop time increases to increasing cycle length. The test data all exhibit large margins to the drop time criteria. Future drop test times will be monitored to confirm that margin is maintained. The following tabulation provides a summary of reactor operating times and control rod drop time tests data since plant restart in 1985.

Outage	Test Interval (Months)	Operating Cycle (Months)	Cycle Length (EFPD)	Avg. Rod Drop Time (Seconds)	Max Rod Drop Time* (Seconds)
5R		13	290	1.245	1.280
6R	22	15	425	1.252	1.280
7R	17	16.5	460	1.254	1.292
8R	18	17.5	520	1.257	1.300

*Acceptance criteria - Drop Time ≤ 1.66 seconds at hot - full flow, and ≤1.40 seconds at hot - no flow.

Only two (2) individual electrical switch failures were identified in the above testing and were related to single control rods within group #5. Review of axial power shaping rod loss of power testing indicates no deficiencies. The Babcock & Wilcox Fuel Company Control Rod Assembly is designed for long-term, multi-cycle use. Mechanical design life is limited by total accumulated neutron fluence and is not affected by specific cycle lengths.

The proposed refueling interval change will have no effect on operability of the Control Rod Drive System since functional tests cited above have demonstrated reliable operation, and no evidence of significant system performance degradation over service time has been identified. The two electrical switch failures are considered isolated occurrences. Additional Technical Specification required control rod movement testing of each rod every two weeks when the reactor is critical provides added assurance of component availability. Therefore, the proposed change has no effect on the safety function of the Control Rod Drive System.

2. Technical Specification Table 4.1-2, Item 3, Pressurizer Safety Valves, currently requires setpoint testing of 50% of the valves each refueling period. The pressurizer code safety valves protect the reactor coolant system against overpressure. The purpose of this surveillance is to exercise the valves to assure service readiness for operation. Technical Specifications require both pressurizer code safety valves to be operable with a lift setting of 2500 psig ±1%. The proposed change will extend the interval between successive tests to once every 24 months. Evaluation of the last six (6) surveillance data points from 1985 to 1990 indicates one valve tested satisfactorily, three valves tested high, and two valves tested low. Of primary importance are the instance where the valve lifted above 2525 psig. The highest out-of-tolerance lift point was 2579 psig which is 2.1% out-of-tolerance. In no instance did the tested valve fail to operate or give evidence that it was incapable of providing over pressure protection in accordance with the ASME B&PV Code criteria of 110% of design pressure (2750 psig safety limit). Valve service time ranged from 9 months to 36 months so there is no evidence that setpoint drift is a function of time in service. The FSAR Chapter 14 accident analyses which take credit for the pressurizer safety valves are the Startup Accident and Main Feedwater Line Beak Accident. Review of these analyses indicate that a 3-4% higher lift

pressure above the setpoint valve of 2500 psig would not affect the analysis results. The proposed refueling interval change will have no effect on the operability of the pressurizer safety valves since functional tests cited above have demonstrated that the valves remain capable of protecting the RCS against overpressure and that there is no impact on accident analysis results due to setpoint errors slightly above the current tolerance criteria. Therefore, the proposed change has no effect on the safety function of the Pressurizer Safety Valves.

3. Technical specification Table 4.1-2, Item 4, Main Steam Safety Valves (MSSV), currently requires setpoint testing of 25% of the valves each refueling period. The main steam safety valves protect the OTSG's and the main steam system from overpressurization. The purpose of the surveillance is to exercise the valves to assure service readiness for operation. The proposed change will extend the interval between successive tests to once every 24 months, and increases the sampling percentage from 25% to approximately 50% to be consistent with ASME Section XI requirement to test each MSSV once during a five (5) year period. Review of the last five (5) surveillance data points from 1985 to 1989 has shown a consistent history of setpoint drift. The drift is consistently low with few exceptions (11 data points of 48 individual valve tests) drifting high, thus, the setpoint out of tolerance is usually conservative. No valve has failed to function or been incapable of providing overpressure protection in accordance with the ASME B&PV Code. The valves have always functioned properly and fully opened without exceeding the ASME B&PV Code set pressure limitation of 110% of system design pressure. There is no evidence that drift is a function of time in service. The proposed refueling interval change will have no effect on the operability of the MSSV's since functional tests cited above have demonstrated that the valves remain capable of providing overpressurization protection for the OTSGs and Main Steam System. Therefore, the proposed change has no effect on the safety function of the Main Steam Safety Valves.

CHANGE NO. 5 POST-ACCIDENT MONITORING INSTRUMENTATION

1. Technical Specification Table 4.1-4, Item 2, Containment High Range Radiation instrumentation calibration is currently specified to be performed on a refueling interval basis. These radiation monitors provide high alarm setpoints and post-accident monitoring. The purpose of this surveillance is to verify proper detector response and alarm setpoints. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of historical data from 1985 to 1990 using a linear regression model, accounting for drift, and considering uncertainty associated with the linear regression model as well as random uncertainty associated with detector response, results in a predicted detector response at 30 months with a 95% confidence as follows:

RMG - 22 +10.76% -15.23%

RMG - 23 + 4.5% -13.4%

The alarm setpoint determination for each of these detectors assumes a system inaccuracy of 36%. Therefore, the protective function has not been impacted. Design basis accident analyses take no credit for the function of these detectors. The proposed refueling interval change will have no effect on component availability since the instrumentation has demonstrated reliable operation and expected setpoint drift over the extended calibration interval is acceptable. Additionally, Technical Specification required weekly channel checks and monthly tests will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of the Containment High Range Radiation post-accident monitoring instrument channels.

