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APR 8 1977

Docket No. 50-346

MEMORANDUM FOR: D. B. Vassallo, Assistant Director for LWRs, DPM

FROM: D. F. Ross, Jr., Assistant Director for Reactor Safety, DSS

SUBJECT: DAVIS BESSE UNIT NO. 1 SER OPEN ITEMS FOR REACTOR SYSTEMS BRANCH

Reactor Systems Branch has evaluated the applicant's response to the open items identified in the staff's December 1976 SER on Davis Besse Unit No. 1. The enclosed conclusions provide information for your use in the next SER Supplement. Please note that further work is needed on the following:

4.4 ROD BOW

5.2.2 OVERPRESSURE DURING STARTUP OR SHUTDOWN - Applicant must modify setpoints to ensure DHR relief valve would actuate prior to automatic closure of isolation valves. Acceptable long-term fix is required before start of 2nd fuel cycle. Additionally, first fuel cycle operation is contingent upon a favorable response of the Materials Engineering Branch to a memorandum from T. Novak dated March 17, 1977.

5.5.3 DHR ISOLATION - New proposal to be submitted by applicant on locking out power, and on high pressure-low pressure isolation.

15.2.2 FEEDWATER STOP VALVE CLOSURE TIME - Further information required.

The additional requirements under Sections 5.2.2 and 15.2.2 are of short-term interest (prior to power operation) in granting TECO an Operating License. With no further information from the applicant, we would require that the closure test of the feedwater stop valves be changed from 16 seconds to 14.5 seconds in the Station Technical Specifications.

Original Signed By
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for
D. F. Ross, Jr., Assistant Director
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Davis Besse Open Items

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SER OPEN ITEMS

FOR DAVIS BESSE UNIT NO. 1

The following discussions relate to the identified section in the staff SER for Davis Besse Unit No. 1 dated December 1976.

4.4 Thermal and Hydraulic Design

Rod bowing was identified by the staff as a matter for which operational penalties can be imposed if necessary. In Reference 1, the NRC staff requested the applicant to evaluate a rod bow penalty for Davis Besse Unit No. 1 and to revise the Station Technical Specifications to accommodate any rod bow penalty. The applicant responded in Reference 2 with a revision to his Technical Specifications. A review of this submittal was performed by the staff and a determination was made that the applicant's evaluation of the amount of reduction in DNBR to account for rod bow was not adequate. The applicant stated that the amount of rod bow which the proposed Technical Specification is based upon, including credit for thermal margins, is 5.9%. Staff calculations indicate that this amount of rod bow is predicted to be exceeded after a burnup of 5651 $\frac{MWD}{MT}$. If the B&W proposed rod bow model has not been approved by the Regulatory Staff upon completion of 100 effective full power days of operation, then the Technical Specifications must be revised to reflect the rod bow model used by the staff for B&W plants. The staff model yields the following DNBR penalties as a function of burnup:

Burnup (MWD/MTU)	DNBR Penalty
0-15,000	8.2%
15,000-24,000	9.8%
24,000-33,000	11.2%

The staff also requested that the applicant implement a program of inspection and test of the core internal vent valves. Surveillance requirement 4.4.10.1b provides such a test and is acceptable to the staff. Also, we will require that reports to the NRC be made should any loose parts monitoring anomalies be attributed to a vibrating vent valve or vent valve components.

5.2.2 Overpressure Protection

The staff evaluated incidents known as pressure transients (events that have exceeded the Technical Specification temperature-pressure (P-T) limits of the reactor vessel) and issued a technical report in November 1976. (Reference 3). The report concluded in part that pressure transients are a special concern during plant startup and shutdown because, at these relatively low temperatures, the vessel has less toughness than at operating temperatures. Irradiation increases the temperature at which steel attains increased toughness. The Appendix G limits change during the life of the vessel as it becomes irradiated, and because it would be impractical to change these limits continuously, they are calculated for discrete periods of time. Thus, the P-T limits in effect at a given time may be based on properties expected in the vessel five or more years in the future, making them conservative during the early portion of this period. The report concluded that large safety margins to failure exist for unirradiated reactor vessels, and new plants can be permitted to be licensed under existing safety criteria. Nevertheless, the staff concluded that administrative procedures and overpressure protection devices should be upgraded to reduce the likelihood of pressure transient events.

On December 7, 1976, the applicant submitted an analysis to show compliance with Appendix G pressure-temperature limits during startup and shutdown.

(Reference 4). The staff reviewed this submittal and requested further information from the applicant. The applicant responded on February 18, 1977 (Reference 5) with a discussion which provides further assurance that Appendix G limits would not be violated. Additionally, the staff requires that the applicant make a modification which ensures that the DHR relief valve would actuate prior to automatic closure of the isolation valves. This change would allow the relief valve to be available for mitigating the consequences of an overpressure event.

While the above means are acceptable to the staff to minimize the likelihood of exceeding Appendix G limits for the first fuel cycle, additional means must be incorporated prior to the start of the second fuel cycle to further reduce the potential for exceeding Appendix G limits. The applicant has proposed a long-term solution which is under review. We will condition the operating license to require that the licensee implement, prior to the end of the first regularly scheduled refueling outage, a long-term means of overpressure protection that is acceptable to the staff.
[NOTE TO DPM: See Cover Letter Comment on Section 5.2.2.]

