failure mode analysis, and consultation with the respective system engineer. The evaluations conclude that the subject SSCs are highly reliable, that presently do not exhibit time dependent failure modes of significance, and that there is no indication that the proposed extension could cause deterioration in the condition or performance of the subject SSCs. There are no known mechanisms that would significantly degrade the performance of the evaluated equipment during normal plant operation. Although there have been generic or repetitive failures of some components in the past, which may have affected the ability of the SSCs to consistently and successfully perform their safety function, those items have been resolved through design changes and rework such that they have not recurred. There have been no repetitive failures or time dependent failures that were significant in nature which would have prevented the SSCs from performing their intended safety function.

Deletion of the restriction ''during effect on safe operation of the plant is given prior to conduct of a particular surveillance in a condition or mode other than shutdown.

Since the proposed changes only affect the surveillance intervals for SSCs that are used to mitigate accidents [sic], the changes do not affect the probability or consequence of a previously analyzed accident. While the proposed changes will lengthen the intervals between surveillances, the increase in intervals has been evaluated. Based on the reviews of the surveillance tests, inspections, and maintenance activities, it is concluded that there is no significant adverse impact on the reliability or availability of these SSCs.

Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed changes. The proposed changes have no adverse impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements.

From the evaluations performed on the subject SSCs there are no indications that potential problems would be cycle-length dependent or that potential degradation would be significant for the time frame of interest and, therefore, increasing the surveillance interval to the bounding limit of 30 months (24 months plus 25%) will have little, if any, adverse affect on safety.

The proposed changes to the surveillance intervals are still consistent with the basis for the intervals and the intent and method of performing the surveillance is unchanged. Deletion of the restriction ''during shutdown'' where this restriction is stated will permit performance of certain maintenance and testing activities during conditions or modes other than shutdown. North Atlantic will ensure, through the implementation of appropriate administrative controls, that proper regard to their effect on safe operation of the plant is given prior to conduct of a particular surveillance in a condition or mode other than shutdown. In addition, use of the subject SSCs during normal plant operation, combined with their previous history of availability and reliability, provide assurance that the proposed changes will not affect the reliability of the subject SSCs. Thus, it is concluded that the subject SSCs would be available upon demand to mitigate the consequences of an accident and, therefore, there is no impact on the margin of safety.

The NRC staff has reviewed the licensee's analysis, and based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Cecil O. Thomas.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: July 2, 1998.

Description of amendment request: The proposed amendment would revise the updated Final Safety Analysis Report (FSAR) by changing FSAR Sections 9.7.2, ''Service Water,'' and 9.4, ''Reactor Building Closed Cooling Water,'' to discuss the use of various types of internal protective coatings and liners used in the piping and components of the systems. The proposed change also indicates that periodic maintenance, surveillances, and inspections would be conducted to ensure that coating or liner degradation would be promptly detected and corrected to provide reasonable assurance that the systems can perform their safety-related functions.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change does not involve significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The SWS [Service Water System] provides cooling water directly or indirectly to a multitude of mitigating and support systems such as safety injection, containment spray, and RBCCW [Reactor Building Closed-Cooling Water]. Therefore either directly or indirectly, the SWS is credited in the mitigation of virtually all analyzed operating events and accidents. However, there are no failures of the SWS which would directly initiate any of the licensing basis accidents. Therefore, the probability of occurrence of accidents previously evaluated is not increased by this activity.

The SWS is comprised of two separate and independent trains, each capable of providing the cooling capacity required for normal and accident operation. Therefore, the failure of a single heat exchanger or train will not influence the consequences of an accident. Only a common mode loss of SWS function could affect accident consequences. It can be postulated that lining material could be released as a result of the SWS response to an accident or as a result of a seismic event, resulting in heat exchanger blockage in both trains (common mode). However, the discussion below provides the basis for concluding that lining degradation will not increase the consequences of an accident.

In response to a Safety Injection Actuation Signal or a Loss of Normal Power event, the quantity of flow in safety related SWS heat exchangers may increase significantly, imparting higher loads on the pipe linings than are typically present during normal operation. In spite of this flow increase, it is considered to be much more likely that any lining degradation will occur and be detected under normal operating conditions, and will be corrected prior to the occurrence of an event of the type discussed above. SWS pump flow surveillances, performed periodically during normal operation, subject significant portions of the SWS to flow levels which equal or exceed those expected to occur during accidents. Any degraded lining material prone to be released during an accident is expected to be released during these pump surveillances. The inspections, operating procedures, and surveillances ensure that significant lining releases will be promptly detected and investigated. In addition, SWS design features provide the system with a significant level of protection against degraded lining debris (e.g., standby spare RBCCW heat exchanger and EDG [Emergency Diesel Generator] engine cooler strainers) both during normal operation and while responding to an accident.

