expansion and the unexpanded tube. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The F* distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inner diameter based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the USAR accident analysis.

Implementation of the tubesheet plugging criterion will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the RCS (reactor coolant system) flow margin; thus, implementation of the F* criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

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 requests involve no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: John N. Hannon.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of amendment requests: January 13, 1995.

Description of amendment requests: The proposed amendments would revise Prairie Island Nuclear Generating Plant Technical Specification 4.4.D.1 to change the interval for the performance of the Residual Heat Removal (RHR) System leakage test from once every 12 months to perform the test during each refueling shutdown.

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. *The proposed amendment[s] will not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed changes to the RHR system leakage test interval only involve the leak-tightness of the RHR system for postaccident operation. As such, the proposed changes will have no impact on the probability of an accident previously evaluated.

The extension of the RHR system leakage test interval could increase the possibility of undetected RHR system leakage outside the containment during post accident conditions. However, the possible consequences of leakage from the RHR system outside containment are minor relative to those of the design basis accident. Therefore, because leakage from the RHR system has a minor effect on offsite dose, and since previous testing on a 12 month interval has not found significant RHR system leakage, the extension of the test interval to refueling is not expected to significantly impact the offsite dose consequences of an accident. In addition, it is probable that RHR system leakage would be identified during the normal quarterly functional testing and inspection of the RHR system.

Therefore, for the reasons discussed above, the proposed changes will not significantly affect the probability or consequences of an accident previously evaluated.

2. *The proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously analyzed.*

There are no new failure modes or mechanisms associated with the proposed changes. The proposed changes do not involve any modification of the plant equipment or any changes in operational limits. The proposed changes only modify the interval for the performance of the RHR system leakage test. The performance of the RHR system leakage test on a refueling basis instead of every 12 months cannot create a new or different kind of accident.

Therefore, for the reasons discussed above, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated, and the accident analyses presented in the Updated Safety Analysis Report [USAR] will remain bounding.

3. *The proposed amendment[s] will not involve a significant reduction in the margin of safety.*

The performance of the RHR system leakage test at power is more complex than performing the test during refueling shutdown. It is preferable, from an RHR system reliability and plant safety standpoint, to perform the test during refueling shutdown when the RHR system is already operating and when no changes to the RHR system configuration are required. Any possible increase in the risk to the public health and safety incurred by extending the RHR leak test interval from 12 months to refueling shutdown will be off-set by the reduction in risk obtained by not performing the RHR system leakage test during power operation.

The extension of the test interval would mean that possible RHR leakage could exist undetected for a longer period than allowed by the current Technical Specifications. However, the possible consequences of leakage from the RHR system outside containment are minor relative to those of the design basis accident. In addition, it is probable that RHR system leakage would be identified during the normal quarterly functional testing and inspection of the RHR system.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

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 requests involve no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: John N. Hannon.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: December 23, 1994.

Description of amendment request: The proposed amendment to the Technical Specifications revises the surveillance requirement to perform a visual inspection of containment areas affected by containment entry when containment integrity is established. It is consistent with Item 7.5 of Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

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. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.