2. Technical Specification Table 4.1-4, Item 7, Reactor Coolant System Cold Leg Water Temperature, and Item 8, Reactor Coolant System Hot Let Water Temperature instrumentation calibrations are currently specified to be performed on a refueling interval basis. This instrumentation provides post-accident monitoring function in accordance with Regulatory Guide 1.97. The purpose of this surveillance is to verify proper instrument channel response. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of the historical data from 1984 to 1990 for the RCS Hot Leg Water Temperature channels using linear regression and accounting for drift predicts instrument uncertainties at 30 months as follows:

TY958A/960A	+0.28% -0.47%
TI958A/960A	+0.[illegible]95% -0.[illegible]8%

The allowable tolerance for these instrument channels is ±0.5%. Thus, extension of the calibration interval to 30 months does not adversely impact operation of the Hot Leg Water Temperature instrument channels. The RCS Cold Leg Water Temperature instrument channel refueling interval surveillances were added in 1990. Since there is only one set of recorded surveillance data, linear regression cannot be performed. However, extension of the surveillance interval to 30 months can be supported based on equivalence to the Hot Leg Water Temperature Channels. The electronic modules/instruments in the RCS Cold Leg Water Temperature loops are equivalent to those in the Hot Leg in type, tolerance, manufacturer, number of components and model. Equivalent components are located in equivalent environments. Thus, the assessment of Hot Leg Water Temperature surveillance data can be applied to Cold Leg components, and on this basis extension of the surveillance interval to 30 months is justified. The proposed refueling interval change will have no effect on component availability since the Hot Leg Water Temperature instrumentation has demonstrated reliable operation and expected drift over the extended calibration interval is acceptable, and the Cold Leg Water Temperature instrumentation is expected to respond similarly since it is essentially equivalent to the Hot Leg Water Temperature instrumentation. Additionally, Technical Specification required weekly channel checks will allow operator verification of both instrument channels' performance. Therefore, the proposed change has no effect on the safety function of the RCS Hot Leg or Cold Leg Water Temperature post-accident monitoring instrument channels.

3. Technical Specification Table 4.1-4, Item 9, Reactor Coolant System Pressure channel calibration is currently specified to be performed on a refueling interval basis. This instrumentation provides a post-accident monitoring function in accordance with Regulatory Guide 1.97. The purpose of this surveillance is to verify proper instrumentation channel response. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of pressure transmitter historical calibration data for PT949/PT963 using linear regression, accounting for drift and considering uncertainties associated with the linear regression model, as well as uncertainty associated with the transmitters due to random variation and drift, the total uncertainty at 30 months is predicted to be +0.17%, -0.06% of span. This is within the existing allowable tolerance of ± 0.25%. The proposed refueling interval change will have no effect on component availability since the pressure transmitters have demonstrated reliable operation and expected drift over the extended calibration interval is acceptable. Additionally, Technical Specification required weekly channel checks will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of the RCS Pressure post-accident monitoring instrument channels.

4. Technical Specification Table 4.1-4, Item 10, Steam Generator Pressure channel calibration is currently specified to be performed on a refueling interval basis. This instrumentation provides a post-accident monitoring function in accordance with Regulatory Guide 1.97. The purpose of this surveillance is to verify proper instrumentation channel response. The proposed change will extend the interval between successive calibrations to once every 24 months. An evaluation of pressure transmitter historical calibration data using linear regression, accounting for drift and considering uncertainties associated with the transmitter due to random variation and drift, the total uncertainty at 30 months is conservatively predicted to be +8.7%, -11.9% of span. This is within the existing allowable tolerance of ± 12.5%. The proposed refueling interval change will have no effect on component availability since the pressure transmitters have demonstrated reliable operation and expected drift over the extended calibration interval is acceptable. Additionally, Technical Specification required weekly channel checks will allow operator verification of instrument channel performance. Therefore, the proposed change has no effect on the safety function of the Steam Generator Pressure post-accident monitoring instrument channels.

CHANGE NO. 6 - REACTOR BUILDING

1. Technical Specification Section 4.4.1.3, Isolation Valve Functional Tests, currently specifies remotely operated reactor building isolation valve stroke testing during each refueling period for those valves which cannot be stroked during operation. The Reactor Building Isolation Valves close fluid penetrations not required for operation of the engineered safeguards systems in order to prevent leakage of radioactive materials to the environment. The purpose of this surveillance is to verify proper valve disc movement and valve stroke time limit. The proposed change will extend the interval between successive tests to once every 24 months. Refueling outage surveillance records for the Reactor Building isolation valve stroking test were evaluated for the last four (4) refueling outages. This provided 13 separate data points where one or more of the 13 isolation valves were tested to assess performance of the isolation valves. All valve test data from March, 1985 to January, 1990 (58 months) were determined to be acceptable and no valves have required any corrective action to the stroking mechanisms. The proposed refueling interval change will have no effect on component availability since the Reactor Building isolation valves have demonstrated reliable operation. Therefore, the proposed change has no effect on the safety function of the Reactor Building Isolation Valves.

