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10 CFR 50.90

September 1, 1992

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

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

SUBJECT: Limerick Generating Station, Units 1 and 2
Technical Specifications Change Request

Gentlemen:

Philadelphia Electric Company is submitting Technical Specifications Change Request (TSCR) No. 92-02-0, in accordance with 10 CFR 50.90, requesting an amendment to the Technical Specifications (TS) (Appendix A) of Operating License Nos. NPF-39 and NPF-85. Information supporting this Change Request is contained in Attachment 1 to this letter, and the proposed TS markup pages are contained in Attachment 2.

This submittal requests changes to TS surveillance intervals to facilitate a change in the Limerick Generating Station (LGS), Units 1 and 2, refueling cycles from 18 months to 24 months. The 24 month refueling cycle will require a change from the current 18 month TS surveillance testing interval (i.e., a maximum of 22.5 months accounting for the allowable grace period) to a 24 month testing interval (i.e., a maximum of 30 months accounting for the allowable grace period). These TS changes were evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, and are being proposed accordingly.

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As discussed in our letter dated February 11, 1992, this is the second of three Change Requests being submitted to the NRC to support the current change to 24 month refueling cycles at LGS, Units 1 and 2. This Change Request involves a proposed change to the TS surveillance intervals for non-instrumentation (i.e., non instrument calibration) TS line items, e.g., pump, valve, and flow testing, logic system functional testing, and response time testing. Proposed changes to TS surveillance intervals for instrument calibrations and the remaining TS line items to support 24 month refueling cycles will be requested in a forthcoming (i.e., third) Change Request No. 92-03-0.

The specific TS page markups contained in Attachment 2 reflect the proposed change to 24 month testing for each specific TS Surveillance Requirement identified and evaluated in this Change Request. The TS markups are being provided for information only. The final changed TS pages, which will reflect the combined changes proposed in this and the following Change Request, will be provided with Change Request No. 92-03-0.

Accordingly, we request that the NRC review the TS changes proposed in this Change Request by November 1992 in order to support approval of this and the next Change Request prior to the expiration of the current TS surveillance interval limits.

If you have any questions regarding this matter, please contact us.

Very truly yours,

G. J. Beck, Manager
Licensing Section

Attachments

cc: T. T. Martin, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS
W. P. Dornsife, Director, PA Bureau of Radiological Protection

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

G. R. Rainey, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company; the Applicant herein; that he has read the foregoing Application for Amendment of Facility Operating License Nos. NPF-39 and NPF-85 (Technical Specifications Change Request No. 92-02-0) to facilitate a change in the Limerick Generating Station, Units 1 and 2 refueling cycles from 18 months to 24 months, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

G R Rainey

Vice President

Subscribed and sworn to

before me this 1st day

of September 1992.

Erica A. Santori

Notary Public

Notarial Seal
Erica A. Santon, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires July 10, 1995

ATTACHMENT 1

LIMERICK GENERATING STATION
Units 1 and 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST

"Priority 2 (Non-Instrumentation) Line Item Changes in Support of 24 Month Refueling Cycles"

Supporting Information for Changes - 36 pages

Philadelphia Electric Company (PECo), Licensee under Facility Operating Licenses NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively, requests that the Technical Specifications (TS) contained in Appendix A of the Operating Licenses be amended as proposed herein. The proposed changes are indicated on the associated TS page markups for both LGS, Unit 1 and Unit 2, and are contained in Attachment 2.

The proposed TS changes are requested to facilitate the current change in the LGS, Units 1 and 2 refueling cycles from 18 months to 24 months. The 24 month refueling cycle will require a change from the current 18 month TS surveillance testing interval (i.e., a maximum of 22.5 months accounting for the allowable grace period) to a 24 month testing interval (i.e., a maximum of 30 months accounting for the allowable grace period). These proposed TS changes were evaluated in accordance with the guidance provided in NRC Generic Letter (GL) No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

As discussed in our letter dated February 11, 1992, this is the second of three Change Requests being submitted to the NRC to support the current change to 24 month refueling cycles at LGS, Units 1 and 2. This Change Request involves a proposed change to the TS surveillance intervals for non-instrumentation (i.e., non instrument calibration) TS line items, e.g., pump, valve, and flow testing, logic system functional testing, and response time testing. Proposed changes to TS surveillance intervals for instrument calibrations and the remaining TS line items to support 24 month refueling cycles will be requested in a forthcoming (i.e., the third) Change Request No. 92-03-0.

The TS page markups contained in Attachment 2 reflect the proposed change to 24 month testing for each specific TS Surveillance Requirement (SR) identified and evaluated in this Change Request. The TS page markups are being provided for information only. The final changed TS pages, which will reflect the combined changes proposed in this and the following Change Request, will be provided with Change Request No. 92-03-0. Accordingly, we request that the NRC review the TS changes proposed in this Change Request by November 1992 in order to support approval of this and the next Change Request prior to the expiration of the current TS surveillance interval limits.

This Change Request provides a discussion, description, and a safety assessment for each of the proposed TS changes by "group," information supporting a finding of No Significant Hazards Consideration, and information supporting an Environmental Assessment.

<u>Discussion, Description, and Safety Assessment of the Proposed Changes</u>

Because of the volume of TS SRs to be evaluated, specific line item changes were evaluated within each group identified below. Note that the name of each group is merely an administrative title, and is not intended to mean that all of the specific TS requirements related to the group title have been included. The proposed TS changes generically involve changing the surveillance test interval, typically stated as "at least once per 18 months," to "at least once per 24 months." The proposed TS changes also involve specific additional changes to certain SRs that are required because of the change to 24 month refueling cycles, e.g., a change to the percent sampling required, or a change to the total number of years over which sampling occurs. In addition, the proposed TS changes involve a change to certain TS SRs or associated Bases that indicate conformance to a specific Regulatory Guide related to the system being tested, i.e., the proposed change would indicate that the change to a 24 month testing interval would be an exception to the 18 month testing interval guidance specified in the Regulatory Guide.

The proposed TS changes only involve a change to the surveillance intervals; there are no changes to the SRs themselves with the exceptions indicated above which do not change the overall requirement. Additionally, the proposed TS changes do not change the way in which the surveillances are performed. Also, the proposed changes do not involve any physical changes to plant systems or components. The proposed TS changes are described and evaluated below. These changes were evaluated in accordance with the guidance provided in NRC GL No. 91-04.

(1) AC Power: TS SR 4.8.1.1.1.b; page 3/4 8-3.

TS SR 4.8.1.1.1.b requires that each of the "independent circuits between the offsite transmission network and the on-site Class 1E distribution system shall be demonstrated operable at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit." The Class 1E AC power system for each unit is divided into four divisions. The 4 kV bus of each Class 1E load division is provided with connections to two offsite power sources, designated as preferred and alternate power supplies. In addition, provisions exist for connection to a third offsite power source through a spare transformer if there is a failure of one of the two offsite sources or either of the safeguard transformers. Emergency Diesel Generators (EDGs) are provided as a standby power supply if there is a total loss of the preferred and alternate power supplies. This test is to ensure the capability of transferring unit power from the offsite AC power source to the onsite AC power source. Because of system redundancy, the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(2) Fire Rated Assemblies: TS SR 4.7.7.1; Items a, b, and c; page 3/4 7-31.

TS SR 4.7.7.1 requires that each required fire rated assembly and penetration sealing device separating safe shutdown fire areas or separating portions of redundant systems important to safe shutdown within a fire area shall be verified OPERABLE at least once per 18 months by performing a visual inspection of the following:

a. The exposed surfaces of each fire rated assembly.

b. Each fire window, fire damper, and associated hardware.

c. At least 10% of each type of sealed penetration, except internal conduit seals. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% sample with no apparent changes in appearance or abnormal degradation is to be found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

The surveillance test interval stated in the TS SR 4.7.7.1 for the exposed surface of each fire rated assembly and visual inspection of fire dampers and windows would be increased from once per 18 months to once per 24 months in order to accommodate the 24 month refueling cycle. The penetration seal sampling interval would be changed from at least 10% of each type of penetration seal inspected every 18 months so that each penetration seal is inspected once per 15 years to 12.5% of each type of penetration seal inspected every 24 months so that each penetration seal is inspected once per 16 years. Consistent with the proposed changes and the intent of the existing TS SR, the sample size of the additional seals to be inspected upon identification of changes in appearance or abnormal degradations would be changed from 10% to 12.5%.

The LGS Fire Protection Program uses the defense-in-depth approach aimed at preventing fires and minimizing the effect of any fires that occur. This is accomplished through separation of redundant safety systems, an integrated network of components and equipment providing detection and suppression of fires, component design and layout, administrative controls and procedures, and personnel training. The Fire Protection Program uses the defense-in-depth approach to assure that a fire will not prevent the performance of necessary safe shutdown functions and will not cause undue risk to the health and safety of the public. The Fire Protection Program is formulated such that failure of an active or passive component of one fire protection feature is backed-up by another entirely different fire protection feature (e.g., fire rated assemblies, sprinklers, detection, etc.), and the possibility that multiple fire protection features would be impacted simultaneously by a common time-based failure is not likely nor would the overall effectiveness of the program would be significantly compromised by a single component failure.

The fire rated assemblies and penetration seals provide assurance that a fire can be contained to a single fire area and kept from involving portions of redundant systems important to safe shutdown within a fire area prior to detection and extinguishment. By increasing the refueling cycle length, the time interval between inspection of these fire rated assemblies and penetration seals would be increased. An unsatisfactory condition could therefore remain undetected for an additional 7.5 months (i.e., accounting for the grace period) as a result of the proposed change to the TS SR. The probability and subsequent impact on plant safety is considered to be negligible based on the redundant features provided in the Fire Protection Program.

A review of the surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(3) Containment Leakage: TS SR 4.6.1.2.f; page 3/4 6-4
TS SR 4.6.1.2.g; page 3/4 6-4

TS SR 4.6.2.1.d; page 3/4 6-14

a. TS SR 4.6.1.2.f requires that the "main steam line isolation valves be leak tested at least once per 18 months." The test is required to ensure a leakage rate less than or equal to 11.5 scf/hour for any one main steamline through the isolation valves. By increasing the refueling cycle length, the time interval between testing would be increased. The present testing interval with 25% grace is 22½ months. This evaluation provides basis for extending the testing interval one and one-half (1½) months or an increase of ~6.7% to the current 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," maximum limit of 24 months.

