

OCT 21 1981



Docket Nos.: 50-329/330

Mr. J. W. Cook
Vice President
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. Cook:

SUBJECT: TRANSMITTAL OF PRELIMINARY SER DRAFT SECTION FOR SELECTED
NUREG-0737 ITEMS, MIDLAND PLANT, UNITS 1 AND 2

Enclosed for your review and comment are preliminary draft sections of the NRC Staff's Safety Evaluation Report for Midland Plant, Units 1 and 2 addressing the following items from NUREG-0737:

<u>ITEM</u>	<u>STATUS</u> (See Key Below)
II.B.1 (vents)	1
II.K.2.13 (thermal mechanical report)	2
II.K.2.15 (slug flow)	2
II.K.2.16/II.K.3.40 (seal damage)	2
II.K.2.17 (voiding)	3
II.K.2.19 (seq. aux. flow)	3
II.K.3.1 (auto. block valve)	1
II.K.3.3 (valve failure reporting)	2
II.K.3.5 (auto. R.C. pump trip)	4
II.K.3.30/II.K.3.31 (S.B. LUCA)	4

Key:

1. Additional information required of applicant (see attached list).
2. Applicant has met NRC requirements.
3. Applicant should document applicability of material already filed by licensees of B&W plants (see attached list).
4. The applicant is participating with B&W owners for the generic review of these items and will be required to implement the staff requirements from these reviews.

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Your attention is directed in particular to any open items contained within these draft sections. A principal objective of this transmittal is to provide for timely identification and resolution of any additional analysis, missing information, clarifications or other work necessary to resolve outstanding issues. Please contact the staff's Project Manager regarding the need for any meetings and telephone conferences to this end.

Our review of Item 11.B.1 has utilized the guidance of a proposed Standard Review Plan Section 5.4.12, "Reactor Coolant System High Point Vents". A copy is enclosed.

Your comments, including schedules for completion of any further analyses or other work associated with resolution of open items, are requested within two weeks of this letter.

Sincerely,

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
of Licensing
Division of Licensing

Enclosure: As stated
cc: See next page

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MIDLAND

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OPEN ISSUES

NUREG-0737

1. II.B.1 - REACTOR COOLANT SYSTEM HIGH POINT VENTS

- a) Provide reference to the codes and standards to which the reactor coolant system vents will be designed. Discuss the seismic and environmental qualification and provide appropriate justification.
- b) Discuss independence of power supplies for the vent lines valves.
- c) Provide the procedures that will be used in operational testing of the vents.
- d) Provide the operational procedures for the vents and supporting analyses including venting of gas which might be present in the reactor vessel head.
- e) Since the vents exhaust directly into the containment atmosphere, provide information showing that the proper mixing will be obtained.
- f) Demonstrate that LOCA analyses already performed for Midland bound the cases of inadvertent vent operation or vent line severance or provide new analyses.

2. II.K.3.1 - INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

- a) Provide and justify the values chosen for block valve closure setpoints.
- b) Justify that the signals and power supplies to the redundant block valves are independent.
- c) Discuss the testing program needed to demonstrate operability of the system including justification that the block valves can close under full flow conditions to isolate a stuck open PORV.

3. Document the applicability of evaluations filed by B&W licensees in response to TMI issues II.K.2.17 and II.K.2.19 or provide appropriate evaluations for Midland.

RSB SAFETY EVALUATION
OF
NUREG-0737 TMI-ISSUES

II.B.1 REACTOR COOLANT SYSTEM VENTS

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point system:

- (1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.
- (2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Discussion and Evaluation

The Midland plants have the capability for venting the primary system at three locations. These are at the top U-bends of the two hot legs and the upper head of the pressurizer. One inch vents paths are provided with redundant valves which are operated manually from the control room and are powered from Class IE power supplies. The valves fail closed upon loss of power. Each vent path discharges into the containment atmosphere.

In the event that the reactor coolant pumps are not in operation, heat is removed from the primary system by natural circulation which causes heat generated by the core to be transferred to the steam generators where it is removed. Natural circulation is produced by the unbalanced gravitational forces in the coolant loops between the heated section containing the core and the sections that are cooled by the steam generators. Analyses performed by the applicant have indicated that gas or steam accumulation in the hot legs up to 340 ft³ would not prevent the occurrence of natural circulation by adversely affecting the gravitational forces in the coolant loops.