2. Technical Specification Section 4.4.1.7 (2), Operability of Purge Valves, currently specifies visual examination and durometer testing of the rubber seats on purge valves each refueling interval. The Reactor Building Purge System is not normally in operation while the reactor is critical but is used only prior to containment entry or during refueling. The purge valves provide the safety function of containment isolation and are required to maintain containment integrity when the reactor building atmosphere is being purged. The purpose of the surveillance is to detect degradation (e.g. cracking, brittleness, etc.) and to assure timely cleaning, lubrication and seat replacement. The proposed change will extend the interval between successive examination and tests to once every 24 months. Refueling outage surveillance records for the Reactor Building Purge Valve inspections were evaluated for the last four (4) refueling outages. All four valves had seat replacements in 1985. Valve seats were inspected in 1987, 1988, and 1990. All inspections were satisfactory and resulted in no replacements to the seats. The original Technical Specification requirement was to replace seats the first refueling outage following five (5) years of seat service. Based on industry experience, vendor recommendations and specific TMI-1 purge valve experience, the Technical Specification was amended to require seat replacement as a function of physical inspection and durometer hardness testing and not on years of service. In June and July 1991, visual inspection and hardness testing was performed on three (3) valves. Observed condition was satisfactory and all seats tested 60 or better on the Shore "A" hardness test. Therefore, from March, 1985 to July, 1991, all valve seat inspections and tests have been satisfactory and no valves have required any repair or replacement of

the rubber seats. The proposed refueling interval change will have no effect on component availability since the Reactor Building Purge Valves have maintained their integrity as containment isolation barriers. Additionally, Technical Specification required quarterly leak testing provides a direct indication of the ability of the valve seat to perform its design function. Therefore, the proposed change has no effect on the safety function of the Reactor Building Purge Valves.

3. Technical Specification Section 4.4.4.1.b.2 and 4.4.4.1.b.3, Hydrogen Recombiner System, currently specifies a visual examination of the system to verify no evidence of abnormal conditions, and requires a verification of reaction chamber gas temperature during a system functional test, at least once per refueling interval. The Hydrogen Recombiner System serves as a means of controlling combustible gas concentrations in containment following a loss of coolant accident. The purpose of this surveillance is to verify the structural integrity and proper operation of the system. The proposed change will extend the interval between successive examination and test to once every 24 months. The results of the surveillance tests from 1985 to 1990 indicates that there is no evidence of abnormal conditions in the system and the reaction chamber gas temperature can be maintained greater than or equal to 1200° F for at least four hours. The proposed refueling interval change will have no effect on component availability since the Hydrogen Recombiner System has demonstrated reliable operation. Additionally, Technical Specification required functional testing of the Hydrogen Recombiner System once per six (6) months to demonstrate minimum reaction chamber gas temperature can be maintained for at least two (2) hours provides additional assurance of continued system operability. Therefore, the proposed change has no effect on the safety function of the Hydrogen Recombiner System.

4. Technical Specification Section 4.4.4.1.b.4, Hydrogen Recombiner System, currently specifies that the heater electrical circuits be subject to a continuity and resistance to ground test at least once per refueling interval. The electrical heaters provide the energy needed to recombine the post-accident generated hydrogen with oxygen. The purpose of this surveillance is to provide assurance that the hydrogen recombiner heater electrical circuits are available to perform their post-LOCA function. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of surveillance test results from 1985, 1986, 1988, 1989, and 1991 indicates that each of these tests were successfully performed with no deficiencies. The proposed refueling interval change will have no effect on component availability since the heater elec rical circuits have demonstrated reliable operation. Additionally, Technical Specification required functional tests of the Hydrogen Recombiner System once per six months with a reaction chamber gas temperature maintained at greater than 600°F provides continued assurance of heater circuit operability. Therefore, the proposed change has no effect on the safety function of the Hydrogen Recombiner System.

CHANGE NO. 7 - EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING.

1. Technical Specification Section 4.5.1.1.a, Sequence and Power Transfer Test currently specifies that a test be conducted to demonstrate operability of the emergency loading sequence and power transfer during each refueling interval. The Engineered Safeguards Emergency Sequence and Power Transfer Test is performed to verify that the emergency loading sequence and automatic power transfer circuitry is operable and that the emergency power system will respond promptly and properly when required. This system is needed to mitigate postulated design basis events such as a loss of coolant accident (LOCA) followed by a loss of offsite power. The test is considered satisfactory if the following valves have completed their travel on normal power and transferred to the emergency power source (diesel generator) subsequent to a bus under-voltage condition.

    a. M. U. Pump
    b. D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
    c. R. B. Cooling Pump
    d. R. B. Ventilators
    e. D. H. Closed Cycle Cooling Pump
    f. N. S. Closed Cycle Cooling Pump
    g. D. H. River Cooling Pump
    h. N. S. River Cooling Pump
    i. D. H. and N. S. Pump Area Cooling Fan
    j. Screen House Area Cooling Fan
    k. Spray Pump
    l. Motor Driven Emergency Feedwater Pump