As-found Main Steam Isolation Valve (MSIV) leakage tests have been performed at varying intervals. The inboard and outboard MSIVs are tested simultaneously and the leakage rate reported for this containment penetration path is equal to the total leakage measured. This method assumes a single active failure of the better of the two leakage barriers. A review of surveillance test history indicated there was no correlation between degradation of MSIV leakage rates and the time interval between tests.

Therefore, we have concluded that the impact on safety would be small as a result of increasing this test interval from 22½ months to 24 months. To indicate that the 25% allowable grace period cannot be added to the once per 24 month testing interval, the phrase "not to exceed the requirements of 10CFR50, Appendix J" would be added to the end of the TS SR 4.6.1.2.f in addition to the change of "18" months to "24" months.

b. TS SR 4.6.1.2.g requires that "containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months." These tests are performed to ensure that the TS Limiting Condition for Operation, i.e., that the combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment is less than or equal to 1 gpm times the total number of such valves is satisfied. The tests are Type C tests as defined in 10CFR50, Appendix J, and are therefore subject to the restrictions of 10CFR50, Appendix J. In order to accommodate a 24 month refueling cycle, the time interval between testing would be increased from 18 months (i.e., 22.5 months with grace period) to the limit of 24 months as allowed by 10CFR40, Appendix J. To indicate that the 25% allowable grace period cannot be added to the once per 24 month testing interval, the phrase "not to exceed the requirements of 10CFR50, Appendix J" would be added to the end of TS SR 4.6.1.2.g in addition to the change of "18" months to "24" months.

A review of the surveillance test history indicated that the as-found leakage for each of the tests was less than 20% of the value specified by the TS Limiting Condition for Operation. This data supports the conclusion that the impact on safety, if any, is small when the interval is increased to 24 months as allowed by 10CFR50,

Appendix J.

c. TS SR 4.6.2.1.d requires that a drywell-to-suppression chamber bypass leak test be performed at least once per 18 months. The test is performed to verify that there is not an open bypass leakage path between the drywell and suppression pool. In order to accommodate a 24 month refueling cycle, the time limit between tests would be increased to a bounding limit of 30 months. The TS Limiting Condition for Operation requires the drywell-to-suppression chamber bypass leakage be less than or equal to 10% of acceptable $A/\sqrt{k}$ design value of 0.0500 ft^2.

A review of the surveillance test history indicated that the measured leakage in each case was less than 2% of the value specified by the TS Limiting Condition for Operation. This data supports the conclusion that the impact on plant containment integrity, if any, is small as a result of the change from 18 month to 24 month refueling cycle.

Additionally, TS SR 4.6.2.1.d requires that if any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the NRC. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every nine (9) months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

The past performance history of this test provides high confidence that the probability of a need to invoke accelerated testing is very low. The intent of the testing schedule is that the test be performed every refueling outage, and if the test has two consecutive failures, the test would also be required to be performed mid-cycle until two consecutive tests meet the specified limit. The proposed increase in retest interval from nine (9) months to 12 months, as well as resuming a 24 month test schedule after two successful consecutive tests, is also supported by this evaluation. This is consistent with the aforementioned basis to accommodate the change to 24 month testing. No impact on plant safety will result from the proposed change.

(4) Control Room Emergency Fresh Air Supply System:

TS SR 4.7.2.c; Items 1, 2, and 3; pages 3/4 7-6 (Unit 1), 7-6a (Unit 2), and 7-7
TS SR 4.7.2.e; Items 1, 2, and 3; page 3/4 7-7

TS SR 4.7.2.c.1 requires that at least once per 18 months for the Control Room Emergency Fresh Air Supply (CREFAS) System : "Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the systems flow rate is 3000 cfm ± 10%." TS SR 4.7.2.c.2 requires that at least once per 18 months for the CREFAS system: "Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%." TS SR 4.7.2.c.3 requires that at least once per 18 months for the CREFAS system: "Verifying a subsystem flow rate of 3000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980." TS SR 4.7.2.e.1 requires that at least once per 18 months for the CREFAS system: "Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm ± 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge." TS SR 4.7.2.e.2 requires that at least once per 18 months for the CREFAS system; "Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:

a) outside air intake high chlorine, and
b) manual initiation from the control room."

TS SR 4.7.2.e.3 requires that at least once per 18 months for the CREFAS system: "Verifying that on each of the below radiation isolation mode actuation test signals, the subsystem automatically switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm:

a) outside air intake high radiation, and
b) manual initiation from control room."

These tests are required to ensure that the CREFAS system is capable of performing the system's design safety function. The CREFAS system provides filtration for control room fresh air and recirculated air during a high radiation accident and provides filtration for control room recirculated air during a chlorine or offsite toxic chemical release accident to maintain control room habitability. In accordance with TS SRs 4.7.2.c, 4.7.2.d, 4.7.2.f, and 4.7.2.g, the CREFAS system is required to be tested following filter structural maintenance, fire, chemical release, painting, High Efficiency Particulate Air (HEPA) filter replacement, charcoal adsorber replacement, and after 720 hours of operation. This additional testing would detect potential changes in HEPA filter efficiency and carbon adsorber bypass leakage that would also be detected by conducting the 18 month TS surveillance tests. As required by TS SR 4.7.2.b, the CREFAS system is operated at least once per 31 days on a staggered test basis. This test would determine significant failures affecting flow or filter pressure drop that would also be detected by conducting the 18 month TS surveillance test. In addition, the CREFAS system active components and power supplies are designed with redundancy to meet the single active failure criteria, which will ensure system availability in the event of a failure of one of the system components. Based on the above discussion, and the fact that the CREFAS system is normally in standby, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

Additionally, the surveillance interval for TS SRs 4.7.2.c.1 and 4.7.2.c.2 would be footnoted to indicate that the change to 24 month testing is an exception to the 18 month testing interval guidance specified in Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, dated March 1978.

(5) Contaminated Pipe Inspections: TS Section 6.8.4.a; page 6-14.

TS Section 6.8.4.a requires "A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels." Specifically, the program requires an "Integrated leak test requirement for each system at refueling cycle intervals or less." The change to 24 month refueling cycles would increase the nominal testing interval from 18 months to 24 months. The proposed change to the testing requirement has been evaluated and determined that the impact on safety, if any, is small. This conclusion is based on the fact that most portions of the subject systems included in this program are visually inspected while the plant is operating during normal plant testing and/or operator/system engineer walkdowns. In addition Administrative Guideline AG-57, "Guidance for Plant Performance Observation," requires senior station management to perform housekeeping/safety walkdowns which would also serve to detect any gross leakage. If leakage is observed from these systems, corrective actions would be taken to repair the leakage. Finally, the plant Health Physics radiological surveys would also identify any potential sources of leakage. These walkdowns and surveys provide monitoring of the systems at a greater frequency then once per refueling cycle, and support the conclusion that the impact of the proposed change on plant safety, if any, is small.

A review of the surveillance test results indicated there has not been any evidence of gross external leakage. In fact, several instances were noted where minor leakage identified during conduct of the TS SR had been previously identified during the walkdowns described above. The history of the surveillance tests results for the contaminated pipe inspections supports the above conclusion.

(6) Control Rod Drive: TS SR 4.1.3.1.4.a; page 3/4 1-5

TS SR 4.1.3.1.4 requires "The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

1. Close within 30 seconds after receipt of a signal for control rods to scram, and

2. Open when the scram is reset."

The Scram Discharge Volume (SDV) accepts discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. On a quarterly basis throughout the operating cycle, the scram discharge volume vent and drain valves are verified to be operable in accordance with TS SR 4.1.3.1.1.b. This test is capable of identifying potential problems associate with the SDV vent and drain valves. Furthermore, any potential increase in the possibility for blockage to or in the SDV due to increasing the surveillance test interval is small since a review of past system performance did not identify any concerns or problems associated with the SDV. Therefore, increasing the length of the SR interval will not have an impact on the operability of the SDV vent and drain valves.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(7) DC Power: TS SR 4.8.2.1.c; Items 1, 2, 3, and 4; page 3/4 8-11
TS SR 4.8.2.1.d; Items 1 and 2; pages 3/4 8-11 and 8-12
TS SR 4.8.2.1.f; page 3/4 8-12
Bases 3/4.8.2; page B 3/4 8-2

TS SRs 4.8.2.1.c.1 and 4.8.2.1.c.2 require that once every 18 months for the DC power system, the battery cells, cell plates, and battery racks be inspected for physical damage, terminal connections, and corrosion. Also, TS SRs 4.8.2.1.c.3 and 4.8.2.1.c.4 require once every 18 months testing the resistance of each cell-to-cell and terminal connection is less than 150 x 10^{-6} ohm, excluding cable intercell connections, and that the battery chargers will supply the required currents at a minimum of 132 volts for at least eight hours. TS SRs 4.8.2.1.d.1 and 4.8.2.1.d.2 require that once every 18 months for the DC power system the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for the design duty cycle when the battery is subjected to a battery service test, and be adequate to supply a dummy load while maintaining the battery terminal voltage greater than or equal to 105 volts for the nominal 125-volt batteries and 210 volts for the nominal 125/250-volt batteries. TS SR 4.8.2.1.f requires that once per 18 months for the DC power system during shutdown, performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests or is below 90% of the manufacturers rating.

