PRE-~~DECISIONAL~~ & SENSITIVE

FORM 23-I: Confirmatory Order Modifying License (Effective Immediately) (Reactor Licensee)

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	Docket No. ________
(LICENSEE)	)	License No. ________
(Facility Name)	)	EA-YY-XXX

CONFIRMATORY ORDER MODIFYING LICENSE
(EFFECTIVE IMMEDIATELY)

I - Licensee

Blank No. 1 (Name of licensee) (Licensee) is the holder of Facility Operating License No. Blank No. 2 issued by the Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR Part 50 on Blank No. 3 (Date-1) . The license authorizes the operation of Blank No. 4 (XYZ facility) in accordance with conditions specified therein. The facility is located on the Licensee's site in Blank No. 5 (City, State) .

Jim - 9/28
Comments.
Bill

or Issuing This Order

Section 2.202 of Part 2, Subpart B, of Title 10, *Code of Federal Regulations* and Section 50.109 to Title 10, *Code of Federal Regulations* (10 CFR 2.202 and 10 CFR 50.109, respectively) promulgate the requirements for issuing orders to holders of licenses issued under 10 CFR Part 50.
10 CFR 2.202(a)(1) gives the Commission the authority to issue orders to holders of NRC operating licenses when the Commission determines that a given licensee has failed to respond and resolve the regulatory and safety issues associated with a documented violation of the Commission's requirements, or when the Commission has determined that other potentially hazardous conditions at a facility could exist that would warrant issuance of an order. 10 CFR 2.202(a)(5) givess the Commission the authority

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physical obstructions may also limit the capability of VT-2 examination method ~~equipment~~ to ~~resolve~~ minute amounts of boric acid deposits on the outer surface of the vessel head ~~that would be capable of being detected by the human~~. Cleanliness of reactor vessel heads during the examinations is also a critical aspect, as it is important for visual examination methods to be capable of distinguishing between boric acid residues that result from VHP nozzle leakage and those residues that result from leaks in other reactor coolant system components.

or to see remote visual examination devices.

Currently, the vessel heads of Babcock and Wilcox designed reactors are the only PWR vessel heads that can be effectively examined by VT-2 methods without ~~extensive efforts~~ the need to either remove thermal insulation materials from the vessel heads. ~~Based on information supplied by the industry, extensive efforts are needed to remove the insulation materials from CE and Westinghouse designed vessel heads if visual examinations are to be effective methods of detecting leakage from the CRDM nozzles.~~ As stated in IWA-5242, Section XI of ASME Code there is not a requirement for ~~does not currently require~~ licensees to remove thermal insulation materials when performing ASME VT-2 examinations of the reactor vessel head. There is therefore no guarantee that licensees owning CE or Westinghouse designed will remove the thermal insulation materials from their heads when conducting their VT-2 examinations. Based on these uncertainties, and the fact that the Duke Power did not detect the circumferential cracking in the CRDM nozzles until after it had initiated its repair activities for the nozzles creates a significant uncertainty as to whether the current Section XI ISI methodology for conducting visual examinations and dispositioning recordable flaw indications is capable of detecting the presence of significant O.D. initiated, circumferential cracks in U.S. CRDM nozzles.

considered for deletion

Start @ this point and discuss the special circumstance

This situation constitutes a "special circumstance" in which compliance with the Commission's regulations does not address a safety issue that may have significant risk implications. Regulatory Information Summary (RIS) 01-002, "Guidance on Risk-Informed Decisionmaking in License

The regulations, specifically

10CFR50.55a, identify ASME Section XI as the source for inspection requirements. As described above, however compliance with these inspection requirements ~~does not~~ is not adequate to detect and prevent potential cracking and gross failure of the CRDM head penetrations.

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 first step involves identification of a special circumstance.