5.5.3 Decay Heat Removal System (DHRS)

The staff required the applicant to address the potential for and consequences of an inadvertent closure of a DHR isolation valve during shutdown operations, which would lead to interruption of flow to both DHR pumps with possible pump damage due to cavitation. The applicant has proposed removing power from the two series DHR valves during shutdown operations. While this procedure would reduce the problem of an inadvertent valve closure, we conclude that this proposal could compromise the barrier between the high and low pressure piping by increasing the potential for the plant starting up with only one valve closed. With normal power available to a DHR valve which could be inadvertently left open by the operator, an automatic closure feature now provides backup to this postulated operator error when primary side pressure was increased. With normal power not available to a DHR valve inadvertently left open by the operator, the applicant has stated that sufficient alarms and visual indications would be available to the operator to alert him to take a corrective action; however, no backup automatic closure exists and the plant could continue at power operation with only one barrier available between the high pressure and low pressure piping.

We are continuing to pursue final resolution of this concern with the applicant. In the interim, the power should be maintained to the DHR valves. In addition, to ensure the availability of at least one train of the DHR system, TECO should consider one or more of the following design or operating changes:

1. auto-pump trip on valve closure (each valve tripping only one pump, within division);
2. run one train at a time; and,
3. add pump trips from low flow, pump discharge pressure, hi-temp in mini-flow bypass line, etc.

With regard to the bypass loop containing two manually operated valves around the DHR suction line isolation valves, ACRS in Reference 6 has stated that further attention should be given to the means employed for isolation of the low pressure RHR system from the primary system while the latter is pressurized, and that reliable means be developed to assure such isolation. The staff notes that administrative controls on the manual bypass valves DH21 and DH23 have been changed to require a key to open their normally locked-closed status. Nevertheless, it is the staff's judgment that additional means are necessary to further minimize the potential for inadvertent opening of the bypass valves during high pressure operation. Discussions have taken place with the applicant with regard to a flange of spectacle shape which could be installed between the two bypass valves. Such a spectacle flange would further decrease the potential for the bypass path being opened at the wrong time, yet still retain the capability of maintaining decay heat capability should one of the DHR suction line valves fail in a closed position. The staff will pursue final resolution of this matter with the applicant and will require that modifications be made as soon as is practicable, but no later than the first refueling outage.

We will require a suitable reliability study for the present design as contrasted with a spectrum of hypothesized design variations, such as the spectacle flange concept, to assure that:

1. any change is for the better; and,
2. the final system is safe enough.

6.3.2 ECCS Design

With regard to leak-testing of the check valves in the high pressure to low pressure interface of the LPI discharge line, the applicant has committed to periodically verifying valve integrity per Station Technical Specification 4.05. This surveillance requirement performed on valves CF-28, CF-29, CF-30, CF-31, DH-76 and DH-77 at least each refueling outage in accordance with Reference 7 is acceptable to the staff.

Also the staff requested that the applicant adopt a surveillance requirement in the Station Technical Specifications to verify that the ECCS piping is water solid to minimize the potential for water hammer. Technical Specification 3/4 5-4 provides this requirement and is acceptable to the staff.

6.3.3 ECCS Performance

(See separate staff Safety Evaluation Report)

6.3.4 Tests and Inspections

The NRC staff requested that, prior to issuance of an operating license, a test be conducted at ambient conditions to verify the capability of the ECCS to operate in the recirculation mode. The applicant has completed confirmation testing of the ECCS to operate in the recirculation mode (Reference 8). Head loss data gathered on-site for a flow rate from the containment sump equivalent to the maximum capability of one train were compared to predicted values. The predicted values were shown to be conservative head loss estimates. An investigation of the potential for the formation of vortices in the containment sump was conducted using a 1:2 scale model off-site. Additional grating was installed in Davis-Besse 1 subsequent to the testing to provide additional assurance that unacceptable vortex formation would not occur. The staff concludes that the ECCS containment sump as designed should be free of unacceptable vortices and that adequate NPSH exists to assure that the system will operate as intended.

6.3.5 Conclusions

See preceding discussions of sections 6.3.2, 6.3.3 and 6.3.4.

15.2.2 Accidents

The staff requested that the closure times of steam and feedwater isolation valves assumed in the accident analyses be periodically verified. The proposed Station Technical Specifications were reviewed by the staff and, except for the feedwater stop valve closure time of 16 seconds, are acceptable to the staff. Since the feedwater line break assumed a closure time of 14.5 seconds, further information is required to justify the existing Technical Specification for this valve.

Also, for the main steam line break and feedwater line break, the staff requested that the applicant further examine the potential for single active component failures, such as an isolation valve failure or the opening of an atmosphere vent valve. Additional information was submitted by the applicant and reviewed by the staff. It is the staff's position that the modified description and analyses submitted in the FSAR in conjunction with the responses to staff questions provide acceptable evaluations of these events.

References

1. Letter from John F. Stolz to Mr. Lowell E. Roe dated January 21, 1977.
2. Letter from Lowell E. Roe to Mr. John F. Stolz dated February 8, 1977.
3. NUREG-0138, "Staff Discussion of Fifteen Technical Issues listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff," November 1976.
4. Letter from Lowell E. Roe to Mr. John F. Stolz dated December 7, 1976.
5. Letter from Lowell E. Roe to Mr. John F. Stolz dated February 18, 1976.
6. Letter from M. Bender to Honorable Marcus A. Rowden dated January 14, 1977.
7. Applicant's response to Regulatory Position 6.3.2 dated April 18, 1975.
8. "Final Report on the Davis-Besse Nuclear Power Station Unit 1 ECCS Emergency Sump and Pump Suction Line Testing," December 15, 1976.