UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION CONCERNING THE REQUIREMENTS FOR REDUNDANCY IN RESPONDING TO THE RAPID DEPRESSURIZATION ACCIDENT

PUBLIC SERVICE COMPANY OF COLORADO

FORT ST. VRAIN NUCLEAR GENERATING STATION

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1.0 INTRODUCTION AND BACKGROUND

Public Service Company of Colorado (PSC) committed in 1984 to an upgrade of the Fort St. Vrain Technical Specifications. This commitment was made in a PSC letter dated November 16, 1984. The licensee submitted a final draft of these Technical Specifications (TS) in a letter dated October 11, 1985. This letter specifically addressed how decay heat would be removed following various accident sequences. The staff provided a markup and comments in response to the licensee's submittal in a letter dated December 12, 1985. One issue raised by the staff following this letter was how the plant would respond to the rapid depressurization accident, known as Design Basis Accident No. 2 (DBA-2).

The specific concern raised by the staff was the redundancy of the equipment to be covered by the TS in response to DBA-2. Specifically, in order to respond to DBA-2, the plant was required to have available at least:

-2 helium circulators in one loop,
-1 boiler feed pump (BFP), and
- offsite power.

In normal operation, two circulators would be available and at least one BFP is currently required. However, if single failures were considered, the staff questioned the adequacy of this equipment set. The normal redundancy requirements were not met. Also, there was no onsite power source capable of carrying the required loads. That is, the large, electrically driven, BFP was not designed to be carried by the onsite diesel generators.

The licensee responded to this concern in a letter dated February 28, 1986. The licensee stated that the addtional redundancy and onsite power were not required because the probability of DBA-2 was very low. Specifically, the licensee submitted estimates of the probability of DBA-2 to justify this position.

The staff requested its contractor, Oak Ridge National Laboratory (ORNL), and a subcontractor, Science Applications International Corporation (SAIC), to examine the licensee's submittals. ORNL and SAIC were asked to evaluate the licensee's analysis and to provide alternative evaluations as needed.

The staff noted that no evaluation existed of the capability of the class 1E safety shutdown systems to respond to DBA-2. An earlier staff action dated July 2, 1987, had limited the power of FSV to 82 percent of full power. Given this additional restriction, ORNL was asked to evaluate the class 1E system's response to DBA-2.

2.0 EVALUATION

Detailed technical evaluations of the licensee's submittal have been performed by ORNL and SAIC. These are summarized in the enclosed Technical Evaluation Reports (TERs).

Evaluation of the Licensee's Estimates for the Probability of DBA-2

The ORNL TER questioned the adequacy of the underlying assumptions in the licensee's estimates. The licensee's estimates were based on generally accepted rupture frequencies for pressurized water reactor (PWR) vessels. Following this same methodology, ORNL concluded that the licensee's estimates assumed very high quality in the initial fabrication of the prestressed concrete reactor vessel (PCRV). This high quality level was also assumed present in the fabrication of the vessel penetrations and the associated closures. The licensee's estimates also assumed that the PCRV structure was subject to a high level of inservice inspection (ISI) that supported a reliable structure. ORNL concluded that the lack of a detailed ISI program for the PCRV, did not support the licensee's analysis of the rupture frequency for the PCRV.

The staff does not agree with ORNL's conclusions on this issue. The staff notes that efforts are continuing to formulate an adequate ISI program for FSV. This is being addressed in other licensing actions (see Amendment No. 51 to the FSV license dated March 9, 1987). Therefore, the staff does not find that the licensee's estimates are in doubt only because of ISI concerns.

ORNL has performed a separate estimate of the probability of DBA-2. ORNL's estimate is based on more rigorous estimation process, with more conservative assumptions about the factors that contribute to the final result. This conservatism compensates for the lack of applicable data to the situation being analyzed. ORNL estimates the probability of DBA-2 at 3x10E-5 per year. In a separate analysis, SAIC concluded that a credible range for this estimate was 7.4x10E-4 to 3x10E-6 per year. Thus, the ORNL and SAI estimates are in agreement. Given the conservative assumptions of these analyses, the staff finds the above estimates to be an acceptable basis for evaluating the DBA-2 accident scenario.

