Dan O'Neal 10/2/2001

The NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. Circumstances may arise in which new information reveals an unforeseen hazard or a substantially greater potential for a known hazard to occur, such as identification of a design vulnerability or an issue that substantially increases risk. For CRDM nozzle degradations, the NRC believes a "special circumstance" exists in which compliance with the Commission's regulations does not address a safety issue that may have significant risk implications for adequate protection of public health and safety.

The judgement regarding adequate protection derives from a diverse set of considerations, such as acceptable design, construction, operation, maintenance, ,modification, and quality assurance measures, together with compliance with NRC requirements. Furthermore, quantitative risk estimates serve as an important measure of plant safety, but do not embody the full range of considerations that enter into the judgement regarding adequate protection. However, Final Policy Statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement", Vol. 60, p. 42622, August 16, 1995,) states that probabilistic risk assessment (PRA) methods can be used to derive valuable insights, perspective, and general conclusions as a result of an integrated examination of facility designs, facility response to initiating events, expected interactions among facility structures, systems, and components, and between the facility and its operating staff.

Regulatory Information Summary (RIS) 01-002, "Guidance on Risk-Informed Decisionmaking in License Amendment Reviews," provides a process for the staff to consider whether a "special circumstance" rebuts the presumption that compliance with the regulations provides adequate protection of public health and safety. Although developed for staff reviews of license amendment requests, the process in RIS 01-002 is appropriate for other regulatory decisionmaking purposes because it addresses the fundamental requirement for operation of a nuclear reactor, that there is reasonable assurance of adequate protection for the public health and safety.

Application of the RIS 01-002 process to this issue has three steps:

1. identification of a "special circumstance" involving a risk factor not addressed by regulations;
2. assessment of the factor with respect to the five safety principles of risk-informed decisionmaking to establish whether its effect is sufficiently large to rebut the assumption that adequate protection is achieved by compliance with existing regulations; and
3. identification of an adequate basis for establishing reasonable assurance of adequate protection when the factor is considered.

The current regulation requires inspections to be performed in accordance with ASME Code requirements. However, the Code specifies procedures which are inadequate to detect the subject degradation because it cannot detect the amount of leakage that is expected to occur before CRDM housing failure and loss of coolant accident (LOCA) results. So, a "special circumstance" exists with respect to this issue, satisfying step one in the RIS-01-002 process.

The second step is to evaluate the issue with respect to the safety principles and integrated decisionmaking process described in Regulatory Guide (RG) 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the

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Licensing Basis", in which adequate protection considerations for this issue are addressed by the five safety principles. The circumstance is acceptable if the change:

1. meets current regulations unless it is explicitly related to a requested exemption or rule change,
2. is consistent with "defense-in-depth philosophy,"
3. maintains sufficient safety margin,
4. results in only a small increase in core damage frequency or risk, and consistent with the intent of the Commission Safety Goal Policy Statement ("Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," Federal Register, Vol. 51, p. 30028, August 4, 1986), and
5. the impact of the proposed change should be monitored using performance measurement strategies.

With respect to these criteria, the "special circumstance" of CRDM nozzle inspections that are inadequate to detect degradation that could result in failure satisfies only the first. These inspections do meet the current regulations because the regulations only reference the inadequate ASME Code requirements. This circumstance is inconsistent with the second principle, maintaining the "defense-in-depth philosophy," because the regulations are not adequate to prevent the failure of the reactor coolant pressure boundary, which is one of the barriers to release of radioactive materials from the reactor core. Thus, one barrier is potentially lost. The third principle is not met because safety margins are not maintained by the ASME Code inspection requirements. Pressure boundary leakage can remain undetected and minimum wall thickness requirements can be violated without detection before gross failure occurs.

The use of PRA in the fourth principle is consistent with the Commission Final Policy Statement, and PRA methods are used to derive valuable insights, perspective, and general conclusions as a result of an integrated examination of facility designs, facility response to initiating events, expected interactions among facility structures, systems, and components, and between the facility and its operating staff. The fourth principle is not clearly met. This principle is addressed using the acceptance guidelines in Section 2.2.4 of RG 1.174. The acceptance guidelines can be used to question whether adequate protection is assured, but they are not intended to define adequate protection.