Each unit is provided with physically separate and independent onsite dc electric power systems. There are four divisions of Class 1E dc power for each unit. Divisions I and II are 125/250V systems. Divisions III and IV are 125V systems. In addition, each unit has a 250V non-Class 1E dc system, and a 125/250V non-Class 1E dc system. Each of the four Class 1E dc systems, including the battery bank, charger and distribution system, is independent of the other Class 1E dc systems and of each non-Class 1E dc system. The Class 1E battery loads are diversified among the different Class 1E dc systems so that each system serves loads that are identical and redundant, or are different but redundant with respect to plant safety, or are backup equipment to ac driven equipment. No provision exists for transferring loads between redundant dc systems. Thus, sufficient independence and redundancy exist to ensure performance of minimum safety functions, assuming that there is a single failure. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of the surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

In addition, the proposed change would require a change to Bases Section 3/4.8.2 to indicate that a change to 24 month testing would be an exception to the 18 month testing interval guidance specified in Regulatory Guide 1.129, "Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," dated February 1978, and IEEE Standard 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

(8) Emergency Core Cooling Systems: TS SR 4.5.1.c.2.a; page 3/4 5-5
TS SR 4.5.1.d.2.b; page 3/4 5-5
TS SR 4.7.3.c.2; page 3/4 7-10

a. TS SR 4.5.1.c.2.a requires that once every 18 months for the High Pressure Coolant Injection (HPCI) system: "The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥ 200 psig plus head and line losses, when steam is being supplied to the turbine at 200 + 15, - 0 psig." This SR requires a pump, valve, and flow test for the HPCI system at 200 psig + 15, - 0 psig. This test is required to ensure that the HPCI system is capable of performing the system's design basis safety function prior to increasing reactor pressure above the system minimum operating pressure. The HPCI system is provided to assure that the reactor core is adequately cooled to limit fuel temperature in the event of a small break in the nuclear system and loss of coolant accident which does not result in rapid depressurization of the reactor vessel. By increasing the refueling cycle length, the time interval between testing of the HPCI system at 200 psig would be increased. As required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI Inservice Testing (IST) program and TS section 4.5.1.b, the HPCI system is tested every 3 months to ensure that the required flow is developed. Although this test is conducted at a nominal reactor vessel pressure of 1000 psig, it would detect significant failures of the HPCI turbine or pump that would also be detected by conducting the 18 month TS surveillance test and that could lead to the failure of the HPCI system to perform its safety function. In addition, the HPCI system is one of the several ECCS, and as such is provided with redundant systems such as the Automatic Depressurization System (ADS) and the Low Pressure Coolant Injection (LPCI) system which will ensure a safe shutdown in the event of a HPCI system failure. Based on the above discussion, we have concluded that the impact of the proposed change, if any, on system availability is small.

b. TS SR 4.5.1.d.2.b requires that once every 18 months for the Automatic Depressurization System (ADS): "Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig and observing that either:

   1) the main turbine control valve or bypass valve position responds accordingly, or

   2) there is a corresponding change in the measured steam flow."

   This SR requires that the ADS valves, i.e., a subset of the Safety Relief Valves (SRVS), be lifted once every 18 months. This test is required to ensure that these valves are capable of lifting and performing their safety function which is to provide a back-up means to depressurize the reactor vessel to allow low pressure ECCS to provide coolant make-up to the reactor vessel. The testing interval was originally established as once per 18 months based on the equipment availability during the refueling outage. As stated above, the ADS valves together with low pressure ECCS, serve as a redundant systems to the HPCI system. We note also that the historical problems with corrosion-induced bonding of the pilot disc to the pilot seat that contributes to setpoint drift of the LGS Target Rock two-stage SRVS does not affect the ADS function of these valves. Finally, there are more valves provided in the system than required by the design analysis. Based on the redundant capability of the overall plant and the fact that the test scheduling was originally based on an outage opportunity rather than specific time-based requirements, we have concluded that the impact of the proposed change, if any, on system availability is small.

c. TS SR 4.7.3.c.2 requires that once every 18 months for the Reactor Core Isolation Cooling (RCIC) system; "Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, -0 psig." This test is to ensure the RCIC system is capable of performing the system's design function prior to increasing reactor pressure above the system's minimum operating pressure. The RCIC design function is to provide a means to ensure that sufficient reactor water inventory is maintained in the reactor vessel to ensure adequate core cooling. The system is designed to provide this capability in the event of: 1) isolation of the reactor vessel while maintaining the plant in Hot Standby, 2) reactor vessel isolation and loss of reactor feedwater, and 3) the start of a complete plant shutdown under conditions of loss of the normal feedwater system before the reactor is depressurized to a level at which the shutdown cooling mode of the Residual Heat Removal (RHR) system can be placed into operation. In addition to the 18 month 150 psig test, RCIC is also tested on a quarterly basis as required by TS SR 4.7.3.b and the ASME Section XI IST program. These quarterly tests, although required to be performed at a nominal reactor vessel pressure of 1000 psig, are designed to test the performance of the RCIC system and as such would detect significant failures of the RCIC turbine or pump that would also be detected by the 150 psig surveillance. Furthermore, the HPCI system will provide the same safety function. Based on the fact that the RCIC system is tested on a greater frequency than 18 months and plant design includes an alternate high pressure injection system, we have concluded that the impact of the proposed change, if any, on system availability is small.

A review of the surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusions.

(9) Emergency Service Water: TS SR 4.7.1.2.b.1; page 3/4 7-4
TS SR 4.7.1.2.b.2; page 3/4 7-4

a. TS SR 4.7.1.2.b.1 requires verifying, at least once every 18 months, that "Each Automatic valve actuates to its correct position on its appropriate ESW pump start signal." This test is required to verify that automatic actuation of the appropriate Emergency Service Water (ESW) system and Residual Heat Removal Service Water (RHRSW) system valves will occur during design accident conditions. The ESW system supplies cooling water to the Emergency Diesel Generators (EDGs) whenever they are operating, and to safety-related heat exchangers whenever the normal service water system is unavailable. During ESW system testing, the ESW pumps may be manually started or an ESW pump will start automatically any time its associated EDG starts. Upon start of each ESW pump, a signal is generated to ensure the alignment of that pump's associated ESW and RHRSW system valves. ESW component flow testing, and ASME Section XI Inservice Testing (IST) required Pump, Valve, and Flow testing will also test the subject valves.

   In addition, each EDG is tested monthly, which requires the ESW pump to autostart. The component flow testing, IST, and the EDG testing would detect any significant failures of valves not actuating as a result of an ESW pump start signal. The ESW system is comprised of two redundant loops. Either loop is sufficient to remove the design heat loads. Additionally, the systems cooled by the ESW System, i.e., Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) system heat exchangers and the EDGs, are redundant systems. Therefore, based on the above discussion, we have concluded that the impact of the proposed change, if any, on system availability is small.

b. TS SR 4.7.1.2.b.2 requires, once per 18 months, that "each ESW pump starts automatically when its associated Diesel Generator is started." Part of the acceptance criteria of the monthly EDG test is to verify that there is a pressure differential across the EDG heat exchangers to ensure there is ESW flow removing the necessary heat load. The monthly EDG test does not result in a manual start the ESW pumps, but instead requires manual start of the EDG and verifies the autostart of the associated ESW pump. In addition, there are temperature alarms on the EDG coolers that would identify failure of an ESW pump autostart. Additionally, there is redundancy built in to the ESW system so that any EDG can be supplied by any ESW pump if operator action is taken. Since the autostart function of the ESW pump is verified during the monthly EDG testing and there is redundancy built into the ESW system design, we have concluded that the proposed change in surveillance interval has no impact on system availability.

A review of surveillance test history demonstrated that there was no evidence of any failures that would invalidate the above conclusions.

(10) Halon System: TS SR 4.7.6.4.c.2; page 3/4 7-25

TS SR 4.7.6.4.c.2 requires the Remote Shutdown Panel Room 540 (raised floor) and the Auxiliary Equipment Room 542 (raised floor) Halon system piping be demonstrated OPERABLE at least once per 18 months by performing a system "flow" test to assure no blockage. The proposed change would require deleting SR 4.7.6.4.c.2 and replacing it with a new SR 4.7.6.4.d that would state "At least once per 24 months by performing a system flow test to assure no blockage."

The Halon system provides protection for the cables routed beneath the raised floor panels in the Remote Shutdown Panel (RSP) Room and the Auxiliary Equipment Room (AER). TS SR 4.7.6.4.c.2 requires a flow test of the Halon system piping be performed every 18 months. The Halon system flow test requires that the raised floor panels in the RSP Room and the AER be removed to gain access to the Halon system nozzles. Station personnel elect lifting the raised floor panels during an outage because unintentionally jarring or dropping a panel could cause an operational concern due to the sensitive electronic equipment in the area. This test is conducted by connecting a hose from the Service Air system and verifying airflow of each nozzle. This test assures that the piping is free of obstructions.

When the Halon systems were installed in the RSP room and the AER, the testing methodology for these systems was the performance of a full discharge test. A full discharge test was recommended by National Fire Protection Association (NFPA) Pamphlet 12A, "Halon 1301 Fire Extinguishing Systems," to verify the agent concentration, agent concentration hold time, and system functionality, and to verify that the system piping was unobstructed. A full discharge test was performed on all the LGS Halon systems by the Pre-Operational Test Program after installation. NFPA 12A no longer recommends full discharge testing be conducted because Halon and Freon 122 (i.e., a test gas) are classified as ozone-depleting substances that are destroying the stratospheric ozone. NFPA 12A has changed their testing requirements such that the piping be verified unobstructed by performing a system puff (i.e., flow) test after installation is completed. NFPA 12A does not require a system piping "flow" test be performed on a routine basis because the incidence of piping obstructions after a successful installation puff test is rare. Therefore, the proposed change to the testing interval for the Halon system piping would still exceed current requirements established by NFPA standards.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(11) Logic System Functional Test:

TS Table 4.3.2.1-1; Item 3.d; page 3/4 3-28

TS SR 4.3.1.2; page 3/4 3-1	TS SR 4.5.1.d.2.a; page 3/4 5-5
TS SR 4.3.2.2; page 3/4 3-10	TS SR 4.6.1.4.c.1; page 3/4 6-7
TS SR 4.3.3.2; page 3/4 3-32	TS SR 4.6.1.4.c.2; page 3/4 6-7
TS SR 4.3.4.1.2; page 3/4 3-42	TS SR 4.6.3.2; page 3/4 6-18
TS SR 4.3.4.2.2; page 3/4 3-47	TS SR 4.6.5.2.1.b; page 3/4 6-48
TS SR 4.3.5.2; page 3/4 3-52	TS SR 4.6.5.2.2.b; page 3/4 6-50
TS SR 4.3.9.2; page 3/4 3-112	TS SR 4.7.3.c.1; page 3/4 7-10
TS SR 4.5.1.c.1; page 3/4 5-4	TS SR 4.7.3.c.3; page 3/4 7-10
TS SR 4.5.1.c.2.b; page 3/4 5-5	TS SR 4.7.8.b; page 3/4 7-33

a. TS SR 4.3.1.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the Reactor Protection System (RPS) instrumentation. This testing is to confirm the ability of the RPS to perform its intended function. The RPS is provided to automatically initiate a reactor scram to preserve the integrity of the fuel cladding, preserve the integrity of the reactor coolant system, minimize the energy which must be absorbed following a LOCA, and prevent inadvertent criticality. The RPS is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will actuate that trip system. The actuation of both trip systems will result in a reactor scram. By increasing the refueling cycle length, the time interval between logic system functional tests and simulated automatic operation of all channels would be increased. Based on the inherent equipment reliability, as demonstrated by years of operating experience in the nuclear and non-nuclear industry, and the channel redundancy within the RPS design, we have concluded that the impact of the proposed change on system availability, if any, is small.