Steam might form in the reactor coolant loops from coolant flashing and pressurizer drainage in a severe overcooling transient. Steam would also be produced in the reactor coolant loops in the event of a loss-of-coolant accident, and hydrogen from metal water reactor reaction might be generated in the event that the core was inadequately cooled. Analyses of potential overcooling transients by the applicant and the staff have demonstrated that sufficient void formation to block natural circulation will not occur at the Midland Plants. See NUREG-0660 Item II.E.5.1 of Appendix ____* of this SER. Following a LOCA, heat removal from the primary system by natural circulation would be required only for small

*Appendix number to be supplied by LPM.

break sizes of .01 ft² and less. For these break sizes the applicant's LOCA analyses, discussed in Section 6.3 of this SER, demonstrated that, although natural circulation could be temporarily lost, it would be reestablished in a boiling condensation mode before the core could become uncovered. The core was shown to be adequately cooled so that no noncondensable gas from metal-water reaction would be produced for small breaks.

Although analyses of potential design basis transients and accidents do not predict a need for the high point vents, the system provides additional defense in depth to ensure that if natural circulation is lost from steam or gas formation in the reactor coolant loops it can be reestablished manually from the control room.

We have reviewed the proposed High Point vent system for compliance with the recommendations of NUREG-0737 Item II.3.1 and the proposed Standard Review Plan 5.4.12 for Reactor Coolant System High Point vents, and have requested that the applicant submit additional information. We will report our conclusions in a Supplement to this SER.

II.K.2.13 THERMAL MECHANICAL REPORT - EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

DISCUSSION AND EVALUATION

The applicant has referenced a report evaluating the thermal stresses which might occur within the reactor vessel in the event of a small break LOCA and a complete loss of feedwater (BAW-1628). For this condition the reactor core would be cooled by high pressure injection flowing through the core and out the pressurizer relief or safety valves, i.e., feed and bleed. This mode of operation would be required for potential break sizes of 0.01 ft² and smaller in the event all feedwater were lost. Larger break sizes would discharge sufficient steam so that the reactor system would depressurize whether or not feedwater were available. The report indicates that reactor vessel damage might occur from prolonged feed and bleed operation and that the possibility of damage would be lessened by operator action to throttle HPI flow to keep the reactor coolant from going below 100°F subcooled at the core outlet.

Since the auxiliary feedwater system at the Midland plant is redundant and designed to safety grade criteria (See SER Section 10.4), a small break LOCA accompanied with a complete loss of feedwater is not a design basis event and we consider this item to be resolved for Midland. However, the NRC staff is pursuing a long term evaluation of low temperature thermal shock for all reactor types, and Midland will be evaluated against any new staff requirements developed from this study.

II.K.2.15 EFFECTS OF SLUG FLOW ON STEAM GENERATOR TUBES

Position

Although the staff believes that the potential for slug flow was not great in Babcock and Wilcox (B&W) plants because of the venting path provided by the internal vent valves, the staff required that a confirmatory evaluation of the effects of slug flow on steam generator tubes be performed by the licensees to assure that the tubes could withstand any mechanical loading which could result from slug flow.

Discussion and Evaluation

This issue requests applicants to investigate the loading on steam generator tube sheets induced by slug flow during natural circulation. In order to bound the magnitude of the loading on the steam generator tubes during natural circulation, the applicant assumed that the slug traveled at a velocity of 64.4 ft/sec. This corresponds to full power, forced flow operation, and bounds any pressure differential which could develop under natural circulation. By use of the dynamic momentum equation, the force on the upper tube sheet was evaluated to be an order of magnitude lower than the theoretical buckling load of 800 lb_f.

Based on this analysis, the staff concludes that slug flow induced by natural circulation will not result in adverse loading on the steam generator tubes. As such, we consider Action Item II.K.2.15 resolved.

II.K.2.16/II.K.3.40 REACTOR COOLANT PUMP SEAL DAMAGE

Position

Evaluate the impact of reactor coolant pump seal damage and leakage due to loss-of-seal cooling upon loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small-break loss-of-coolant accident (LOCA) with subsequent reactor coolant pump (RCP) seal damage.

(The requirements of II.K.3.40 have been incorporated into II.K.2.16).

DISCUSSION AND EVALUATION

The reactor coolant pumps for the Midland units are single stage, centrifugal pumps manufactured by Byron Jackson. The pump seals are cooled by water supplied by two safety grade systems which are powered from emergency power buses. Seal injection is supplied from the High Pressure Injection Pumps and seal cooling is supplied by the safety grade portion of the Component Coolant Water System.

During loss of offsite power the seal injection coolant would not be interrupted long enough to cause seal damage and no damage to the reactor coolant pump seal will occur. On this basis we conclude that the licensee adequately addressed our concerns of Item II.K.2.16 of NUREG-0737 and the requirements of this item have been met.

