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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 3, 1981

Docket No. 50-266

Mr. Jerry E. Mendl
WEPA Coordinator
Wisconsin Public Service Commission
State of Wisconsin
Hill Farms State Office Building
Madison, Wiscon'n 53702

Dear Mr. Mendl:

This is in response to your letter of February 23, 1981, wherein you asked about the possible costs of operating a facility with radioactive crud buildup and reactor vessel embrittlement. This response reflects the clarifications offered by both you and Mr. Peter Anderson of Wisconsin's Environmental Decade, Inc. in a conference call on February 23, 1981.

The question about crud stems from the knowledge that Commonwealth Edison Company has requested permission to decontaminate the reactor coolant system at their Dresden Nuclear Power Station, Unit No. 1 (a dual cycle boiling water reactor). Information related to this decontamination operation (including costs), and the reasons for it, are contained in NUREG 0686, "Final Environmental Statement Related to Primary Cooling System Chemical Decontamination at Dresden Nuclear Power Station, Unit No. 1" (copy enclosed). As discussed in Section 2.3 of this document, the reason for the request for decontamination at Dresden-1 is a special case. It should not be inferred from this that each nuclear plant will need to undergo this evolution after an operating period of about 20 years. Such is not the case. The crud levels at Point Beach are typical for pressurized water reactors. Although the possibility of a full-scale chemical decontamination at Point Beach over its lifetime cannot be ruled out, there is nothing to indicate at this point that it would be necessary. We should add that the steam generators at Surry and Turkey Point are going to be or have been replaced without the need for chemical decontamination of the entire reactor coolant system. Rather, local decontamination has been used to reduce occupational radiation exposure. These measures would probably be employed by WEPCO for either steam generator replacement or tube sleeving. No specific cost information is available, but the nature of the operation leads us to conclude that the cost is small.

As to your question about reactor vessel embrittlement, we learned from you that the question stems from a discussion of Unresolved Safety Issues presented in the NRC Annual Report for 1979. We have enclosed the pertinent pages of this report for your convenience. The origin of this issue and its plan for resolution are summarized therein. Point Beach Unit 1 is one of the 20 older operating plants included in this summary.

Generically, this issue is now satisfactorily resolved. Its resolution will be issued as NUREG-0744 to be titled, "Resolution of Reactor Vessel Material Toughness Safety Issue". This document is scheduled for publication on April 15, 1981 for public comment.

As for Point Beach Unit 1, we fully expect that the pressure vessel will maintain adequate toughness and safety margins for the remaining life of the unit.

We have also reviewed, as requested, your draft response to Mr. Anderson's letter of February 2, 1981 which you enclosed with your letter. We should point out that the San Onfre sleeving operation is not going as well as expected. Southern California Edison is having difficulty obtaining a leak-tight joint between the top end of the sleeve and the steam generator tube in certain areas. The problem is believed to be due to sludge. The Point Beach Unit 1 steam generators do not have a sludge problem of this magnitude, and therefore it may well be that this would not pose a problem for Wisconsin Electric Power Company. Edison is now installing leak-limiting sleeves in areas where leak-tight sleeves have proved difficult.

Your proposed response A.3 (second paragraph) warrants minor comment. In Appendix A to our Safety Evaluation Report of November 30, 1979, the NRC staff made two in-leakage calculations: the first assumed that a crack existed in a steam generator tube in the mid-depth of the tube sheet (about 10 inches); the second, more conservative calculation assumed that a steam generator tube had a complete circumferential ("guillotine") break 0.5" below the top of the tubesheet.