b. TS SR 4.3.2.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the containment Isolation Actuation instrumentation. Additionally, TS Table 4.3.2.1-1 Item 3.d identifies the Standby Liquid Control System (SLCS) Initiation trip function for Reactor Water Cleanup (RWCU) System Isolation shall be demonstrated operable by the performance of a Channel Functional Test at least once per 18 months. This testing is to ensure the effectiveness of the instrumentation used to mitigate the consequences of accidents through isolation of the reactor systems. The purpose of this system is to prevent the gross release of radioactive materials to the environment from the fuel or a break in the reactor coolant pressure boundary. SLCS initiation is designed to provide injection of a liquid neutron absorber into the reactor after the Redundant Reactivity Control System (RRCS) initiation. Isolation of the RWCU system on SLCS initiation precludes removal of the neutron absorber from the reactor water. By increasing the refueling cycle length, the time interval between logic system functional tests and simulated automatic operation of all channels of the Isolation Actuation instrumentation and SLCS initiation trip function for RWCU isolation would be increased. Based on the inherent equipment reliability, as demonstrated by years of operating experience in the nuclear and non-nuclear industry, and the channel redundancy within the isolation system design, we have concluded that the impact of the proposed change on system availability, if any, is small.

c. TS SR 4.3.3.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the Emergency Core Cooling System (ECCS) actuation instrumentation. This testing is to ensure the initiation of actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. The ECCS actuation instrumentation automatically initiates and controls the ECCS to prevent fuel cladding temperatures from reaching 2200°F and responds to a need for emergency core cooling regardless of the physical location of the malfunction or break that causes the need. By increasing the refueling cycle length, the time interval between logic system functional testing of the ECCS actuation instrumentation would be increased. Based on the inherent equipment reliability, as demonstrated by years of operating experience in the nuclear and non-nuclear industry, and channel redundancy within the ECCS design, we have concluded that the impact at the proposed change on system availability, if any, is small.

d. TS SR 4.3.4.1.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip (RPT) actuation instrumentation. This testing is to ensure an ATWS RPT mitigates the consequences of an ATWS event by tripping the recirculation pumps early in the event to reduce core flow and thereby reduce the core power generation. The ATWS RPT system trips the recirculation pump motors when either a turbine stop valve closure or turbine control valve fast closure occurs. By increasing the refueling cycle length, the time interval between logic system functional testing of the ATWS RPT actuation instrumentation would be increased. Based on the inherent equipment reliability, as demonstrated by years of operating experience in the nuclear and non-nuclear industry, and the channel redundancy within the ATWS RPT System design, we have concluded that the impact of the proposed change on system availability, if any, is small.

e. TS SR 4.3.4.2.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the End-Of-Cycle Recirculation Pump Trip (EOC-RPT) system instrumentation. The EOC-RPT is a supplement to the reactor trip. During the main turbine trip and generator load rejection events, the EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the coolant void collapse in the core during these two of the most limiting pressurization events. By increasing the refueling cycle length, the time interval between logic system functional testing of the EOC-RPT actuation instrumentation would be increased. Based on the inherent equipment reliability, as demonstrated by years of operating experience in the nuclear and non-nuclear industry, and the inherent channel redundancy within the EOC-RPT System design, we have concluded that the impact of the proposed change on system availability, if any, is small.

f. TS SR 4.3.5.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the Reactor Core Isolation Cooling (RCIC) system actuation instrumentation. This test is to ensure the actions required to ensure adequate core cooling in the event of the reactor isolation from its primary heat sink and loss of feedwater flow to the reactor vessel. The RCIC design function is to provide a means to ensure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. The system is initiated automatically after receiving a reactor vessel low water signal and produces a design flow rate within 30 seconds. The system then functions to provide makeup water flow to the reactor vessel, then automatically shuts down when the reactor vessel reaches a preset water level. By increasing the refueling cycle length, the time interval between logic system functional testing of the RCIC actuation instrumentation would be increased. Based on inherent equipment reliability, as demonstrated by years of operating experience in the nuclear and non-nuclear industry, and the design that includes the HPCI system which will provide this same function, we have concluded that the impact of the proposed change on system availability, if any, is small.

g. TS SR 4.3.9.2 requires that Logic System Functional Tests and simulated automatic operation of all channels shall be performed at least once per 18 months for the Feedwater/Main Turbine Trip System actuation instrumentation. This is to ensure the feedwater system/main turbine trip system functions as designed in the event of failure of the feedwater controller under maximum demand. By increasing the refueling cycle length, the time interval between logic system functional tests and simulated automatic operation of all channels of the feedwater/main turbine trip system actuation instrumentation would be increased. During normal plant operation, the feedwater control system automatically regulates feedwater flow into the reactor vessel. Feedwater flow is regulated by controlling the speed of the turbine-driven feedwater pumps that deliver the required flow to the reactor vessel. A feedwater control signal is produced from a level controller and manual/automatic transfer station whose output is a function of the level and flow errors in the system. Loss of the feedwater control signal to the feedwater pump turbine signal is alarmed in the feedwater control circuit and causes the turbine speed control system to lock the turbine speed "as is" and initiates an alarm in the control room. The feedwater control system has no safety function and is not required to operate after a design bases accident. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

h. TS SR 4.5.1.c.1 requires that the Emergency Core Cooling Systems (ECCS) shall be demonstrated operable at least once per 18 months. For the Core Spray (CS) system, the Low Pressure Coolant Injection (LPCI) system, and the High Pressure Coolant Injection (HPCI) system, this includes performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. This testing is to ensure the ECCS functions as designed by providing protection against postulated LOCAs caused by ruptures in primary system piping. The CS, LPCI, and HPCI systems are subsystems of the ECCS injection network. The CS system provides inventory makeup and core spray cooling during large primary system pipe breaks and inventory makeup following small primary system pipe breaks after ADS has been initiated. The LPCI system provides reactor vessel inventory makeup following large primary system pipe breaks and inventory makeup following small primary system pipe breaks after ADS has been initiated. The HPCI system maintains the reactor vessel inventory after small breaks which do not depressurize the reactor vessel. By increasing the refueling cycle length, the time interval between ECCS system functional tests would be increased. The ECCS network has built-in redundancy so that no single failure prevents the starting of sufficient ECCS to provide adequate coolant to the reactor vessel. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

i. TS SR 4.5.1.c.2.b requires that the ECCS shall be demonstrated operable at least once per 18 months. For the HPCI system, this includes verifying that HPCI pump suction is automatically transferred from the condensate storage tank (CST) to the suppression pool on a CST water level-low signal and on a suppression pool water level-high signal. The HPCI system is provided to ensure that the core is adequately cooled to limit fuel clad temperature in the event of a small break in the primary system piping and a LOCA which does not result in rapid depressurization of the reactor vessel. The HPCI system initially injects water from the CST. When the water level in the CST falls below a predetermined level or the suppression pool level is high, the pump suction is automatically transferred to the suppression pool. By increasing the refueling cycle length, the time interval between verification of automatic suction transfer of the HPCI system would be increased. The HPCI system is one of several ECCS, and as such is provided with redundant systems such as ADS and LPCI, which will ensure a safe shutdown in the event of a HPCI failure. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

j. TS SR 4.5.1.d.2.a requires that the ECCS shall be demonstrated operable at least once per 18 months. For the Automatic Depressurization System (ADS), this includes performing a system functional test that includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation. This testing ensures that the ADS performs its design safety function, which is to provide a means to depressurize the reactor vessel to allow low pressure ECCS to provide coolant makeup. By increasing the refueling cycle length, the time interval between system functional tests of the ADS would be increased. The ADS is independent of any other system of the ECCS. In the event that the RCIC system or HPCI system cannot maintain the reactor vessel water level, the ADS reduces the reactor pressure so that flow from the LPCI and/or the CS systems can be injected into the reactor vessel. The ADS employs selected safety relief valves (SRVs) for depressurization of the reactor and has two independent and redundant trip systems. The SRVs associated with the ADS are equipped with remote manual switches so that the entire system can be operated manually as well as automatically. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

k. TS SR 4.6.1.4.c.1 requires that each Main Steam Isolation Valve-Leakage Control System (MSIV-LCS) subsystem shall be demonstrated operable at least once per 18 months. This includes the performance of a functional test that includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each interlock and timer operates as designed, each automatic valve actuates to its correct position, and the blower starts. Additionally, TS SR 4.6.1.4.c.2 requires verifying that the blower(s) develops at least the required vacuum at the rated capacity specified below:

a) Inboard valves, 15" H_2O at 100 scfm.

b) Outboard valves, 15" H_2O at 200 scfm.

