7590-01

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
HAROLD R. DENTON, DIRECTOR

In the Matter of	)	
	)	
Iowa Electric Light & Power Company, et al (Duane Arnold Energy Center)	)))	Docket No. 50-331 (10 CFR 2.206)

DIRECTOR'S DECISION UNDER 10 CFR 2.206

In a request dated July 17, 1979, Citizens United for Responsible Energy (CURE) requested that the Director of Nuclear Reactor Regulation initiate a proceeding to modify Facility License No. DPR-49, for the Duane Arnold Energy Center (DAEC), such that compliance with General Electric Company recommendation "BWR Coolant Oxygen Control," NEDO-23631, June 1977, would be a limiting condition for operation. Notice of receipt of this request was published in the FEDERAL REGISTER on September 11, 1979 (44 FR 52912). For the reasons set forth in this decision, the request is denied.

In support of the request, the petitioner cites the cracks discovered in June 1978 in the DAEC recirculation system inlet nozzle safe-ends. The cause of the cracks was determined to be intergranular stress corrosion cracking (IGSCC). The petitioner states that such cracks indicate the Iowa Electric Light & Power Company (IELP) has failed to adequately address the serious problems created by IGSCC and that implementation of NEDO-23631 is necessary to ensure the continued safe operation of DAEC.

FACTUAL BACKGROUND

On June 17, 1978, a plant shutdown permitted the investigation of the source of a 3 gallon per minute (gpm) leak that had existed since June 14th (Plant Technical Specifications allow 5 gpm unidentified leakage). The leak was due to a through wall crack in one of the eight, Alloy 600 recirculation-inlet nozzle transition pipe ends (safe-ends). Non-through-wall cracks were subsequently found in the other seven safe ends.

An analysis by the NRC Pipe Crack Study Group (NUREG-0531, "Investigation and Evaluation of Stress Corrosion and Cracking in Piping of Light Water Reactor Plants," February 1979) concluded that the cracking was due to design and fabrication factors that led to high residual plus applied stresses and the presence of a crevice which promoted a coolant impurity concentration phenomena.

All eight inlet nozzle safe-ends were replaced. The replacement safe-ends and thermal sleeves were redesigned to reduce peak stresses and eliminate the previous crevice configuration. The safe-end/thermal sleeve adapter combination was fabricated from Alloy 600 the same type material as the original combination. The new design was evaluated and found acceptable. The staff concluded that there was reasonable assurance that IGSCC should not occur in the pressure boundary of the new design.

DISCUSSION

For BWRs in this country the long standing principle of reactor coolant chemistry control has been to maintain the water as completely free of impurities as possible. The recirculating coolant water is continuously purified in a clean-up system so that even trace impurity levels are exceptionally low. During operation the water decomposes by radiolysis in the nuclear flux, to produce both oxygen and hydrogen. Although the bulk of these gases are stripped into the steam and removed from the system at the condenser, a small amount stays in the recirculating water. The oxygen content of the circulating coolant water therefore depends upon the equilibrium conditions between the radiolysis of the primary coolant water, the stripping of the gas into the steam and the small quantities in the make up water. Because the BWR cycle is an open cycle, heretofore it has not been economical to inject additives into the circulating water, as is done in PWRs, to lower the oxygen level of the coolant. The oxygen level of the coolant is thus a function of the reactor power level. Because of the above, the Technical Specifications of the Duane Arnold plant, as well as those of other operating BWRs, do not require the monitoring or control of the oxygen level in the coolant. Tests of the primary coolant oxygen content of the Duane Arnold plant were performed during the early start-up tests and were consistent with the G.E. data developed during tests on a number of other BWR plants. The tests at DAEC indicate a level of 200 to 250 ppb of oxygen. Other BWR plants have similar oxygen levels.

The NRC Pipe Crack Study Group did an extensive study of the influence of oxygen in BWR primary coolant on IGSCC. (NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants.") The Group recommended that control of oxygen during shutdown and start-up be exercised wherever possible and said that reduction of oxygen levels during steady state operation of BWRs is desirable. However, the report also stated that "the data do not exist at the present time to determine if oxygen control during shutdown and start-up will prevent IGSCC in BWR piping." Thus, there is no assurance that the reduction of oxygen in these operational modes in a plant that had previously operated with the normal oxygen levels would prevent IGSCC in the future. For example, intergranular stress-corrosion cracks were found in Vermont Yankee core spray lines (not alloy 600) about two years after the utility started its practice of deaeration during start-up.

