

50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes do not affect the allowable suppression chamber average water temperatures provided in the TS. The changes do not affect previously evaluated events described in the UFSAR [Updated Final Safety Analysis Report] including all DBAs [Design Basis Accidents] and other operational transients.

The surveillance is extraneous because Action b of LCO [Limiting Condition for Operation] 3.6.2 directs the plant operators to commence a plant shutdown if the suppression chamber temperature cannot be restored. These changes do not affect plant systems, structures or components (SSCs).

Therefore, the proposed changes do not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not affect the design function or operation of a plant SSC. No physical or procedural changes are associated with this LCR [License Change Request]. As a result, no new credible failure mechanisms, malfunctions, or accident initiators are related to this change. Additionally, no new modes of plant operation are created.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes include the deletion of a surveillance requirement. This change is prompted by an LCO action statement, which prevents the plant from performing the surveillance. As a result, this change does not impact safety margins specified in the Hope Creek licensing basis.

Therefore, the proposed changes do 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.

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Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: May 21, 2003.

Description of amendment request:

The proposed amendment would change the source term for the Dose Calculation Methodology to the Alternate Source Term (AST). This change would result in design modifications to the Control Room Emergency Air Treatment System (CREATS), eliminate the requirement for the Containment Post Accident Charcoal Filters, and revise both the reactor coolant dose equivalent I-131 specific activity limit and the containment spray NaOH concentration limit.

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 the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The function of the CREATS is to provide a safe environment for the operators in the event of an accident, and thereby allow them to perform their accident mitigation responsibilities. The physical changes to the CREATS were designed to enhance the ability of the system to perform that function. The new system is an improvement in reliability, redundancy and leak tightness over the existing system. The change in design has no impact on accident initiation frequencies. Therefore the physical changes to the plant do not increase the probability or consequences of a previously evaluated accident.

The proposed Technical Specification changes involving the CREATS reflect the new system configuration and current industry guidance. The specifications ensure system functionality and protection of the operators under postulated accident conditions.

The new dose analysis indicates that the radiation dose to the operators and the public is acceptable without crediting the post accident charcoal filters removed from Technical Specification 3.6.6 and 5.5.10, and also bounds the change to the Reactor Coolant System activity limits in Technical Specification 3.4.16. The change to the dose conversion factor definition in Technical Specification [S]ection 1.1 is consistent with the new analysis.

The reference to ICRP-30 [International Commission on Radiological Protection Publication No. 30] in the Dose Equivalent I-131 definition is consistent with the new analysis and Standard Tech Specs, NUREG1431, ["Standard Technical Specifications Westinghouse Plants."]

All calculated doses are within the regulatory limits prescribed in 10CFR50.67. In addition, with the exception of one calculated Exclusion Area Boundary (EAB) dose, all dose numbers are within the guidelines of Reg Guide 1.183, ["Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,"] and Standard Review Plan (SRP) 15.0.1. This above-mentioned dose is in one particular direction from the source. The associated accident is the Locked Rotor Accident, which was not previously evaluated for dose at Ginna. The 100% fuel failure assumption used in this accident is widely considered to be overly conservative. Additionally, extra margin is built into the calculation because RG&E [Rochester Gas & Electric Corporation] assumed 500 gallons per day (GPD) of Steam Generator (SG) tube leakage per SG. Since the primary release pathway for this accident is SG tube leakage, and Reg Guide 1.183 (reference 3) allows an assumed tube leakage equal to the Tech Spec allowable leakage (~150 GPD/SG at Ginna), RG&E assumed a release rate of ~3.3 times greater than required. The calculated dose (2.7 Rem) is well below the regulatory limit of 25 Rem and only slightly greater than the published guideline of 2.5 Rem. Given the localized nature, associated probability/risk, and conservatism in this analysis, the calculated dose is considered acceptable.

Iodine removal was not credited in the existing analysis of doses for Equipment Qualification. Therefore, even though the Containment Post Accident Charcoal Filters will be removed from Tech Specs as a result of this amendment, it is not necessary to re-analyze these doses.

The Toxic Gas in-leakage analysis is bounded by the assumed in-leakage in the dose analysis. The amendment also does not hinder or change the ability to mitigate smoke infiltration as described in NEI [Nuclear Energy Institute] 99-03, Control Room Habitability Guidance.

This change has no impact on accident initiators, will not affect the ability of the operators to perform their designated functions, and removal of the requirement for CNMT [Containment] Post Accident Charcoal Filters is shown to be acceptable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

For the proposed changes, a different kind of accident would involve a situation where the operators would become incapacitated or otherwise be prevented from fulfilling their function. The new system differs in that the cooling in the emergency mode is from direct expansion of R-22 refrigerant. A rupture of the coils could introduce the refrigerant into the Control Room environment. However, the charge of refrigerant R-22 in cooling system will be limited such that a rupture in the cooling coils would not exceed nationally accepted toxicity standards.

The radiation and/or toxic gas exposures are shown to be acceptable, and the ability