This testing is to ensure that the system prevents direct release of fission products to the atmosphere that could leak through closed MSIVs after a LOCA. Specifically, the MSIV-LCS (i.e., located upstream and downstream from outboard MSIVs) is designed to minimize the release of fission products that could bypass the Standby Gas Treatment System (SGTS) after a postulated LOCA. This is accomplished by directing the leakage through closed MSIVs to bleed lines that pass the leakage flow into an area served by the Reactor Enclosure Recirculation System (RERS). The flow is induced by a blower that maintains the pressure in the steam lines just slightly negative with respect to atmosphere, ensuring the MSIV leakage passes through the blower and on into the RERS and finally the SGTS before release to the atmosphere. The MSIV-LCS is manually initiated when the operator has determined the need for the system's operation based on high drywell pressure and low reactor water level, and the reactor vessel and main steam line pressure permissives are satisfied. By increasing the refueling cycle length, the time interval between system functional tests and verification of blower vacuum of the MSIV-LCS would be increased. By virtue of the two redundant systems (i.e., upstream and downstream from outboard MSIVs), and that the system is designed so that effects from a single active component failure will not affect the integrity or operability of the main steam lines or MSIVs, we have concluded that the impact of the proposed change on system availability, if any, is small.

l. TS SR 4.6.3.2 requires each primary containment automatic isolation valve shown in TS Table 3.6.3-1 be demonstrated operable during cold shutdown or refueling at least once per 18 months by verifying that on a containment isolation test signal, each automatic isolation valve actuates to its isolation position. This ensures the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. The purpose of the primary containment automatic isolation valves is to automatically isolate fluid lines that penetrate the primary containment in the event of postulated accidents to prevent or limit the release of radioactive materials. By increasing the refueling cycle length, the time interval between verification that the containment automatic isolation valves will actuate on a containment isolation test signal would be increased. The containment isolation system is provided with redundancy so that the active failure of any single valve or component does not prevent containment isolation. Also, to ensure valve operability and leak-tightness, periodic testing of the containment isolation system is performed during reactor operation. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

m. TS SR 4.6.5.2.1.b requires each reactor enclosure secondary containment ventilation system automatic isolation valve shown in TS Table 3.6.5.2.1-1 shall be demonstrated operable at least once per 18 months by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position. TS SR 4.6.5.2.2.b requires each refueling area secondary containment ventilation system automatic isolation valve shown in TS Table 3.6.5.2.2-1 shall be demonstrated operable at least once per 18 months by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position. These tests are required to ensure that the reactor enclosure and refueling secondary area containment are capable of performing their design safety function. The reactor enclosure and refueling area secondary containment are designed to minimize any ground level release of radioactive material which may result from an accident within the reactor enclosure or refueling area. By increasing the refueling cycle length, the time interval between verification that reactor enclosure and refueling area secondary containment ventilation system automatic isolation valves will actuate on a containment isolation test signal would be increased. The secondary containment isolation is an active safety-related function of the reactor enclosure and refueling area heating, ventilation, and air conditioning (HVAC) systems during normal operation. The isolation valves for secondary containment isolation are redundant (i.e., two in series) and fail closed. If an active failure disables one of the two valves, the other still performs the isolation function. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

n. TS SR 4.7.3.c.1 requires the RCIC system be demonstrated operable at least once per 18 months by performing a system functional test that includes simulated automatic actuation and restart, and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded. Additionally, TS SR 4.7.3.c.3 requires verifying the suction for the RCIC system is automatically transferred from the condensate storage tank (CST) to the suppression pool on a CST water level-low signal. The purpose of the RCIC system is to provide a means to ensure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. The system is designed to provide this capability in the event of: 1) isolation of the reactor vessel and maintaining the plant in Hot Standby, 2) reactor vessel isolated and loss of reactor feedwater, and 3) the start of a complete plant shutdown under conditions of loss of normal feedwater system before the reactor is depressurized to a level at which the shutdown cooling system can be placed into operation. By increasing the refueling cycle length, the time interval between system functional tests and verification of suction transfer of the RCIC system would be increased. The HPCI system will provide the same safety function as the RCIC system. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

o) TS SR 4.7.8.b requires that the Main Turbine Bypass System be demonstrated operable at least once per refueling cycle by performing a system functional test which includes simulated automatic actuation, and by verifying that each automatic bypass valve actuates to its correct position. The Main Turbine Bypass System is required to limit peak pressure in the main steam lines and to maintain reactor pressure within acceptable limits during events that cause rapid pressurization such that the safety limit Minimum Critical Power Ratio (MCPR) is not exceeded. Failure of the valves to open for any reason, such as a mechanical malfunction or insufficient vacuum in the condenser, causes the pressure in the reactor vessel to increase, ultimately lifting the safety relief valves. The test scheduling is based on an outage opportunity rather than specific time-based requirements because the performance of this surveillance test during power operation increases the chance of causing a reactor scram by challenging the RPS. The existing TS require each turbine bypass valve to be stroked through at least one complete cycle of full travel every 31 days. Therefore, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history for Items a. through n. demonstrated that there is no evidence of any failures which would invalidate the above conclusion. Surveillance testing of the Main Turbine Bypass System (i.e., Item o.) was recently added to the TS, and therefore, little or no plant specific test results are available. However, the lack of test history does not invalidate the conclusion that the impact of the proposed change on system availability, if any, is small.

(12) Manual Initiation:

TS Table 4.3.2.1-1; Items 1.h, 2.c, 3.f, 4.g, 5.g, 6.j, 7.g, and 7.h; pages 3/4 3-27 through 3/4 3-31
TS Table 4.3.3.1-1; Items 1.d, 2.e, 3.f, and 4.g; pages 3/4 3-40 and 3/4 3-41
TS Table 4.3.5.1-1; Item d.; page 3/4 3-56

a. TS Table 4.3.2.1-1 in conjunction with TS SR 4.3.2.1, requires each Isolation Actuation Instrumentation channel to be demonstrated operable by the performance of a Channel Functional Test at least once per 18 months. A Manual Initiation channel functional test of Main Steam Line Isolation (item 1.h), Residual Heat Removal (RHR) System Shutdown Cooling Mode Isolation (item 2.c), Reactor Water Cleanup (RWCU) System Isolation (item 3.f), High Pressure Coolant Injection (HPCI) System Isolation (item 4.g), Reactor Core Isolation Cooling (RCIC) System Isolation (item 5.g), Primary Containment Isolation (item 6.j), and Secondary Containment Isolation (items 7.g and 7.h) is required to satisfy this TS requirement to ensure the effectiveness of instrumentation used to mitigate the consequences of accidents by isolation of the reactor systems. By increasing the refueling cycle length, the time interval between channel functional testing would be increased. Each individual subsystem of the Isolation Actuation Instrumentation has a provision for its own manual actuation, and no single failure in the actuation portion of the subsystem will prevent a manual or automatic actuation of the other subsystems. The purpose of the Isolation system is to prevent the gross release of radioactive materials to the environment from the fuel or a break in the reactor coolant pressure boundary. The Isolation system automatically isolates the appropriate pipelines that penetrate primary containment whenever monitored variables exceed preselected setpoints.

b. TS Table 4.3.3.1-1 in conjunction with TS SR 4.3.3.1 requires each Emergency Core Cooling Systems (ECCS) Actuation Instrumentation channel to be demonstrated operable by the performance of a Channel Functional Test at least once per 18 months. A Manual Initiation channel functional test of the Core Spray (CS) System (item 1.d), the Low Pressure Coolant Injection (LPCI) Mode of the Residual Heat Removal (RHR) System (item 2.e), the High Pressure Coolant Injection (HPCI) System (item 3.f), and the Automatic Depressurization System (ADS) (item 4.g) is required to satisfy this TS requirement to ensure the actuation instrumentation provides actions to mitigate the consequences of accidents that are beyond the operators's control. By increasing the refueling cycle length, the time interval between channel testing would be increased. Each individual subsystem of the ECCS has a provision for its own manual initiation and no single failure in the initiation portion of the subsystems will prevent a manual or automatic initiation of redundant portions of the subsystems. The purpose of the ECCS is to ensure that the fuel is adequately cooled if there is a design basis accident. The monitoring instrumentation of the CS, LPCI Mode of RHR, HPCI systems, and ADS monitor and, if necessary, initiate the appropriate responses.

c. TS Table 4.3.5.1-1 in conjunction with TS SR 4.3.5.1 requires each Reactor Core Isolation Cooling (RCIC) Actuation Instrumentation channel to be demonstrated operable by the performance of a Channel Functional Test at least once per 18 months. A Manual Initiation channel functional test of the RCIC System (item d) is required to satisfy this TS requirement to ensure actuation instrumentation provides actions that ensure adequate core cooling in the event of reactor vessel isolation from its primary heat sink and the loss of feedwater flow. By increasing the refueling cycle length, the time interval between channel testing would be increased. The RCIC system is automatically initiated after receipt of a reactor vessel low water level signal. The RCIC system function is also provided by the HPCI System. The operator controls for the system are arranged to allow manual and remote manual operation which will ensure system availability/ operability in the event of a failure of one of the system components.

The Actuation Instrumentation Systems trip functions are automatically controlled by logic circuitry or manually controlled from the control room. The manual initiation trip function is only required to be channel functionally tested once per 18 months, but the automatic trip functions that are controlled by logic circuitry are channel functionally tested more frequently (i.e., a maximum duration of at least once per 92 days). This more frequent TS testing interval would detect the deterioration or malfunctic of equipment and also demonstrate the operational readiness of the systems. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(13) Reactor Recirculation Pump Motor-Generator (MG) Set Stop:

TS SR 4.4.1.1.2; page 3/4 4-2

TS SR 4.4.1.1.2 specifies that "Each Pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 109% and 107%, respectively, of rated core flow, at least once per 18 months." The purpose of this test is to ensure that the designed mechanical and electrical stops for the reactor recirculation pump MG set are verified to be functioning properly to ensure any speed transient is limited. The design of both the electrical and mechanical stops is such that they should not be susceptible to drift or degradation over time. This is based on the fact that the electrical stop is a mechanical device which employs a cam that actuates micro switches. This cam position is mechanically fixed to ensure no movement. The mechanical stop is a metal block which is bolted in place and will physically prevent the movement of the scoop tube beyond the previously established point. The design of both speed stops is such that they should not be susceptible to any time-based degradation. Accordingly, the impact of the proposed change on component function, if any, is small.