The guidance indicates that CDF increases of 10^{-5} /yr or greater above the baseline should not be allowed for plants with a total baseline CDF between 10^{-5} /yr and 10^{-4} /yr. For plants with a total baseline CDF above 10^{-4} /yr, a CDF increase of 10^{-6} /yr or greater above the baseline should not be allowed. If the CRDM nozzle degradation were to continue unnoticed, the estimated increase in core damage probability would approach the conditional (given MLOCA initiation) core damage probability(CCDP) for a medium LOCA as the CRDM nozzle failure becomes imminent. To evaluate a numerical increase in CDF, a point estimate of the initiating event frequency for the medium LOCA should be evaluated from the probabilistic fracture mechanics. However, staff believes that no reasonable point estimate for the initiating event frequency is currently attainable primarily from a lack of data and insufficient understanding of the CRDM fracture mechanics.

The CCDP values for the subject plants are on the order of 5.3×10^{-3} or larger for a

This contradicts the next paragraph which gives a study's result.

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medium LOCA, according to the Individual Plant Examination (IPE) study performed for Generic Letter 88-20. Using the above CCDP value of 5.3 x 10^{-3} for the plants with a total baseline CDF between 10^{-5} /yr and 10^{-4} /yr, the initiating event frequency higher than 1.8 x 10^{-3} could result in an unacceptable increase in CDF. In a recent study of the Probabilistic Fracture Mechanics (PFM), the cumulative probability of a CRDM nozzle failure was evaluated after 20 effective full power years of operations at 600° F without inspections, and the event frequency was estimated as 1.0 x 10^{-2} /yr or larger based on a conservative average over 20 years. The study was performed for CRDM nozzle failure using the Weibull probability function, where the parameters for the fracture model were determined by comparison with the data for the Oconee and ANO-1 units and data for the PWSCC of steam generator tubes. With this event initiation frequency of 1.0 x 10^{-2}, the increase in CDF would be at least 5.0 x 10^{-5} or more. This is well above RG 1.174 guidance value of 10^{-5}/RY for CDF increments that would be considered only when total CDF is shown to be below 1x 10^{-4}/RY.

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Finally, the fifth principle is not satisfied because the basis for licensee analysis, which indicates risk levels below RG 1.174 numerical guidelines, are based on assumptions tha can not be verified without performing the inspections that are adequate to detect the form of degradation being modeled. Therefore, assessment with respect to these safety principles rebuts the assumption that compliance with the regulations in this "special circumstance" is sufficient to provide reasonable assurance for adequate protection of the public health and safety.

The third and final step for application of the RIS 01-002 process involves identification of an adequate basis for establishing reasonable assurance of adequate protection when the "special circumstance" is considered. The General Design Criteria (GDCs) in 10CFR50 Appendix A establish a general statement of the Commission's perspectives on the factors that are sufficient to achieve "adequate protection", particularly with three GDCs applicable to this case. GDC 14 states that "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage or rapidly propagating failure, and of gross rupture." Criterion 30 states that "Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage." Criterion 32 states that "Components of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural integrity and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel." Taken as a whole, these GDCs make it clear that the reactor coolant pressure boundary is to be maintained in a leaktight and structurally sound condition, with extremely low probability of gross failure.

Clearly, failure to inspect a portion of the reactor vessel in a manner that is sufficient to detect the extent of degradation caused by a mechanism known to be degrading other plants in that portion of the vessel is inconsistent with these GDCs. The level of degradation that has been found in other plants, if left undetected and uncorrected, would result in a gross failure of the reactor coolant pressure boundary

In summary, the staff does not have reasonable assurance that adequate protection is achieved by plants that do not perform inspections that are sufficient to detect this type of degradation.

On that basis, the Commission may issue an order to require licensees with highly susceptible plants to perform inspections adequate to detect the CRDM nozzle degradation before margins are lost and gross rupture is possible.