UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

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December 15, 1978 CRESWELL

MEMORANDUM FOR: J. F. Streeter, Chief, Nuclear Support Section 1

FROM: J. S. Creswell, Reactor Inspector

SUBJECT: ANALYSIS SUPPORTING CONTINUED OPERATION OF DAVIS-BESSE NUCLEAR POWER STATION UNIT 1 WITH DUAL LEVEL SETPOINT CONTROL OF THE STEAM GENERATORS

I have read the subject analysis and have the following comments on statements made in the analysis:

1. Statement: This (35") level was chosen to provide adequate natural circulation capability and additional margin for the maintenance of indicated pressurizer level.

   Comment: There are level requirements for other than decay heat removal, e.g., during transient period after trip heat removal is required for stored energy in the system.

2. Statement: Since November 1977, the Davis-Besse Unit 1 operators have successfully manually controlled the steam generator levels on each occasion that auxiliary feedwater was automatically started by the steam and feedwater rupture control system (SFRCS).

   Comment: Since the plant startup, operators have had procedural instructions to control AFW level to control plant cooldown. Review of the November, 1977 event reveals that the level went to 120" after AFW was initiated. The November event was characterized by a great deal of stress on the operators and an event of this magnitude has not occurred in the last year. The point is whether operator action can be relied upon under stressful conditions.

   We also note that on April 29, 1978, an event occurred which results in a SFAS actuation when RCS pressure decreased below 1600 psi.

3. Statement: Manual control does not reduce the capability of each auxiliary feedwater pump (AFP) to deliver 800 gpm to each SG. At a 35" indicated level, an average total flow capability of less than 400 gpm per SG with both SG's in service is required to remove the decay heat from the RCS.

   Comment: FSAR questions and answers concerning Section 15.1.7 (4/18/75) states:

   The auxiliary feedwater system is designed to remove decay heat if the main feedwater system fails or is isolated from the steam generators. The auxiliary feedwater design requirements for a loss of main feedwater transient are more severe than the auxiliary feedwater requirements following other transients such as LOCA's or steam line breaks.

   An auxiliary feedwater flowrate of 800 gpm is required within 40 seconds of a loss of all main feedwater to prevent rupturing the pressurizer quench tank rupture disc due to excessive pressurizer relief valve discharge and to achieve a smooth transition to natural circulation if the four reactor coolant pumps lose motive power. The auxiliary feedwater system design must meet this requirement assuming a loss of normal electrical power and a single active failure.

4. Statement: The analysis shows that for a reactor trip from 15% of full power with the loss of all reactor coolant pumps and main feedwater pumps...

   Comment: An analysis should be performed under the following conditions:

   1. Reactor power 15%
   2. Loss of main feed pumps
   3. The reactor coolant pumps do not trip (no loss of offsite power)
   4. Level at 120" and 35"

   In addition, we note that the analysis was performed with the following limitations:

   a. Tcold went off scale during the November 29 event so that this data was not available.

b. The amount of makeup flow experienced during the November 29 event or other events has to be estimated since the flow can only be measured for values less than 160 gpm and flow can go to at least twice this value.

c. The loop 1 hot leg temperature was not available for this event.

d. The B&W analysis "The Effect of Net Makeup Flow After Reactor Trip on Pressurizer Level of Davis-Besse 1", dated October 6, 1978, recommends that the method for correcting pressurizer level not be treated as an accurate or reliable technique for determining true minimum pressurizer level during any future reactor trip transients.

In summary, the allowance of any administrative controls over auxiliary feedwater should be taken advisedly considering operating history.

J. S. Creswell
Reactor Inspector

cc: G. Fiorelli
T. N. Tambling
R. C. Knop