A review of the surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(14) Reactor Mode Switch Shutdown Function:

TS Table 4.3.1.1-1; Item 11, page 3/4 3-8
TS Table 4.3.6-1; Item 7, page 3/4 3-61

TS Table 4.3.1.1-1 in conjunction with TS SR 4.3.1.1 requires each reactor protection system (RPS) instrumentation channel be demonstrated operable by the performance of a Channel Functional Test at least once per 18 months. The reactor mode switch shutdown position (Item 11 of this TS table) is one functional unit of this channel functional test. TS Table 4.3.6-1 in conjunction with TS SR 4.3.6 requires each control rod block instrumentation (RBI) channel be demonstrated operable by the performance of a Channel Functional Test at least once per 18 months. The reactor mode switch shutdown position (Item 7 of this TS table) initiates a control rod block. This testing is to ensure the operability of the "Reactor Mode Switch in Shutdown RPS trip and Rod Block." By increasing the refueling cycle length, the time interval between channel functional testing would be increased. The manual positioning of the reactor mode switch is governed by the reactor startup (and shutdown) procedure and is the normal method for shutting down the reactor, which requires operator action for initiation. The RPS and RBI function automatically with various plant inputs. The reactor mode switch interfaces with these systems. Therefore, in the event of any undetected reactor mode switch failure, the RPS will continue to provide automatic scram capability. Based on this design, we have concluded that the impact of the proposed change on the RPS and RBI availability is small.

A review of surveillance test history demonstrated that there is no evidence of any failure which would invalidate the above conclusion.

(15) Primary Containment Hydrogen Recombiner System:

TS SR 4.6.6.1.b; Items 2, 3, and 4; page 3/4 6-57

a. TS SR 4.6.6.1.b.2 requires, at least once every 18 months, that the integrity of all heater electrical circuits be verified by performing a resistance-to-ground test within 30 minutes following the TS required functional test. This test is required to ensure that the heater electrical circuits have not been degraded to an unacceptable level of performance due to the high operating temperature achieved during the test. By increasing the refueling cycle length, the time interval between surveillance tests to check the heater electrical circuits would be increased. However, in addition to this 18 month TS surveillance test, TS SR 4.6.6.1.a requires that at least once per six (6) months; (a) a channel check of all main control room recombiner instrumentation, (b) a trickle heat circuit check, (c) heater coil check, and (d) a verification of valve operation by stroking all of the valves to their proper position be performed. Additionally, the hydrogen recombiner blower is operated to verify rated flow is attained. These six (6) month tests give added assurance that the system remains operable during power operation and will perform its safety function. When the hydrogen recombiner packages are in the "standby" mode, not being tested or otherwise operated, the trickle heaters are energized in order to keep the insulated enclosure warm. Also, there are two 100% capacity, redundant recombiners. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

b. TS SR 4.6.6.1.b.3 requires, at least once per 18 months, through visual verification, that there is no evidence of abnormal conditions within the hydrogen recombiner; i.e., loose wiring or structural connections, deposits of foreign material, etc. This verification is required to ensure that the recombiner is maintained in an operational condition. The present testing interval was established based on equipment availability during power operation. As stated above, other TS required hydrogen recombiner tests performed at least once per six (6) months will provide reasonable assurance that the equipment will remain operable. Also there are two 100% capacity, redundant recombiners. Based on redundant equipment and additional required testing, we have concluded that the impact of proposed change on system, availability if any, is small.

c. TS SR 4.6.6.1.b.4 requires, at least once per 18 months, during a hydrogen recombiner functional test, that the minimum heater outlet gas temperature increases to greater than or equal to 1150°F within 120 minutes and is maintained for at least an hour. This test is required to demonstrate that the recombiner can generate enough heat to ensure complete recombination of any hydrogen with oxygen, within the time frame assumed in the accident analysis in the Safety Analysis Report. As with the above TS sections, the other TS required hydrogen recombiner testing performed at least once per six (6) months gives added assurance that the recombiners will operate. These more frequent tests do not test the hydrogen recombiners for temperature output for the duration required by the current 18 month TS surveillance test, but because this equipment is operated only during required testing, the heating elements should not degrade during the periods between tests. Based on the discussion above we have concluded that the impact of the proposed change, on system availability, if any is small.

A review of the surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(16) Remote Shutdown System: TS SR 4.3.7.4.2; page 3/4 3-76

TS SR 4.3.7.4.2 requires, once every 18 months for the Remote Shutdown System, that "each of the remote shutdown control switch(es) and control circuits required by TS Table 3.3.7.4-1 shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s)." Combined with the instrumentation listed in TS Table 3.3.7.4-1, these components provide the ability to carry out reactor shutdown functions from outside the control room and bring the reactor to cold conditions in a safe and orderly fashion. The remote shutdown system is designed to control the required shutdown systems irrespective of shorts, opens, or grounds in the control room control circuits that may have resulted from an event causing an evacuation. The remote shutdown capability, by itself, does not perform any safety-related or protective function, and does not fall within the criteria set by IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." This system interfaces with safety-related systems, such as RHR and RCIC, and during normal operation, becomes part of and meets the design criteria for these systems. Administrative procedures control access to the remote shutdown panels, significantly reducing their exposure to physical wear and degradation. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(17) Reactor Enclosure HVAC System:

TS SR 4.6.5.4.b; Items 1, 2, and 3; page 3/4 6-55
TS SR 4.6.5.4.d; Items 1 and 2; page 3/4 6-56

TS SR 4.6.5.4.b.1 requires that at least once per 18 months for the Reactor Enclosure Recirculation System (RERS): "Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 60,000 cfm ± 10%." TS SR 4.6.5.4.b.2 requires that at least once per 18 months for the RERS system: "Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%." TS SR 4.6.5.4.b.3 requires that at least once per 18 months for the RERS system: "Verifying a subsystem flow rate of 60,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980." TS SR 4.6.5.4.d.1 requires that at least once per 18 months for the RERS system: "Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating a filter train at a flow rate of 60,000 cfm ± 10%, verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge." TS SR 4.6.5.4.d.2 requires that at least once per 18 months for the RERS system: "Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:

a) manual initiation from the control room, and
b) simulated automatic initiation signal."

These tests are required to ensure that the RERS system is capable of performing the system's design safety function. The RERS system filters reactor enclosure air following a design basis accident to reduce the concentration of radioactive halogens and particulates potentially present in the reactor enclosures. In accordance with TS SRs 4.6.5.4.b, 4.6.5.4.c, 4.6.5.4.e, and 4.6.5.4.f, the RERS is required to be tested following HEPA filter and charcoal adsorber structural maintenance, fire, chemical release, painting, HEPA filter replacement, charcoal adsorber replacement, and after 720 hours of operation. This additional testing would detect potential changes in HEPA filter efficiency and charcoal adsorber bypass leakage that would also be detected by conducting the 18 month TS surveillance tests. As required by TS SR 4.6.5.4.a, the RERS system is operated at least once per 31 days. This test would determine significant failures affecting flow or filter pressure drop that would also be detected by conducting the 18 month TS surveillance test. In addition, the RERS system active components and power supplies are designed with redundancy to meet the single active failure criterion, thereby ensuring system availability in the event of a failure of one of the system components. Based on the above discussion, and the fact that the RERS system is normally in standby, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

In addition, TS SRs 4.6.5.4.b.1 and 4.6.5.4.b.2 would be changed to indicate that a change to 24 month testing would be an exception to the 18 month testing interval guidance specified in Regulatory Guide 1.52, Revision 2, March 1978.

(18) Response Time: TS SR 4.3.1.3; page 3/4 3-1
TS SR 4.3.2.3; page 3/4 3-10
TS SR 4.3.3.?; page 3/4 3-32
TS SR 4.3.4.?.3; page 3/4 3-47
TS SR 4.7.8.c; page 3/4 7-33

a. TS SR 4.3.1.3 requires the Reactor Protection System (RPS) response time of each reactor trip functional unit shown in TS Table 3.3.1-2 be demonstrated to be within its limit at least one per 18 months. Additionally, each test shall include at least one channel per trip system such that all channels are tested at least once every 'N' times 18 months where 'N' is the total number of redundant channels in a specific reactor trip system. This SR requires response time testing for the following RPS reactor trip functional units: Average Power Range Monitor Neutron Flux - Upscale, Reactor Vessel Steam Dome Pressure-High, Reactor Vessel Water Level Low - Level 3, Main Steam Isolation Valve Closure, Turbine Stop Valve Closure, and Turbine Control Valve Fast Closure-Trip Oil Pressure Low. RPS response time testing is required to provide assurance that the protective functions associated with each RPS functional unit channel are completed within the time limit assumed in the safety analyses. The RPS is provided to automatically initiate a reactor scram to preserve the integrity of the fuel cladding and the reactor coolant system, minimize the energy which must be adsorbed following a LOCA, and prevent inadvertent criticality. The RPS is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will actuate that trip system. The actuation of both trip systems will produce a reactor scram. By increasing the refueling cycle length, the time interval between testing of the RPS response time would be increased. Based on the inherent equipment reliability and channel redundancy within the RPS design, we have concluded that the impact of the proposed change on system availability, if any, is small.

b. TS SR 4.3.2.3 requires the Isolation System instrumentation response time of each isolation trip function shown in TS Table 3.3.2-3 be demonstrated to be within its limit at least once per 18 months. The response time given for all listed trip functions, except RWCU system differential flow high and MSIV isolation, includes 10 seconds for Emergency Diesel Generator (EDG) starting and three (3) seconds for sequence loading delays. Additionally, each test shall include at least one channel per trip system such that all channels are tested at least once every 'N' times 18 months, where 'N' is the total number of redundant channels in a specific isolation trip system. This SR requires response time testing for EDG starts and the following Isolation System trip functional units: 1. Main Steam Line Isolation - 1.a) Reactor Vessel Water Level Low, Low Level 2 and Low, Low, Low Level 1, 1.b) Main Steam Line Radiation High, 1.c) Main Steam Line Pressure Low, 1.d) Main Steam Line Flow High; 2. RHR System Shutdown Cooling Mode Isolation - 2.a) Reactor Vessel Water Level Low, Level 3; 3. Reactor Water Cleanup System Isolation - 3.a) RWCU Differential Flow High, 3.e) Reactor Vessel Water Level Low, Low, Level 2; 4. High Pressure Coolant Injection System Isolation - 4.a) HPCI Steam Line Differential Pressure High, 4.b) HPCI Steam Supply Pressure Low; 5. Reactor Core Isolation Cooling System Isolation - 5.a) RCIC Steam Line Differential Pressure High, 5.b) RCIC Steam Supply Pressure Low; 6. Primary Containment Isolation - 6.a) Reactor Vessel Water Level Low, Low Level 2 and Low, Low Level 1, 6.b) Drywell Pressure High. The purpose of these instrument response times is to prevent the gross release of radioactive materials to the environment from the fuel or a break in the reactor coolant pressure boundary. By increasing the refueling cycle length, the time interval between testing of the Isolation System instrumentation response time would be increased. Based on the inherent equipment reliability and channel redundancy within the Isolation System Instrumentation design, we have concluded that the impact of the proposed change on system availability, if any, is small.