Evaluation of Core Damage Frequency Given a DBA-2 Event

The DBA-2 event is only significant from an accident mitigation viewpoint if the plant cannot recover without fuel (core) damage. By definition of the event, there is no mitigation of the initial depressurization (blowdown). Following the initial depressurization, the operators must begin forced circulation cooling to protect the fuel from damage. SAIC independently estimated the probability that core cooling could not be achieved, given that DBA-2 had occurred. SAIC estimated that core cooling could not be provided at 2.5x10E-3 per demand. The staff reviewed the detailed calculations on which the SAIC estimates were based. This included the event tree, the fault tree, and the data used in the analysis. The staff has the following observations:

(1) The failure of several valves in the emergency condensate line to change positions is not modeled in the fault tree.

(2) The fault tree is not sufficiently detailed in some areas, such as on page A-20 of the SAIC report. Inadequate net positive suction head could be entered with an OR gate to the top event on that page.

(3) It is not clear if the support system dependencies were considered in the SAIC analysis. However, a review of common support systems revealed no obvious dependencies or common failure modes.

The failure rate data used to quantify these events compared favorably with the Integrated Reliability Evaluation Program data base, i.e., the data was more conservative. The staff believes that this conservatism offsets the above identified deficiencies. The staff finds that the SAIC calculations are sufficiently accurate to serve as a conservative and credible estimate of core damage frequency given a DBA-2 event.

Based on the above estimates, the probability of DBA-2 followed by not being able to cool the fuel is estimated at 7.5x10E-8 per year. This failure rate is very low when compared to other reactor accident scenarios (sequences). ORNL and SAIC have also translated these results into dose levels to the public. The conclusion drawn from these calculations is that the current level of protection provided against the consequences of DBA-2 is adequate.

Reduced Cooling Scenarios

The staff also requested that ORNL perform an analysis of the class 1E core cooling system's ability to respond to the DBA-2 accident scenario. ORNL had already performed evaluations of these systems and their capability to remove decay heat following other accident sequences. In these other evaluations the reactor was assumed to remain pressurized. In general, the staff has found that evaluations by ORNL and the licensee agree closely.

The calculations also agree favorably with the actual data observed in the FSV reactor. Hence, the staff concluded that the ORNL model will accurately reflect the plant's response to DBA-2.

ORNL calculated the plant response to DBA-2 at several power levels. Decay heat was removed only by the class 1E systems. This consists of a single helium circulator powered by a single condensate or firewater pump. This system is capable of operating with onsite power. ORNL concluded that even at 82 percent power, less than one percent of the fuel would be damaged following DBA-2. (82 percent power is the current FSV operating limit.) It should be noted that a one percent failure rate is small. The FSAR assumes a fuel failure rate of 10 percent over the full cycle life of the fuel (FSAR 3.7.4.1.2). Thus, the staff concludes with current plant power limits the class 1E systems can also provide adequate decay heat removal following DBA-2.

The staff notes that these calculations demonstrate that the level of protection afforded for DBA-2 at 82 percent power is equivalent to protection for light water reactors (LWR). Specifically, an LWR would have two full trains of class 1E systems to mitigate a major failure, such as a loss of coolant accident. At FSV, this level of protection is provided up to the current operating limit of 82 percent of full power.

3.0 CONCLUSION

The staff concluded that the probability of DBA-2 is not as low as assumed by the licensee. However, the more conservative estimate provided by ORNL is still low. The probability of DBA-2 occurring and being followed by fuel damage is negligible. Additionally, the class 1E systems are adequate to limit fuel damage within the reactor's current maximum power level restrictions.

Therefore, the staff concludes that no changes are required to the Technical Specifications to assure additional redundancy to respond the DBA-2. The staff concludes that the licensee's proposals are acceptable as proposed in the Technical Specification Upgrade Program.

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