Sincerely,

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Divison of Licensing

Enclosures:
1. NUREG-0686
2. NRC Annual Report, pp. 75, 76

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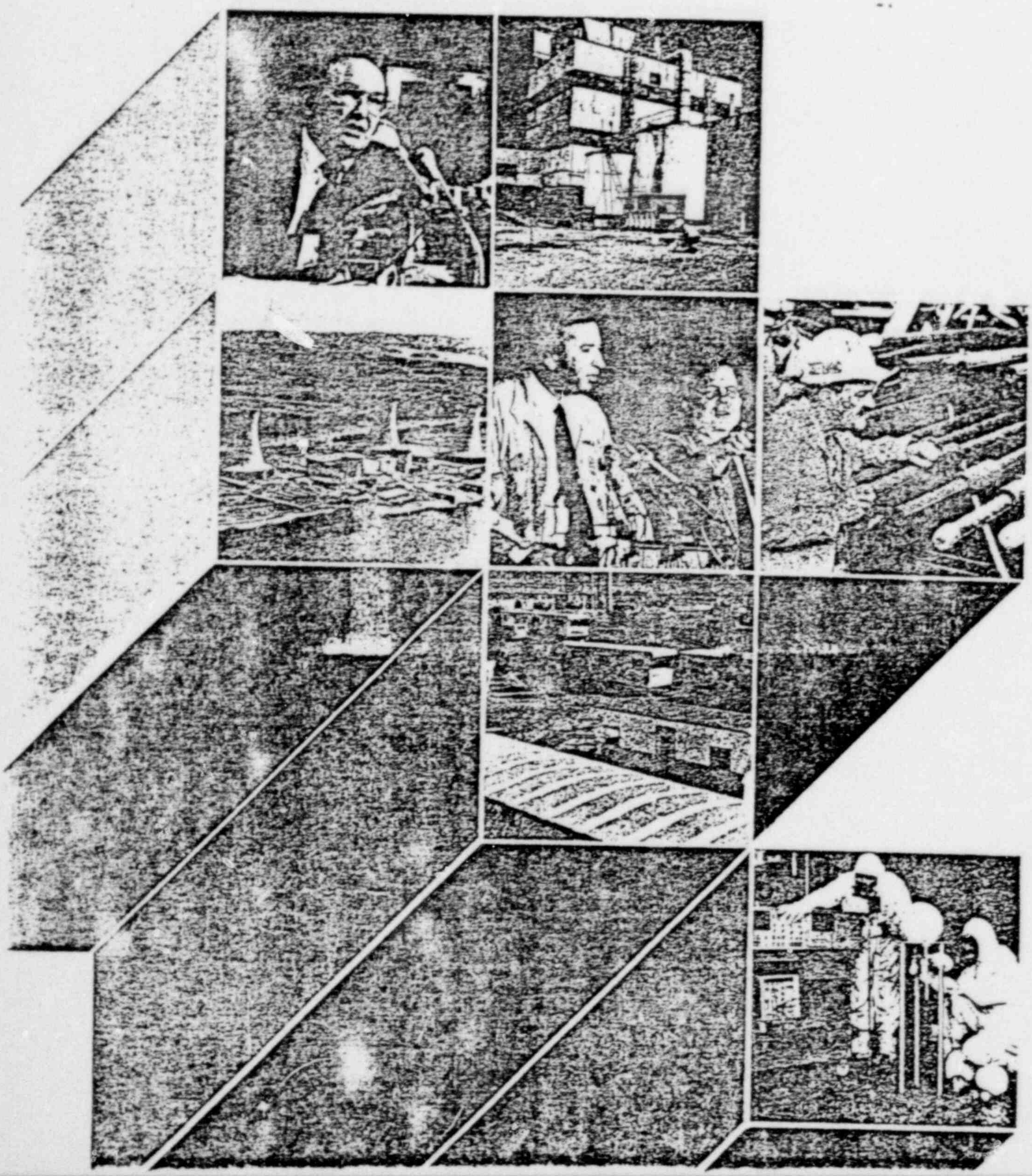
1979
Annual Report



changes, to lengthen the time to crack initiation and to slow crack growth are taken into account in the determination of inspection techniques and criteria.

The CRD nozzle issue will be resolved by a combination of actions which includes nozzle inspection and repairs and some CRD system notifications. Certain system modifications recommended by General Electric involved cutting and capping the nozzle and return line but that action would reduce the capability to direct high pressure water through the CRD system when the vessel is otherwise isolated. Although this system is not normally expected to perform this function in safety analyses, the capability played a major role in keeping the core covered during the incident at Browns Ferry Unit 1 on March 22, 1975. As a result of its review of these modifications, the NRC has concluded that only a limited number of plants will be allowed to modify the CRD system in accordance with the GE recommendations. Unless the licensees of the remaining plants demonstrate, by testing, that sufficient flow is available to the reactor vessel with the return line removed, they will be required to retain the return line, rerouted to the feedwater line or a similar suitable connection that doesn't have the potential for cracking in the reactor vessel nozzle. The staff's evaluation, conclusions, and guidance on the CRD return line nozzle issue will also be included in the February 1980 NRC staff report referred to above.

Plant-specific implementation of the generic licensing positions developed under this task (with the exception of future inservice inspection questions) has already begun.

Reactor Vessel Material Toughness

Nuclear reactor pressure vessels are required to have an adequate margin of protection against fracture in the presence of relatively large postulated flaws. This requirement is imposed for the sake of conservatism, even though extensive, periodic inservice inspection programs serve to provide protection against the presence of such flaws. Fracture mechanics—the engineering method used to establish the failure margin—employs a quantitative material property called fracture toughness to calculate the conditions under which catastrophically rapid crack propagation will occur. Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessels, three facts are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, the technical specifications for power reactors set limits on the operating pressure during heatup and cooldown operations. These restrictions assure that the combination of pressure and temperature will remain well below that which might cause brittle fracture of the reactor vessel if a significant flaw were present in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these technical specifications over the life of the plant.