c. TS SR 4.3.3.3 requires the Emergency Core Cooling Systems (ECCS) response time of each ECCS trip function shown in TS Table 3.3.3-3 be demonstrated to be within the limit at least once per 18 months. Additionally, each test shall include at least one channel per trip system such that all channels are tested at least once every 'N' times 18 months, where 'N' is the total number of redundant channels in a specific ECCS system. This SR requires response time testing for the following ECCS trip functions: Core Spray System, Low Pressure Coolant Injection Mode of RHR System, and High Pressure Coolant Injection System. The ECCS actuation instrumentation automatically initiates and controls the ECCS to prevent fuel cladding temperatures from reaching 2200°F and responds to a need for emergency core cooling, regardless of the physical location of the malfunction or break that causes the need. By increasing the refueling cycle length, the time interval between testing of the ECCS actuation instrumentation response time would be increased. Based on the inherent equipment reliability and channel redundancy within the ECCS design, we have concluded that the impact of the proposed change on system availability, if any, is small.

d. TS SR 4.3.4.2.3 requires the End-Of-Cycle Recirculation Pump Trip (EOC-RPT) system response time of each trip function shown in TS Table 3.3.4.2-3 be demonstrated to be within its limit at least once per 18 months. Additionally, each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The measured time shall be added to the most recent breaker arc suppression time and the resulting EOC-RPT system response time shall be verified to be within its limit. In addition to the change from "18 months" to "24 months," TS SR 4.3.4.2.3 would also be revised such that both types of channel inputs would be tested "at least once per 48 months" instead of "at least once per 36 months" to coincide with the proposed change to 24 month testing. TS SR 4.3.4.2.3 requires response time testing of the following EOC-RPT functions: Turbine Stop Valve Closure and Turbine Control Valve Fast Closure. The EOC-RPT system is a supplement to the reactor trip. During main turbine trip and generator load rejection events, the EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the coolant void collapse in the core during these two of the most limiting pressurization events. By increasing the refueling cycle length, the time interval between testing of the EOC-RPT System response time would be increased. Based on the inherent equipment reliability and channel redundancy within the EOC-RPT System design, we have concluded that the impact of the proposed change on system availability, if any, is small.

e. TS SR 4.7.8.C requires the Main Turbine Bypass System response time be demonstrated to be less than or equal to the value specified in the Core Operating Limits Report at least once per refueling cycle. The Main Turbine Bypass System is required to limit peak pressure in the main steam line and to maintain reactor pressure within acceptable limits during events that cause rapid pressurization such that the safety limit Minimum Critical Power Ratio (MCPR) is not exceeded. Failure of the valves to open for any reason, such as a mechanical malfunction or insufficient vacuum in the condenser, causes the pressure in the reactor to increase, ultimately lifting the safety relief valves. The Core Operating

Limits Report provides the response time limits for the Main Turbine Bypass System and is issued for each fuel cycle. The response time limits are based on operability requirements assumed in the Feedwater Controller Failure analysis in the Cycle Specific Transient Analysis. Performing the surveillance tests during the refueling outage insures TS requirements are met for each new fuel cycle. Performance of this surveillance test during power operation would increase the chance of causing a reactor scram by challenging the Reactor Protection System (RPS). Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history was conducted. Equipment/components utilized in the design of the Response Time group systems were chosen based on reliability as demonstrated by years of service in both the nuclear and non-nuclear industry. The original surveillance test intervals were based, in part, on this inherent reliability. The review of Response Time group surveillance test history was performed to detect evidence of time-based equipment/component failures. Should time-based failure modes exist, multiple equipment/components could fail during this longer surveillance test interval, possibly reducing the reliability/redundancy of the subject systems. This review accounted for the information in NRC Bulletin No. 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount." This Bulletin describes a possible time-based failure mode that could effect Rosemount transmitter response times; however, more frequent testing plus instrument trending of the affected transmitters precludes this condition from affecting the above conclusions. The surveillance test history was analyzed as a group, without regard to the specific Response Time group TS requirements due to the similarity in the trip circuitry and components used. Certain of the failures would be discovered during more frequent (i.e., quarterly) testing.

The review of surveillance test history for Items a. through d. demonstrated that there is no evidence of any failures which would invalidate the above conclusion. Additionally, surveillance testing of the Main Turbine Bypass System (i.e., Item c.) was recently added to the TS. Only one Unit 1 surveillance test has been performed, with satisfactory results, during a forced outage. However, the lack of test history does not invalidate the conclusion that the impact of the proposed change on system availability, if any, is small.

(19) Standby Liquid Control: TS SR 4.1.5.d; Items 1, 2, and 3; page 3/4 1-20

a. TS SR 4.1.5.d.1 requires, at least once per 18 months, "Initiating at least one of the Standby liquid control system loops,....by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. All injection loops shall be tested in 3 operating cycles." This requirement is to ensure the operability of the Standby Liquid Control system (SLCS), which is an independent redundant method to the control rods to establish and maintain the reactor subcritical. By increasing the refueling cycle, the time interval between testing of the SLCS would be increased. This test verifies the operation of the SLCS pumps, the injection valves, and the alarms in the control room which verify the system is operating or operable. Functional testing of the SLCS pump is performed on a quarterly basis throughout the operating cycle. This is accomplished by recirculating demineralized water from the test tank. The SLCS is equipped with three independent explosive-actuated injection valves. These valves have a high firing reliability and the charges are monitored for continuity in the control room. A loss of continuity in the circuits will result in an alarm in the control room. At LGS, the SLCS loops are independent of each other, and only two loops are required to be operable to meet the requirements of the TS. This feature provides redundancy in the system in that the plant can operate with one loop out of service. Therefore, increasing the length of the refueling cycle will have minimal impact on the availability of the SLCS system.

b. TS SR 4.1.5.d.2 requires, at least once per 18 months, "Demonstrating that all heat traced piping is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water." This test is performed to verify that the heat tracing is preventing sodium pentaborate precipitation in the piping between the SLCS storage tank and pump inlet, therefore, verifying the pump inlet is unblocked. On a daily basis, while operating in Operational Conditions 1 through 5, the temperature of the SLCS storage tank and pump piping suction temperature is recorded, thereby providing an adequate means of identifying blockages in the pipe due to sodium pentaborate precipitation. Therefore, increasing the length of the refueling cycle will have minimal impact on the availability of the SLCS system.

c. TS SR 4.1.5.d.3 which requires, at least once per 18 months, "Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized." The heaters are a backup heat source which maintains the solution temperature at 75°F to 85°F, to prevent precipitation of the sodium pentaborate from the solution. The SLCS tank contains two electric heaters. Heater "A" is used to maintain the solution temperature in "Auto" or "Manual" modes while heater "B" is used in manual only during solution mixing. Heater "A" is only needed to be operable to maintain the SLCS system as operable. Heater "B" is not needed during normal operation. Heater "A" will automatically initiate in the unlikely event that the tank solution drops below its low temperature setpoint of 75°F. Additionally, a low tank temperature alarm (i.e., 70°F) would alert the operators in the unlikely event that temperatures dropped below the solution setpoint and heater "A" failed to operate properly. The operators would then be able to manually restore the "B" heater, as required, to maintain solution temperature. Also, while in Operational Conditions 1 through 5, daily temperature readings are monitored for the solution tank. Therefore, increasing the length of the refueling cycle will have a minimal impact on the availability of the SLCS.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(20) Secondary Containment:

TS SR 4.6.5.1.1.c; Items 1 and 2; page 3/4 6-46
TS SR 4.6.5.1.2.c; page 3/4 6-47
Bases 3/4.6.5; page B 3/4 6-5

a. TS SR 4.6.5.1.1.c.1 requires that at least once per 18 months for the Reactor Enclosure Secondary Containment Integrity: "Verifying that one standby gas treatment subsystem will draw down the reactor enclosure secondary containment to greater n or equal to 0.25 inch of vacuum water gauge in less than or ual to 121 seconds with the reactor enclosure recirculation system in operation." TS SR 4.6.5.1.1.c.2 requires that at least once per 18 months for the Reactor Enclosure Secondary Containment Integrity: "Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the reactor enclosure secondary containment at a flow rate not exceeding 1250 cfm with wind speeds of ≤ 7.0 mph as measured on the wind instrument on Tower 1 elevation 30' or, if that instrument is unavailable, Tower 2, elevation 159'." In addition, TS Bases 3/4.6.5 states "Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment." These tests are required to ensure that the reactor enclosure secondary containment is capable of performing its design safety function. The reactor enclosure secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident within the reactor enclosure. The additional testing required by TS SRs 4.6.5.1.1.a, 4.6.5.1.1.b, 4.6.5.3.a, and 4.6.5.3.g would determine significant failures effecting the reactor enclosure secondary containment that would be detected by conducting the 18 month TS surveillance test. Based on the above discussion, we have concluded that the impact of the proposed change on reactor enclosure secondary containment availability, if any, is small.

b. TS SR 4.6.5.1.2.c requires that at least once per 18 months for the Refueling Area Secondary Containment Integrity: "Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm." This test is required to ensure that the refueling area secondary containment is capable of minimizing any ground level release of radioactive material which may result from an accident within the refueling area. Refueling area secondary containment is required when 1) irradiated fuel is being handled in the refueling area secondary containment, or 2) during core alterations, or 3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel. The additional tests required by TS SRs 4.6.5.1.2.a, 4.6.5.1.2.b, 4.6.5.3.a, and 4.6.5.3.g would determine significant failures effecting the refueling area secondary containment that would be detected by conducting the 18 month TS surveillance test. Based on the above discussion and the fact that Refueling Area Secondary Containment Integrity will continue to be verified prior to refueling operations, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of the surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

21) Standby Gas Treatment System:

TS SR 4.6.5.3.b; Items 1, 2, 3, and 4; page 3/4 6-53
TS SR 4.6.5.3.d; Items 1, 2, and 3; pages 3/4 6-53 and 3/4 6-54

TS SR 4.6.5.3.b.1 requires that at least once per 18 months for the Standby Gas Treatment System (SGTS): "Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 3000 cfm ± 10%." TS SR 4.6.5.3.b.2 requires that at least once per 18 months for the SGTS system: "Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%." TS SR 4.6.5.3.b.3 requires that at least once per 18 months for the SGTS system: "Verify that when the fan is running the subsystem flow rate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980." TS SR 4.6.5.3.b.4 requires that at least once per 18 months for the SGTS system: "Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gauge while operating at a flow rate of 2400 cfm ± 10%." TS SR 4.6.5.3.d.1 requires that at least once per 18 months for the SGTS system: "Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm ± 10%." TS SR 4.6.5.3.d.2 requires that at least once per 18 months for the SGTS system: "Verifying that the fan starts and isolation valves necessary to draw a suction from the refueling area or the reactor enclosure recirculation discharge open on each of the following test signals:

a) Manual initiation from the control room, and
b) simulated automatic initiation signal."