    Following successful transfer to the emergency diesel, the diesel generator breaker is opened to simulate trip of the generator, then reclosed to verify block load on the reclosure. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of the surveillance test data history was performed to determine if any recurring deficiencies were encountered which may impact safety due to the extended test interval. The latest test data available for evaluation was conducted on January 6, 1990. During this testing, deficiencies were noted such as valve MU-V3 closing (procedure did not specify required block), EF-P-2A started in excess of allowable band (timing relays with poor repeatability), and EF-P-2A did not start when taken out of pull to lock (error in testing method in procedure). Earlier deficiency reports for the testing showed the only recurring problem to be the Emergency Feedwater Pump starting in excess of allowable band due to relay timers with poor repeatability. The existing relays are being changed out with a higher quality timing relay which will eliminate this concern in the future. Additional periodic testing and surveillance programs are performed in accordance with Technical Specification requirements as described below:

a. Engineered Safeguards Actuation Logic is tested at a three month interval, which demonstrates the operability of the emergency loading sequence logic by exercising the safeguard logic channels for the components listed above, including the Motor Driven Emergency Feedwater Pump auto start contact. The pump is started quarterly by EFW actuation logic testing.

b. A monthly test of the emergency power system is performed to demonstrate that the emergency diesel generator will start after receipt of a manual start signal and can supply load up to its nameplate rating.

c. The emergency diesel generator units are subjected to an annual surveillance inspection and mechanical overhaul to ensure the units are maintained in peak operating condition.

The proposed refueling interval change will have no effect or component availability since the equipment and actuation logic have demonstrated reliable operation and additional surveillance provide added assurance of system and component availability. Therefore, the proposed change has no effect on the safety function of the emergency loading sequences and automatic power transfer circuitry or on the emergency power system.

2. Technical Specification Section 4.5.2.1, High Pressure Injection, currently specifies that High Pressure Injection (HPI) System pumps and high point vents be vented and a system test conducted during each refueling interval. The HPI System provides emergency core cooling for small-break Loss-of-Coolant Accidents by injecting borated water from BWST into the RCS cold legs. The purpose of this surveillance is to verify operability of the system pumps and valves by demonstrating acceptable HPI flow and proper valve movement. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of surveillance tests from 1985 to 1990 indicates all tests were performed satisfactorily. Additionally, quarterly testing of the pumps and valves provides additional indication of system conditions between refueling outages. The proposed refueling interval change will have no effect on system or component availability since the components have demonstrated reliable operation and additional quarterly testing provides added assurance of system and component availability. Therefore, the proposed change has no effect on the safety function of the HPI System.

3. Technical Specification Section 4.5.2.2, Low Pressure Injection, currently specifies that Low Pressure Injection (LPI) System pumps and high point vents be vented and a system test conducted during each refueling period. The LPI System provides emergency core cooling for Loss-of-Coolant-Accidents by injecting a stored supply of borated water into the reactor vessel. The purpose of this surveillance is to verify operability of the system pumps and valves by demonstrating acceptable LPI flow and proper valve movement. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of surveillance tests from 1985 to 1990 indicates all tests were performed satisfactorily. Additionally, quarterly testing of the LPI pumps and valves provides added assurance of LPI System component availability between refueling outages. The proposed refueling interval change will have no effect on system or component availability since the components have demonstrated reliable operation and additional quarterly testing provides added assurance of system and component availability. Therefore, the proposed change has no effect on the safety function of the LPI System.

4. Technical Specification Section 4.5.2.3, Core Flooding, currently specifies that a system test be conducted during each refueling period. The Core Flooding System provides core protection in the event of a major reactor coolant system rupture. This system floods the core with borated water without dependence on actuation signals, electrical power supplies or operator action. The purpose of this surveillance is to verify operation of the system by demonstrating proper operation of the Core Flooding tank discharge line check and isolation valves. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of surveillance test results from 1985 to 1990 indicates that the check and isolation valves in the core flood tank discharge lines have operated properly. Test results have verified that all valves in the flow path have opened. The proposed refueling interval change will have no effect on system or component availability since the system has continually demonstrated reliable operation. Therefore, the proposed change has no effect on the safety function of the Core Flooding System.

5. Technical Specification Section 4.5.3.1.a.1, Reactor Building Spray System, currently specifies a system test be conducted at each refueling interval, simultaneously with the test of the emergency loading sequence. The Reactor Building Spray System provides containment cooling and reduction of airborne fission products following a Loss-of-Coolant-Accident. The purpose of this surveillance is to verify operability of the system pumps and valves. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of the five (5) surveillance test results conducted from 1985 to 1990 indicates that all tests were performed satisfactorily. Additionally, quarterly testing of the Reactor Building Spray System components provides added assurance of system component availability between refueling outages. The proposed refueling interval change will have no effect on system or component

availability since the components have demonstrated reliable operation and additional quarterly testing provides added assurance of system and component availability. Therefore, the proposed change has no effect on the safety function of the Reactor Building Spray System.