Much of the early technical support for deaeration as a remedy for IGSCC in BWRs came from two primary sources: 1) the belief that IGSCC was related to the number of reactor shutdowns greater than 24 hours long and 2) laboratory data showing that IGSCC was less severe in 0.2 ppm dissolved oxygen compared to 8 ppm dissolved oxygen (as in air saturated water). The belief that IGSCC was related to the number of reactor shutdowns, originated with a statistical analysis of a limited number of incidents, performed in a General Electric study published in 1974 (NEDO 21000). A more recent analysis of a larger number of incidents, established that the apparent correlation between shutdowns and cracking may not be a real correlation but only part of a more complex phenomenon. Unfortunately, the preliminary laboratory experiments conducted to support this thesis, were done in an unrealistic BWR environment, air saturated water at operating temperature. Actual in-reactor measurements of temperature/oxygen combinations showed that this severe condition rarely if ever occurs in operating BWRs. Therefore, laboratory studies that compare the severe condition of air saturated water at operating temperature to the low oxygen condition attainable with deaeration cannot produce a valid evaluation of the benefits of deaeration.

Recent experiments, in an EPRI program, using corrosion potential measurements in operating reactors and in laboratories indicate that the corrosion potential remains high in spite of deaeration. This suggests that deaeration alone will not stop IGSCC.

Although the G.E. staff study report (NEDO 23631) recommended that deaeration during start-up and shutdown be implemented in BWRs, the G.E. Co. to this date has not recommended it to the BWR owners and does not include it in its present design.

It is recognized that reduction of the oxygen content in the reactor coolant may be expected to provide a means for limiting the extent of IGSCC, however, this principle has not been fully developed. The NRC staff considers the recommendations made in NEDO 23631 worthy of being pursued further for development. The staff is not now requiring implementation of the recommendations of this report. There is insufficient data available to evaluate the secondary or synergistic effects of lowering the oxygen level of the coolant by deaeration, e.g., increased corrosion of the installed ferritic piping.

Studies have been initiated on various methods for controlling oxygen in BWRs including vacuum deaeration and hydrogen or amine additions. These studies will evaluate the corollary effects of oxygen reduction in the BWR coolant. The studies are being closely followed by the NRC staff and will be considered in resolving the IGSCC problem.

A review of BWR operating experience showed that the safe-end cracking at DAEC is the first example of IGSCC in Alloy 600 exposed to BWR water environment. Laboratory tests have shown that very high stresses (above yield) and tight crevice conditions, both of which were present in the original DAEC design, are significant factors in initiating stress-corrosion cracking in Alloy 600.

EVALUATION

The new safe-end design for DAEC has removed from the primary pressure boundary the tight crevice, and the weld which caused the high residual stresses in the original design. Therefore, there is reasonable assurance that stress corrosion cracking will not occur in the pressure boundary of the new design. The new design will allow increased circulation which will prevent the concentration of detrimental chemical species that can occur in a tight crevice.

In addition, an inservice inspection program for the safe-ends was instituted to detect any cracks should they occur. The program complies with the recommendations of Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, NUREG-0313, July 1977, and NUREG-0313 (Revision 1), October 1979, in which the staff concluded that existing plant design, inservice inspection programs, and leak detection monitoring systems ensure that IGSCC does not constitute a present safety problem for operating plants, and, therefore, constitute an acceptable basis for continued plant operation.

Under 10 CFR 50.109, the Commission may require "backfitting" of a facility if "such action will provide substantial, additional protection which is required for the public health and safety." Backfitting is defined as the addition, elimination or modification of structures, systems or components of a facility after the construction permit has been issued. Since the reduction of IGSCC is a long-term goal, this section of the regulations appears to be the proper basis for implementing the recommendations of NEDO-23631, if such implementation is deemed appropriate based on further study.

10 CFR 50.36(c)(2) defined limiting conditions for operation as "the lowest functional capability or performance levels of equipment required for safe operation of the facility." While NEDO-23631 may prove a desirable means of reducing the incidence of IGSCC, even then it may very well not be <u>required</u> for safe operation because of the long term effects and the short term crack detection by the inservice inspection and leak detection, and therefore, it would not qualify as a limiting condition for operation.

Notwithstanding the assurance provided by the new safe-end design, the NRC staff considers the recommendations made in NEDO-23631 worthy of being pursued further. It is recognized that reduction of the oxygen content in reactor coolant may be expected to provide a means of limiting the extent of IGSCC. However, this principle has not been fully developed, and is not now a requirement for present safe plant operation. Studies have been initiated on various methods for controlling oxygen to establish whether the presence of oxygen has any bearing on IGSCC. These studies are being closely following by the NRC staff.

CONCLUSION

For the reasons stated in this decision, I have determined not to modify Facility License No. DPR-49 for the DAEC such that compliance with NEDO-23631 is a limiting condition for operation. Accordingly, the request of the petition is denied.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D,C, 20555, and the local public document room for the Duane Arnold Energy Center, located at the Cedar Rapids Public Library, 426 Third Avenue, S.E., Cedar Rapids, Iowa 52401. A copy of this decision will also be filed with the Secretary of the Commission for its review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

In accordance with 10 CFR 2.206(c) of the Commission's Rules of Practice, this decision will constitute the final action of the Commission twenty (20) days after the date of issuance, unless the Commission on its own motion institutes review of this decision within that time.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 24th day of September, 1980