evaluation report transmitted February 2, 1982, assumed that containment was completely abrogated and all radioactive materials released from the containment refueling pool are assumed to be released to the outside atmosphere. The requested amendments to Technical Specification 3.8.A.1.a modify the use of containment to mitigate the consequences of a fuel handling accident in containment, however, since instantaneous offsite release of all fuel handling accident materials released to containment has already been considered, the probability and consequences of a loss of containment accident are not increased.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

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

The requested amendments to Technical Specification 3.8.A.1.a modify the use of containment to mitigate the consequences of a fuel handling accident in containment. There are no new failure modes or mechanisms associated with the proposed changes, nor do the proposed changes involve any modification of plant equipment or changes in plant operational limits. Previous analyses, including the NRC fuel handling accident safety evaluation for Prairie Island, have already assumed the containment is abrogated. The proposed license amendments may affect the release path for fission products released during a fuel handling accident in containment, but no new or different kind of accident will result.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

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

The margin of safety as defined by the licensing bases fuel handling accident analyses is not reduced. The previous analyses are very conservative, assuming all radioactive material released from [containment] by the fuel handling accident is immediately released to the outside atmosphere, and bound any changes introduced by these requested amendments.

Technical Specification 3.8.A.1.a exists to minimize the consequences of a fuel handling accident in containment. However, with the current Technical Specification 3.8.A.1.a, there will still be releases due to the necessity to open the containment airlocks to evacuate personnel. With implementation of this amendment, the ability of the closed airlocks to contain the accident releases may improve.

Some radioactive material could be released through containment penetrations that are open at the time of the accident. Since it is not likely that containment will be pressurized by a fuel handling accident, the releases are expected to be minimal. This amendment will maintain containment post-fuel handling accident offsite releases well within the limits of 10CFR100 and the current license basis releases.

Therefore, a significant reduction in the margin of safety would not be involved.

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.

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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 9, 1995.

Description of amendment requests: The proposed amendments would revise Prairie Island Nuclear Generating Plant Technical Specification (TS) 4.12, ''Steam Generator Tube Surveillance,'' to incorporate revised acceptance criteria for steam generator tubes with degradation in the tubesheet roll expansion region. These criteria for steam generator tube acceptance were developed by Westinghouse Electric Corporation and are known as F* (''F-Star'') and L* (''L-Star''). These criteria would be utilized to avoid unnecessary plugging and sleeving of steam generator tubes.

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 supporting technical and safety evaluations of the subject criterion demonstrate that the presence of the tubesheet will enhance the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The results of hardrolling of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. The radial preload developed by the rolling process will secure a postulated separated tube end within the tubesheet during all plant conditions. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA loadings.

The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout is not expected for tubes using the F* criterion. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island Plant USAR [Updated Safety Analysis Report]. For plants with partial depth roll expansion like Prairie Island, a postulated tube separation within the tube near the top of the roll expansion (with subsequent limited tube axial displacement) would not be expected to result in coolant release rates equal to those assumed in the USAR for a steam generator tube rupture event due to the limited gap between the tube and tubesheet. The proposed plugging criterion does not adversely impact any other previously evaluated design basis accident.

Leakage testing of roll expanded tubes indicates that for roll lengths approximately equal to the F* distance, any postulated faulted condition primary to secondary leakage from F* tubes would be insignificant.

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

Implementation of the proposed F* criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity will be maintained. Tube bundle leaktightness will be maintained such that any postulated accident leakage from F* tubes will be negligible with regards to offsite doses.

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

The use of the F* criterion has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Reg Guide 1.121 [''Bases for Plugging Degraded PWR Steam Generator Tubes''] (intended for indications in the free span of tubes) and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criterion is any degradation indication in the tubesheet region, more than the F* distance below the bottom of the transition between the roll