6. Technical Specification Section 4.5.3.1.b, Reactor Building Cooling and Isolation Systems, currently specifies a system test be conducted during each refueling period. The Reactor Building Cooling and Isolation System removes heat from the Reactor Building in the event of a Loss-of-Coolant-Accident. The purpose of this surveillance is to demonstrate proper operation of the system by verifying proper actuation of the system valves and establishment of river water flow through the coolers. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of the four (4) surveillance test results conducted from 1985 to 1990 indicates that all tests were performed satisfactorily. Additionally, quarterly testing of the Reactor Building Cooling and Isolation System components provides added assurance of system component availability between refueling outages. The proposed refueling interval change will have no effect on system or component availability since the components have demonstrated reliable operation and additional quarterly testing provides added assurance of system and component operability. Therefore, the proposed change has no effect on the safety function of the Reactor Building Cooling and Isolation System.

7. Technical Specification Section 4.5.4.2, Decay Heat Removal System Leakage, currently specifies leakage testing and visual inspection of the system piping and components during each refueling period. The Decay Heat Removal System provides a means for removing decay heat when the reactor is shutdown, and provides a means of automatically injecting a stored supply of borated water into the reactor vessel (Low Pressure Injection) in the unlikely event of a LOCA. The purpose of this surveillance is to reduce to as low as practicable leakage from systems outside containment that could or would contain highly radioactive fluids during a serious accident or transient. The proposed change will extend the interval between successive tests and inspections to once every 24 months. An evaluation of the results of Decay Heat Removal System leaks inspections from 1985 to 1990 indicates evidence of only a very small amount of leakage. The acceptance criteria states that total system leakage must not exceed six gallons per hour. The largest amount of leakage found during the testing period was 0.13 gallons per hour. The proposed refueling interval change will have no effect on system integrity since leakage testing and inspection history has continually demonstrated minimal system leakage, well within the acceptance criteria. Therefore, the proposed change has no effect on the safety function of the Decay Heat Removal System.

CHANGE NO. 8 - EMERGENCY POWER SYSTEM

1. Technical Specification Section 4.6.1.b, Diesel Generators, currently specifies emergency diesel generator automatic start and load testing every refueling interval. The emergency diesel generators are utilized as an emergency source of power to plant auxiliaries during loss of offsite power conditions. The purpose of this surveillance is to verify that the emergency power system will respond promptly and properly when required, and is performed in conjunction with

   Technical Specification Section 4.5.1.1 requirements (Reference Change No. 7, Item 1). The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of test results, as described in Change No. 7, Item 1, confirmed no recurring deficiencies were encountered. Additionally, Technical Specification required monthly manual diesel generator start and load tests, and annual inspections and mechanical overhauls provide added assurance of component availability. The proposed refueling interval change will have no effect on component availability since the components have demonstrated reliable operation and additional testing, inspection and overhaul provides added assurance of system operability. Therefore, the proposed change has no effect on the safety function of the Emergency Diesel Generators.

2. Technical Specification Section 4.6.2.d, Station Batteries, currently specifies battery load tests be conducted at a frequency not to exceed refueling periods. The 250/125 Volt DC System provides a source of reliable, continuous DC power for control power, diesel generator auxiliaries, vital instrumentation and DC pump motors. The purpose of this surveillance is to verify adequate battery capacity to meet the calculated load requirements. The proposed change will extend the interval between successive tests to once every 24 months.

   The entire B station battery bank was replaced during the 6R refueling outage, and the entire A station battery bank was replaced during the 7R refueling outage. The 8R refueling outage station batteries load test performed in January, 1990 demonstrated the A battery to have 103.08% capacity and the B battery to have 99.08% capacity. The 9R refueling outage load test performed in October, 1991 demonstrated the A battery to have 108.3% capacity and the B battery to have 105.8% These tests were performed utilizing recommendations contained in IEEE Standard 450-1987 as a guideline.

   Pilot cells are subject to weekly and all cells are subject to monthly surveillance readings in accordance with Technical Specifications. These readings include specific gravity, cell temperature, cell liquid level readings and individual cell voltages. These readings provide a reliable indication of the batteries condition. A review of the weekly and monthly reports for the last year was conducted. This review showed that random minor problems were detected and corrected such as high float voltage, low specific gravity and low room temperature. Based on this program a precipitous failure of the battery is extremely unlikely. Deep cycle loading like the load test ages the battery. By extending the load test cycle from 18 months to 24 months, the total number of discharges the battery will be subjected to over its life span will be reduced, thereby prolonging the total life expectancy of the battery. GPUN has revised the load test procedure to address the capacity of each battery section. This will provide better indication that each battery section will satisfy its intended safety function. Additionally, the test procedure adds a 3% conservative factor to the calculated acceptance criteria.

The proposed refueling interval change will have no effect on station battery availability since the components have demonstrated adequate reliability and additional Technical Specification required weekly/monthly monitoring programs provide added assurance that the station batteries are capable of providing adequate voltage. Therefore, the proposed change has no effect on the safety function of the Station Batteries.