For the service time and operating conditions typical of current operating plants, reactor vessel fracture toughness provides adequate margins of safety against vessel failure. Further, for most plants the vessel material properties are such that adequate fracture toughness can be maintained over the life of the plants. However, results from a reactor vessel surveillance program indicate that up to 20 older operating pressurized water reactor pressure vessels were fabricated with materials that will have marginal toughness after comparatively short periods of operation. This issue has been incorporated in the

The protective insulation has been pulled aside following the testing of a weld-repair portion of a six-inch thick pressure vessel. A flaw more than five inches deep and 18 inches long was created in the area which was then subjected to pressure overloads more than double the design pressure, without disruptive failure.

NRC staff's program for the resolution of generic issues as Task A-11.

The fundamental goal of Task A-11 is to provide an improved engineering method by which to assess the safety margin in nuclear reactor pressure vessels and to develop appropriate criteria for the evaluation of normal, transient, or postulated accident conditions under the improved method. This method could then be used to provide such an assessment for those older reactor pressure vessels that will eventually have marginal toughness according to the current method. Because relatively large amounts of prefracture plastic deformation can be expected at high temperatures even in pressure vessel steels of low toughness, the new evaluation method will employ "elastic-plastic" fracture mechanics concepts. The basis for this improved methodology is described in NUREG-0311, "A Treatment of the Subject of Tearing Instability," developed under an NRC-sponsored program at Washington University. Additional Washington University work extending the methodology to reactor pressure vessels was funded by the Department of Energy. The engineering method developed will account for radiation-induced material degradation.

Task A-11 also includes or relies on programs sponsored by the NRC Office of Nuclear Regulatory Research to provide: (1) an improved evaluation of material degradation mechanisms resulting from neutron irradiation, and (2) the development of improved testing methods for use in determining the elastic-plastic properties of materials.

Since last year's report, the following has been accomplished:

- Although delayed, an elastic-plastic fracture test method for routine determination of fracture toughness was developed. Verification of the test method is underway.
- The elastic-plastic fracture mechanics methods of NUREG-0311 were confirmed by work supported by an Electric Power Research Institute program, "Methodology for Plastic Fracture."
- The methods developed in these programs were successfully used by NRC contractors to analyze two pressure vessel burst tests reported in the Heavy Section Steel Technology Program, sponsored by the NRC Office of Nuclear Regulatory Research.
- The potential for restoring by thermal annealing the pressure vessel toughness lost by neutron radiation was shown to be impractical.

Significant delays have developed over the past year as a result of difficulties encountered in extending the new engineering methodology to reactor pressure vessels. There is agreement among experts that the methodology can be extended, but it will require a significantly greater effort than that accomplished under the DOE contract referred to above. The staff will carry out this additional work by contract with the Oak Ridge National Laboratory. Because of this problem, the schedule for completing Task A-11 has slipped about one year, to December 1980.

Fracture Toughness and Potential for Lamellar Tearing of PWR Steam Generator And Reactor Coolant Pump Supports

During the course of licensing review for a specific pressurized water reactor (PWR) a number of questions were raised as to (1) the adequacy of the fracture toughness properties of the material used to fabricate the reactor coolant pump supports and steam generator supports, and (2) the potential for failure due to lamellar tearing of these same supports. The safety concern is that, although these supports are designed for worst-case accident conditions, low fracture toughness or lamellar tearing could cause the support to fail during such accidents. Support failure could conceivably impair the effectiveness of systems designed to mitigate the consequences of the accident. An example of a postulated event sequence of potential concern would be a large pipe break in the reactor coolant system which would severely load the supports, followed by a support failure of sufficient magnitude that a major component such as a steam generator would be displaced resulting in failure of the emergency core cooling system piping needed to provide cooling water to the core.

Because materials and designs similar to those of the PWR originally reviewed have been used in other plants, review of this issue was included in the NRC Program for Resolution of Generic Issues as Generic Task A-12.

A consultant was engaged to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in the later stages of operation license review. This reassessment included review of the materials utilized in the support of 38 potentially affected PWRs. Based on the consultant's evaluation, it was determined that there are 21 plants whose supports are of questionable toughness and, accordingly, further detailed plant-specific review is required. This decision concluded the generic study of this subject under Task A-12. During the plant-specific reviews that will follow, either the structural integrity of the supports must be demonstrated, or measures to assure their structural integrity will be required.

A report describing the NRC staff's safety evaluation and conclusions and describing its plans for implementation (i.e., the more detailed plant-specific reviews referred to above) was issued for comment in November 1979. It is entitled, "Potential for Low Fracture Toughness for Lamellar Tearing in PWR