The current regulation, 10 CFR 50.55a, requires licensees to perform inspections of their vessel heads in accordance with the inservice inspection requirements of Category B-P to Table IWB-2500-1 of Section XI, ASME Boiler and Pressure Vessel Code. For the case of VHPs, ~~However,~~ the Code requirements ~~specifies procedures which~~ are not ~~in~~adequate to assure the structural integrity of CRDM nozzles i.e., ~~against catastrophic failure because~~ the specified ~~required~~ examination ~~criteria allow pre-examination practic~~es ~~(i.e., removal of thermal insulation materials and hence~~ adequate cleaning of the ~~reactor vessel~~ heads ~~is not required) that make it extremely questionable as to whether the [illegible] inspection meth~~ods ~~will~~ is not capable of detecting ~~the type reactor coolant pressure boundary leakage~~ it ~~is intended to monitor~~ for. The Code requirements are therefore

the

cracking in VHP nozzles.

inadequate to monitor for degradation in the VHP nozzles prior to a postulated occurrence of a CRDM nozzle failure, and small-break LOCA scenario. This is contrary to the statement in the Preface to Section XI that states "The rules . . [of Section XI] . . require a mandatory program of examinations, testing and inspections to evidence adequate safety . . [of a nuclear power plant]." Thus, a "special circumstance" exists with respect to this issue. This satisfies step one in the RIS-01-002 process..

The second step is to evaluate the issue with respect to the safety principles and integrated decision-making process described in Regulatory Guide 1.174 (RG 1.174), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The five safety principles are that the circumstance is acceptable if it:

1. meets current regulations,
2. is consistent with "defense-in-depth philosophy,"
3. maintains sufficient margin,
4. results in only a small increase in core damage frequency, and
5. the basis for the risk estimate is 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 of these principles, i.e., These inspections do meet the current regulations ~~because the regulations only reference the inadequate ASME Code requirements.~~ ~~This~~ The 'special circumstance' does not satisfy ~~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 three barriers to release of radioactive materials from the reactor core. Thus, one barrier is potentially lost. The third principle is not ~~met~~ satisfied because 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 fourth principle is not ~~met~~ because core damage frequency can eventually increase to the relatively high numerical value for the conditional core damage probability (CCDP) for the loss-of-coolant accident (LOCA) that would result from gross CRDM nozzle failure. The CCDP values for the subject plants are on the order of 5×10^{-3}/reactor-year for a medium-to-small LOCA. This is well above RG 1.174 guidance value of 1×10^{-5}/RY for CDF increments that would be considered only when total CDF is shown to be below 1×10^{-4}/RY. Finally, ~~the circumstance cannot meet~~ the fifth principle is not satisfied because the basis for any licensee analysis that shows risk levels below RG 1.174 numerical guidelines must be based on assumptions that cannot be verified without performing the types of inspections that are capable of detecting the form of degradation being modeled. ~~Therefore, assessment with respect to these safety principles rebuts~~ In summary, four of the five safety principles this special circumstance does not satisfy, and, therefore, 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 is not valid.

got stronger words

The third and final step for application of the RIS 01-002 process involves identification of an adequate basis for establishing reasonable assurance of protection when the special circumstance is considered. ~~establishing an alternative basis for reasonable assurance.~~ The Commission has compiled a number of general design criteria (GDC) for the design, fabrication, construction, testing and performance of structures, systems and components important to safety in Appendix A to 10 CFR Part 50. The GDC establish a general statement of the Commission's perspectives on the factors that are sufficient to achieve "adequate protection." Three GDCs are relevant 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

Where does this come from

their structural integrity and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel." Taken as a whole, these GDCs emphasize that the Commission considers that it is extremely important from a safety standpoint to maintain the reactor coolant pressure boundary in a leaktight and structurally sound condition, with extremely low probability of gross failure.

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, given the special circumstance, Therefore, 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 these bases, pursuant to the provisions and criteria in 10 CFR 2.202, the Commission has the authority to issue an order requiring licensees with most highly ranked (susceptible) CRDM nozzles to perform inspections that will be capable of detecting the CRDM nozzle degradation or leakage before the safety margins for the nozzles are lost and gross rupture is possible. On these bases, pursuant to 10 CFR 50.109(a)(4)(ii), the staff would not be required to perform a backfit analysis for any order that may be issued under these principles.

By letter dated Blank No. 11 (Date-2) , the Licensee submitted its responses to NRC Bulletin 2001-01 for the Blank No. 12 (XYZ Facility). The Licensee's response to NRC Bulletin 2001-01 indicates that the Blank No. 13 - (provides the plant specific input from Al Hiser that summarizes the technical information in the response to NRC Bulletin 2001-01 for the XYZ facility). Based on the Inadequacy of ASME Section XI ~~uncertainties in the ability of current NDE~~ inspection methods to detect degrading ~~and size recordable PWSCC type flaws in U.S.~~ CRDM nozzle~~s~~, and the inability of the industry to establish a defensibly low initiating event

to reactor pressure vessel head welds