3. Technical Specification Section 4.6.3, Pressurizer Heaters, currently specifies that testing be performed at least once each refueling to demonstrate that pressurizer heater groups 8 and 9 can be transferred from the normal power bus to the emergency power bus and energized, and that the heaters will trip following an Engineered Safeguard (ES) signal. These pressurizer heaters are required to maintain natural circulation conditions in the event of a loss of offsite power. The purpose of this surveillance is to demonstrate that pressurizer heater load can be transferred from the normal power supply to the emergency power supply, and that the ES actuation interlock function properly. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of the four (4) surveillance test results conducted from 1985 to 1990 indicates all tests were successfully performed with no deficiencies relating to the pressurizer heaters, power supply system, or controls. The proposed refueling interval change will have no effect on pressurizer heater availability since these components have continually demonstrated reliable operation. Therefore, the proposed change has no effect on the safety function of the Pressurizer Heaters.

CHANGE NO. 9 - MAIN STEAM ISOLATION VALVES

1. Technical Specification Section 4.8.2, Main Steam Isolation Valves, currently specifies closure time testing be conducted at intervals not to exceed the normal refueling outage. The Main Steam Isolation Valves (MSIV) provide containment isolation in the event of steam line breaks upstream or downstream of the MSIV's. The purpose of this surveillance is to provide assurance that the MSIV's close fully in less than 120 seconds under no flow, no load conditions. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of surveillance test results from 1985 to present, which provides 14 data points, indicates that all MSIV's satisfactorily met the acceptance criteria for all refueling outage closure time tests. At no time over 58 months from March, 1985 to January, 1990 did any valve fail to meet the Technical Specification stroking time limit which would require corrective action. In addition, these valves are tested monthly to check stem movement up to 10% under normal flow and load conditions per Technical Specification 4.8.1 requirements. The stem freedom records from October, 1985 to August, 1991 were evaluated and the valves operated satisfactorily. The proposed refueling interval change will have no effect on MSIV operability since these valves have demonstrated reliable operation and additional monthly testing of valve stem movement provide added assurance of component availability between refueling outage tests. Therefore, the proposed change has no effect on the safety function of the MSIV's.

CHANGE NO. 10 - DECAY HEAT REMOVAL CAPABILITY

1. Technical Specification Section 4.9.1.4 and 4.9.1.5, Emergency Feedwater System - Periodic Testing, currently specifies Emergency Feedwater (EFW) System pump and control valve testing be performed on a refueling interval basis. The EFW system delivers water to the steam generators (OTSG) during a loss of main feedwater, loss of all reactor coolant pumps, upon receipt of a high containment pressure signal or upon a low OTSG level. The purpose of these surveillance tests is to verify operability of EFW system components. The proposed change will extend the interval between successive tests to once every 24 months. Automatic start of the EFW pumps is performed quarterly as part of Technical Specification Table 4.1-1, Items 51.a and 51.d instrumentation channel tests. Accordingly, there is no real change in test frequency by going to a 24 month operating cycle. An evaluation of past test results from 1988 and 1990 for EFW control valve testing indicates no equipment deficiencies were encountered from the inability to manipulate the EFW control valves. An evaluation of the Technical Specification refueling interval surveillance test results from 1983 to 1990 indicates no equipment deficiencies were encountered which resulted in the inability of the EFW pumps to pump water from the condensate storage tanks to the OTSG. The proposed refueling interval change will have no effect on the operability of the EFW System since the components have demonstrated

reliable operation. Additional monthly pump start testing for Inservice Testing requirements, quarterly Technical Specification required EFW control valve actuation logic testing, and quarterly EFW control valve remote manual stroke testing for IST requirements provides added assurance of component availability between refueling outages. Therefore, the proposed change has no effect on the safety function of the EFW system.

CHANGE NO. 11 - REACTOR COOLANT SYSTEM VENTS

1. Technical Specification Section 4.11.1, Reactor Coolant System Vents, currently specifies that each power operated valve in the vent path shall be cycled through one complete cycle of full travel once per refueling interval. The Reactor Coolant Venting System provides venting of gases trapped in the reactor vessel head and in both RCS hot legs and also provides pressurizer degassing capability from a remote location to promote core cooling and to re-establish pressurizer level. The purpose of the surveillance is to verify valve stroke time and remote position indication for vent path valves to ensure the vents are able to perform their design function. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of surveillance test results from 1985 to 1990 indicates that the stroke time for all the RCS venting valves are within acceptable values. In addition, the valve position indication tests have satisfactorily confirmed that a valve remote position indicator in the Control Room agrees with the local valve stem position. The proposed refueling interval change will have no effect on the operability of the Reactor Coolant System Vents since the vent path power operated valves have demonstrated reliable operation. Therefore, the proposed change has no effect on the safety function of the Reactor Coolant System Vents.