TS SR 4.6.5.3.d.3 requires that at least once per 18 months for the SGTS system: "Verifying that the temperature differential across each heater is ≥ 15°F when tested in accordance with ANSI N510-1980."

These tests are required to ensure that the Standby Gas Treatment System (SGTS) system is capable of performing the system's design safety function. The SGTS system filters radioactive particulates and both radioactive and nonradioactive forms of iodine from the air exhausted from the reactor enclosure and/or refueling area to maintain a negative pressure during secondary containment isolation following a postulated accident or abnormal occurrence which could result in abnormally high airborne radiation in the secondary containment. A prefilter is provided in the SGTS duct for air drawn from the refueling area during refueling area isolation. This prefilter normally does not have air flow through it. As required by TS SR 4.6.5.3.a, the SGTS system is demonstrated operable by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates with operable heaters at least once per 31 days. Heaters are used to limit relative humidity of 70% at the charcoal adsorbers. This test would determine significant failures effecting filter pressure drop or heater operability that would be detected by conducting the 18 month TS surveillance test. Additional testing of the SGTS is required by TS SRs 4.6.5.3.b, 4.6.5.3.c, 4.6.5.3.e, 4.6.5.3.f, and 4.6.5.3.g following filter structural maintenance, fire, chemical release, painting, HEPA filter replacement, charcoal adsorber replacement, and after 720 hours of operation. This testing would detect potential changes in HEPA filter efficiency, charcoal adsorber bypass leakage, and refueling prefilter pressure drop that would be detected by conducting the 18 month TS surveillance tests. The SGTS system is normally in standby. In addition, the SGTS system active components and power supplies are designed with redundancy to meet the single active failure criterion, which will ensure system availability in the event of a failure of one of the system components. Based on the above discussion, and that the SGTS system is normally in standby, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

Additionally, TS SRs 4.6.5.3.b.1 and 4.6.5.3.b.2 would be changed to indicate that the 24 month testing interval would be an exception to the 18 month testing interval guidance specified in Regulatory Guide 1.52, Revision 2, March 1978.

(22) Thermal: TS SR 4.8.4.2.2; page 3/4 8-27

TS SR 4.8.4.2.2 requires that once every 18 months: "A channel functional test of all those valves which are bypassed only under accident conditions (valves with spring-return-to-normal control switches) shall be performed to verify that the thermal overload protection will be bypassed under accident conditions." This SR requires a channel functional test for all Class 1E motor operated valves (MOVs). The channel functional surveillance test frequency is required at least every 18 months to ensure that the thermal overload protection on all MOVs, that are bypassed only under accident conditions (i.e., valves with spring-return-to-normal control switches), will be bypassed under accident conditions. The bypassing of the MOVs thermal overload protection continuously by integral bypass devices ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The TS SRs for demonstrating the bypassing of the thermal overload protection continuously are met by functionally testing the automatic operation of the MOV and ensuring that the motor thermal overload protection design does not change and is in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977. By increasing the refueling cycle length, the time interval between channel functional testing would be increased; however, a review of a typical thermal overload bypass circuit indicates that the primary operating component is a spring return-to-normal control switch. Based on the above discussion, we have concluded that the impact of the proposed change on system availability, if any, is small.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusion.

(23) Valves: TS SR 4.4.3.2.2.a; page 3/4 4-10
TS SR 4.6.3.4; page 3/4 6-18
TS SR 4.6.3.5.b; page 3/4 6-18

a. TS SR 4.4.3.2.2.a requires: "Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit at least once per 18 months." The pressure isolation valves (PIVs) provide the interface for low pressure systems with the reactor coolant system. Each containment penetration has redundant valves to assure isolation of the low pressure portion of the system from reactor pressure. In addition, the low pressure system is provided with pressure relief capability, and pressure is continuously monitored with alarms in the control room on high pressure. Leakage through these valves is maintained within acceptable limits to reduce the probability of gross valve failure and a consequent inter-system Loss of Coolant Accident (LOCA). Based on the above discussion, we have concluded that the proposed change to the surveillance frequency would have a negligible impact on the ability of the PIVs to function as designed.

b. TS SR 4.6.3.4 requires: "Each instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow." Excess flow check valves are installed outside primary containment on instrument lines which penetrate primary containment to minimize leakage in the event of an instrument line failure outside primary containment. A line failure downstream of the excess flow check valve would result in line isolation. In the event of a line failure between the source and the excess flow check valve, there is a restricting orifice inside primary containment as close to the source as practical to assure minimization of leakage in the event of a line failure. The excess flow check valves are equipped with position indicating switches which energize lights in the reactor enclosure. Based on the above discussion, we have concluded that the proposed change to the surveillance frequency will have a negligible impact on the ability of the excess flow check valves to function as designed.

c. TS SR 4.6.3.5.b requires: "Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE at least once per 18 months by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of the batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable." In addition to the proposed change from "18 months" to "24 months" for testing individual explosive squibs, TS SR 4.6.3.5.b would be revised to require testing all the TIP explosive squibs "at least once per 120 months" in place of "at least once per 90 months" to coincide with the proposed change to 24 month testing.

The Traversing In-Core Probe (TIP) shear valve is provided as a back-up isolation device for the TIP guide tube isolation valve. The isolation valve closes automatically upon receipt of a containment isolation signal after the TIP cable has been retraced. The shear valve can isolate the line if the TIP cannot be retracted or if the automatic isolation valve fails to close. The shear valves are explosive type valves designed to shear the cable and seal the guide tube. Actuation is by operator action from the control room. Continuity of the TIP shear valve squib firing circuits is continuously monitored in the control room to provide additional assurance that the TIP shear valves will operate as designed. In accordance with the ASME Section XI Inservice Testing (IST) Program, the automatic isolation valve is given a full stroke exercise test, a fail safe test, and a stroke time test quarterly to verify its operability as the main isolation valve for these lines. Also, the proposed increase in the explosive charge testing interval would still comply with the IST requirements of ASME Section XI. Specifically ASME Section XI, paragraph IWV-3610 requires testing of the explosive charges in at least 20% of the explosive valves every two years, and that charges shall not be older than 10 years. Based on the above discussion, we have concluded that the proposed change to the surveillance frequency will have a negligible impact on the ability of the TIP shear valve explosive charges to function as designed.

A review of surveillance test history demonstrated that there is no evidence of any failures which would invalidate the above conclusions.

<u>Safety Assessment Summary</u>

The proposed TS changes involve a change in the surveillance testing intervals from 18 months to 24 months to facilitate the current change in the LGS Unit 1 and Unit 2 refueling cycles to 24 months. The proposed changes are to the surveillance frequencies only, and do not involve a change to the TS surveillance requirements themselves or the way in which the surveillances are performed. Additionally, the impact of the proposed TS changes on the availability of equipment or systems required to mitigate the consequences of an accident, if any, is small based on other, more frequent testing or the availability of redundant systems or equipment. A review of surveillance test history demonstrated that there was no evidence of any failures that would invalidate the above conclusions.

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed changes to the LGS TS, to facilitate a change from 18 month to 24 month refueling cycles, do not constitute a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

1. The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed TS changes involve a change in the surveillance testing intervals to facilitate the current change in the LGS Unit 1 and Unit 2 refueling cycles from 18 months to 24 months. The proposed TS changes do no physically impact the plant nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS changes do not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed TS changes do not impact the TS surveillance requirements themselves, other than the frequency, nor the way in which the surveillance are performed. Additionally, the proposed TS changes do not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the change in the frequency of surveillance testing. Also, the proposed TS changes do not affect the availability of equipment or systems required to mitigate the consequences of an accident because of other, more frequent testing and/or the availability of redundant systems or equipment. Furthermore, an historical review of surveillance test results indicated that any failures identified were unique, non-repetitive, and not related to any time-based failure modes, and that there was no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed TS changes do not increase the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed TS changes involve a change in the surveillance testing intervals to facilitate the current change in the LGS Unit 1 and Unit 2 refueling cycles from 18 months to 24 months. The proposed TS changes do not introduce nor increase the number of failure mechanisms of a new or different type than those previously evaluated since there are no physical changes being made to the facility. Additionally, the surveillance test requirements themselves, other than the frequency, and the way surveillance tests are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated there was no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

Although the proposed TS changes will result in an increase in the interval between surveillance tests, the impact on system availability, if any, is small based on other, more frequent testing or redundant systems or equipment. Furthermore, a review of surveillance test history demonstrated that there is no evidence of any failures that would impact the availability of the systems. Accordingly the assumptions in the plant licensing basis are not impacted, and therefore the proposed TS changes do not reduce the margin of safety of the affected equipment/components.

Information Supporting an Environmental Assessment

An environmental assessment is not required for the changes proposed by this Change Request because the requested changes conform to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The requested changes will have no impact on the environment. The requested changes do not involve a significant hazards consideration as discussed in the preceding section. The requested changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed these proposed changes to the TS and have concluded that they do not involve an unreviewed safety question, or a significant hazards consideration, and will not endanger the health and safety of the public.