An evaluation was performed to assess the significance of loading on the linings due to a postulated seismic event. The importance of seismic loads depends upon their magnitude relative to normal operating loads, and on their relative frequency of occurrence. Normal operating loads include steady state flow loads as well as transients due to pump swaps and realignments for surveillances. The evaluation determined that normal operating loads are significantly greater than anticipated seismic loads concurrent with steady state flow loads. Therefore, if normal operating loads do not cause lining to become detached, it is very unlikely that a random seismic event would cause detachment. In addition, while flow loads are continuously present in most of the system and normal transients occur many times during an operating cycle, seismic events at the Millstone site are very infrequent (the repetition rate of an OBE [Operating Basis Earthquake] is hundred of years). Should normal operating loads cause lining detachment, it is much more probable that this released material will be detected, and the degraded condition corrected, prior to the occurrence of a seismic event.

Based upon these discussions, and given the random nature of lining degradation and the scrutiny with which the SWS is operated and maintained, it is not considered to be credible that the operability of both SWS trains will be simultaneously impaired by lining degradation and release.

Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

As discussed above, the failure of a single heat exchanger or a single SWS train will not cause an accident. Only a common mode loss of SWS function could create the possibility of a previously unanalyzed accident, and this loss would not directly initiate an accident. However, for the reasons discussed above, lining degradatiion will not cause common mode failures to occur.

Therefore, the change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The margins of safety of the protective boundaries (fuel matrix/cladding, reactor coolant system pressure boundary, and containment) would not be impacted by the postulated release of lining material into the SWS. The accident analyses in the FSAR [Final Safety Analysis Report] demonstrate the performance of the protective boundaries. As discussed previously, it is not considered to be credible that lining degradation will cause a common mode loss of SWS function. Therefore, since the accident analyses credit only one SWS train, released lining would not affect accident analyses assumptions. On this basis, it is concluded that margins of safety as demonstrated by the accident analyses would not be affected by postulated lining material release.

Therefore, the change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Deputy Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: July 17, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) surveillance requirements for the onsite emergency diesel generators (EDGs) to achieve an overall improvement in the EDGs reliability and availability. The proposed changes would modify the requirement for operability tests of an EDG when the other EDG is inoperable, delete the requirement for operability tests when one or both offsite A.C. sources are inoperable, eliminate fast loading of the EDGs except for the 18-month testing, and eliminate fast starts (15 seconds) except for once per 6 months and during the 18-month testing. These proposed changes are generally consistent with the guidance provided in Generic Letter (GL) 84–15, ''Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,'' dated July 2, 1984, and GL 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993. Justification for deviations from the guidance provided in the GLs is provided in the licensee's submittal.

In addition, the licensee proposes to revise the wording in the TS requirements for offsite circuits to be consistent with NUREG–0212, ''Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors,'' Revision 2, fall 1980, and the guidance provided in GL 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate 24-Month Fuel Cycle,'' dated April 2, 1991. The associated TS Bases will be updated to reflect the proposed changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The LCOs [Limiting Conditions for Operation] for Technical Specifications [TSs] 3.8.1.1 and 3.8.1.2 will be changed to require a transmission network between offsite power and the onsite Class 1E distribution system, instead of just between offsite and the switchyard. This change, which will expand the requirement, is consistent with the current Millstone Unit No. 2 interpretation of the required distribution system. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously analyzed.

The diesel generators (DGs) supply power to the emergency busses at Millstone Unit No. 2 in the event of a loss of normal power (LNP). The emergency busses supply the vital equipment used to mitigate the consequences of design basis accidents. Therefore, the diesel generators are vital equipment used to mitigate the consequences of design basis accidents. Failure of the DGs will not cause a design basis accident to occur. However, failure of the DGs will affect the consequences of design basis accidents if a concurrent LNP occurs.

The proposed changes will revise the action requirements regarding operability testing of the DGs. The requirement to test the DGs if offsite circuits are inoperable will be deleted. An inoperable offsite circuit, by itself, will not affect the operability of the DGs. The requirement to test the remaining operable DG if one DG is inoperable will be modified. Testing will not be required provided a common cause failure is not the reason for declaring the DG inoperable. The requirement contained in the first footnote (*) to Technical Specification 3.8.1.1 to complete the test of the remaining DG will be deleted. The need to test the remaining DG will be based on the determination of a common cause failure. These changes will improve DG reliability by reducing the number of unnecessary starts and by requiring more appropriate testing of the DGs when there is a potential for common mode