CHANGE NO. 12 - AIR TREATMENT SYSTEM

1. Technical Specification Sections 4.12.1.1 and 4.12.1.3, Emergency Control Room Air Treatment System, currently requires a HEPA filter and charcoal adsorber bank pressure drop test at system design flow rate, and an automatic initiation test of the Control Building isolation and recirculation dampers at least every refueling interval or once every 18 months, whichever comes first. The Emergency Control Room Air Treatment System provides a habitable environment within the Control Building Envelope to ensure Control Room personnel are adequately protected against the effects of accidental releases of radioactive or toxic gases. The purpose of this surveillance is to verify that the system and associated components are capable of performing its design functions. The proposed change will extend the interval between successive tests to once every 24 months. Both of these tests are performed monthly as part of Technical Specification 4.12.1.2.d which requires operation of the system fan/filter circuit at least 10 hours every month. Therefore, on a monthly basis,

corrective actions can be taken to replace the filter banks if needed. The only factor affecting the differential pressure across the roughing HEPA/charcoal adsorber banks of the filters is dirt accumulation in the filter banks due to extensive operation of the units. The emergency fan/filter units are standby units, and operate just 10 hours per month as part of the surveillance test to determine system operability status. An evaluation of the test data results for an approximate two year period (1989 through 1990) demonstrate that the combined differential pressure across the roughing HEPA/charcoal adsorber banks at design flow remained unchanged for 24 months in an approximate range of 2.1 in. w.g., which is well below the acceptable limit of 6.0 in. w.g. Therefore, the proposed refueling interval change will have no effect on the operability of the Emergency Control Room Air Treatment System since the HEPA/charcoal adsorber bank units and dampers have demonstrated reliable operation over a 24 month period, and system operability is continually ensured by fan/filter circuit operation at least 10 hours every month in accordance with Technical Specification requirements. Therefore, the proposed change has no effect on the safety function of the Emergency Control Room Air Treatment System.

2. Technical Specification Sections 4.12.2.1 and 4.12.2.2.e, Reactor Building Purge Air Treatment System, currently specifies pressure drop testing across the HEPA filter/charcoal adsorber banks at system design flow rate at least once per refueling interval or once per 18 months whichever comes first, and design flow testing of the fans each refueling. The Reactor Building Purge System is used during normal operation and prior to shutdown to purge the reactor building atmosphere in order to reduce airborne radioactivity levels prior to personnel entry, and to filter contaminated reactor building atmosphere during hydrogen purging if operated as a backup to the hydrogen recombiner. The purpose of this surveillance is to verify that the system filter is not clogged and that the system flow is still within design flow to ensure that exhaust capability and iodine removal capability are not degraded. The proposed change will extend the interval between successive tests to once every 24 months. An evaluation of filter D/P test data for the last three (3) surveillance tests (1987-1990) indicates that filter D/P and system flow rate test have continually performed within acceptance criteria. Further, it is noted that the rate increase of the filter D/P is 0.5 in. w.g. in 2 years, which is well within the acceptable limit of 6.0 in. w.g. at design flow. During normal plant operation, the purge isolation valves are limited to 30° open which limits the system flow rate to 14,000 CFM from a design flow rate of 50,000 CFM. At this limited flow rate, reactor building dust (if any) that could clog the HEPA filters can not be easily captured by the air flow at the inlet. Therefore, clogging of the HEPA filters is unlikely to occur during any additional hours of intermittent operation during the extension period. The proposed refueling interval change will have no effect on the operability of the Reactor Building Purge Air Treatment System since the HEPA filter/charcoal adsorber bank units and fan units have demonstrated reliable operation, and the additional period of plant

operation is not expected to degrade filter performance.

Technical Specification 4.12.2.2.a is administratively revised to clarify the refueling interval requirement for test and sample analysis. The refueling interval surveillance requirement ensures proper filter performance prior to fuel handling during a refueling outage shutdown. Therefore, there is no impact on the basis for this surveillance.

Therefore, the proposed change has no effect on the safety function of the Reactor Building Purge Air Treatment System.

3. Technical Specification Section 4.12.3.1, Auxiliary and Fuel Handling Building Air Treatment System, currently specifies a pressure drop test across the HEPA filter and adsorber bank at system design flow, charcoal sample analysis, and a system fan design flow rate test be conducted at least once per refueling interval. The Auxiliary and Fuel Handling Building Air Treatment System maintains a negative pressure in these buildings with respect to the outside environment, and to filter air normally exhausted from potentially radioactive areas of these buildings. The purpose of the D/P and flow test is to verify that the filters are not clogged and that the system flow is within design flow to ensure that exhaust capability and iodine removal capability are not degraded. The charcoal sample analysis ensures that the iodine removal efficiency of the charcoal is within the Technical Specification limit of 90% The proposed change will extend the interval between successive tests and analysis to once every 24 months. The fan flow test is performed monthly as part of Technical Specification 4.12.3.2.d which requires operation of the system fans at least 10 hours every month. Therefore, on a monthly basis, corrective actions can be taken to replace the fans or filters if needed. Any additional filter clogging over the extended plant operating interval will be compensated for by static pressure regulating dampers in the supply and exhaust ducts of each building, which will modulate dampers to maintain the required negative building pressure. Review of charcoal sample analysis results over an 18 month operating cycle indicates a charcoal efficiency of still over 99%, and after a period of 2 1/2 years the efficiency was approximately 96%.

Maintenance experience indicates for the most part filters are not normally replaced until six or seven years of use due to degradation of the potassium iodide impregnation. The proposed refueling interval change will have no effect on the operability of the Auxiliary and Fuel Handling Building Air Treatment System since system operability is continually ensured by fan operation at least 10 hours every month in accordance with Technical Specification requirements, and building negative pressure is maintained by static pressure regulators which automatically adjust for any reduced flow rate due to filter clogging. The additional period of operation is not expected to degrade filter performance. The charcoal adsorbers have demonstrated insignificant reductions of efficiency over the current cycle length and typically maintain acceptable removal efficiencies for a period of six to seven years. Therefore, the proposed change has no effect on the safety function of the Auxiliary and Fuel Handling Building Air Treatment System.

CHANGE NO. 13 - REACTOR INTERNALS VENT VALVES

1. Technical Specification Section 4.16.1, Reactor Internals Vent Valves, currently requires a visual inspection of accessible valve surfaces, verification of valve position, and verification of force required to fully open the valves each refueling shutdown. The eight (8) vent valves relieve the pressure generated by steaming in the core following a LOCA so that the core will remain sufficiently cooled. The purpose of this surveillance is to assure the operability of each valve and to detect and repair any degradation of sealing surfaces and hinges. The proposed change will extend the interval between successive testing and inspections to once every 24 months. An evaluation of surveillance test results from 1985 to 1990 (4 data points for each of the 8 valves) indicates satisfactory performance for all valves, with no observable degradation and all valves operated freely and remained in the closed position. Over this 56 month period, there was no change in valve condition which would have prevented them from performing their safety function. The proposed refueling interval change will have no effect on the operability of the reactor internals vent valves since these valves have demonstrated reliable operation over the period cited above. Therefore, the proposed change has no effect on the safety function of the Reactor Internals Vent Valves.

CHANGE NO. 14 - SHOCK SUPPRESSORS (SNUBBERS)

1. Technical Specification Section 4.17.1.b and 4.17.1.e currently specifies visual inspection of each type of accessible safety related snubber which experienced zero (0) inoperable snubbers per the previous inspection every 18 months; and a functional test of a representative sample of snubbers at least once each refueling interval. The purpose of the inspection and test is to provide assurance of snubber operability and reliability. Snubbers are provided to prevent unrestrained pipe motion under dynamic loads, as may occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The current TMI-1 hydraulic snubber population is 181. The proposed change will extend the intervals between successive visual inspections from 18 months to 24 months, and successive functional tests to once every 24 months. The inspection period corresponding to one inoperable snubber is revised by a similar percentage as compared to the previous schedule. Evaluation of the 100% visual inspection results over the period 1986 to 1990 indicate only one (1) individual hydraulic snubbers failure. Evaluation of the functional tests results performed on 10% of the total population over the period 1986 to 1990 indicate no failure of hydraulic snubbers. The proposed surveillance extensions will have no effect on snubber reliability since the hydraulic snubbers have demonstrated reliable operation over the period cited above. In addition, the nature of the Technical Specification will force a more frequent inspection schedule if the failure rate per inspection period increases, and is therefore self-correcting. Snubber seal service

life is also continuously monitored and maintained to ensure that the seal service life is not exceeded at any time. The above data confirms that the snubber program is more than adequate in minimizing snubber failure. Therefore, the proposed change has no effect on the safety function of the snubbers.

IV. NO SIGNIFICANT HAZARDS CONSIDERATIONS

GPUN has determined that this Technical Specification Change Request involves no significant hazards consideration as defined by NRC in 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment extends the interval between successive refueling outage based surveillances to once every 24 months for those surveillances evaluated herein and, maintains the existing surveillance interval restriction for those systems and equipment not evaluated for extension. This change does not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operations. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded beyond that obtained from the currently defined refueling outage interval. Assurance of system and equipment availability is maintained. This change does not involve any change to system or equipment configuration. Therefore, this change does not increase the probability of occurrence of the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment extends the interval between successive refueling outage based surveillances to once every 24 months for those surveillances evaluated herein and maintains the existing surveillance interval restriction for those systems and equipment not evaluated for extension. This change does not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operation. This change does not involve any change to system or equipment configuration. Therefore, this change is unrelated to the possibility of creating a new or different kind of accident from any previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment extends the interval between successive refueling outage based surveillances to once every 24 months for the surveillances evaluated herein, and maintains the existing surveillance interval restriction for those systems and equipment not evaluated for extension. This change does not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operation. The reliability of systems and components is not degraded beyond that obtained from the currently defined refueling outage interval. Assurance of system and equipment availability is maintained. Therefore, it is concluded that operation of the facility in accordance with the proposed amendment

does not involve a significant reduction in a margin of safety. The proposed extension of the refueling outage interval surveillances to once every 24 months does not degrade the reliability of systems and components beyond that obtained from the currently defined refueling outage interval. Reliable performance of the systems and equipment effected by this change has been demonstrated. Implementation of the proposed amendment will maintain the required level of assurance of system and equipment availability. The surveillance interval for systems and equipment that have not been evaluated for extension are editorially changed to specify the existing refueling interval requirement. Thus, operation of the facility in accordance with the proposed amendment involves no significant hazards considerations.

V. IMPLEMENTATION

It is requested that the amendment authorizing this change become effective upon issuance.