II.K.2.17 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

Position

Analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

DISCUSSION AND EVALUATION

NUREG-0737 requires that all PWR operating reactors and applicants analyze the potential for void formation in the primary system during anticipated transients.

The request evolved from a staff concern that unanticipated primary system void formation during anticipated depressurization transients could degrade system performance by either, (1) "holding up" the primary pressure above the HPI actuation setpoint and/or, (2) trapping steam at the top of the hot leg "candy-cane" bends, thereby impeding natural circulation.

Decrease in reactor system pressure during reactor trip events (without secondary system malfunctions) is limited by the balance of energy removal rate versus the energy insertion rate. The energy insertion rate primarily consists of core (decay) heat, reactor coolant pump heat, and metal heat. The energy removal rate from the primary system consists of heat transfer, corresponding to turbine trip conditions in the secondary side of the steam generators and the metal heat transfer to containment. This balance of insurge and outsurge of energy limits the primary system coolant shrinkage and corresponding expansion of the pressurizer steam volume. For B&W plant designs, the energy balance following a reactor trip typically results in a minimum system pressure of 1750 psia.

The owners of operating B&W designed plants examined mild overcooling operational transients at Davis Besse and Oconee Units I and II. These events significantly drained, but did not empty the pressurizer. Their evaluation concluded that

steam production in the reactor coolant loops was highly improbable for these events.

In analyses performed to determine sensitivity to overcooling transients, the applicant evaluated an unmitigated, large (12.2 ft²) steam line break. For this case, which can be considered a bounding overcooling case, steam production caused natural circulation flow to be temporarily interrupted in the loop containing the pressurizer. The loop without the pressurizer was predicted to continuously remove decay heat. These calculations are discussed in Item II.E.5.1 of Appendix _____* of this SER.

Based on our review of the information provided by licensees of operating plants in response to Item II.K.2.17, and analyses of primary system void formation during limiting events, we have concluded that:

- (1) Primary system void formation is not predicted to occur for operational transients which do not deplete the pressurizer liquid inventory. Evaluation of operational data supports this conclusion.
- (2) Limiting overcooling transients (non-LOCA) may result in temporary void formation in the primary system. This voiding is accounted for in present analysis models and does not result in unacceptable consequences.

The licensees of B&W plants have demonstrated the adequacy of their plant design to accommodate anticipated operational depressurization transients, as described above.

*Appendix number to be supplied by LPM

We require that the applicant confirm the applicability of the above referenced plant operating data to Midland.

II.K.2.19 SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS

Position

Provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater.

DISCUSSION AND EVALUATION

NUREG-0737 requires that all operating pressurized water reactor licensees and applicants benchmark their analytical methods against experimental data of sequential auxiliary feedwater flow to the steam generators.

At a meeting in Bethesda, April 26, 1979, with the owners of Babcock and Wilcox (B&W) reactor plants, we requested a benchmark analysis of sequential auxiliary feedwater flow to the steam generators following a loss of main feedwater. This analysis was provided in a letter from J. Taylor (B&W) to R. Mattson (NRC) dated June 15, 1979, utilizing a sequential auxiliary feedwater flow event at Crystal River Unit 3 as a benchmark. In this analysis, the TRAP-2 Code with a 6-node steam generator model was utilized. All small break analysis presented to the NRC have been performed using the CRAFT-2 Code with a 3-node steam generator model. By letter dated August 21, 1979, we required a benchmark analysis for sequential auxiliary feedwater flow using CRAFT-2 with a 3-node steam generator representation. Each licensee of B&W reactor plants responded with a report which presented analysis of the sequential auxiliary feedwater flow event at Crystal River using the CRAFT-2 Code. This issue was later identified as Item II.K.2.19 of the TMI Action Plan requirements.

The results of this data comparison led to further examination of the Crystal River Unit 3 event which indicated the data to be inappropriate for assessing computer codes. This is attributed to inadequate instrumentation whereby key data required for code assessment were not obtained.

Reviews conducted by our B&O Task Force, following the TMI-2 accident, have concluded that further assessment of the CRAFT-2 Code would be required. The majority of the concerns identified are documented in NUREG-0565. In particular, the neglect of a mechanistic, regime-dependent heat transfer model and the use of a constant, steam generator heat transfer coefficient throughout the transient have been identified as requiring either revision or further justification. This requirement for further justification and/or revision of the small break ECCS model is being performed under TMI Action Plan Item II.K.3.30. We believe that satisfactory resolution of code modeling concerns as part of the Action Item II.K.3.30 will resolve the modeling concerns of II.K.2.19.

The conclusions of our review of Action Item II.K.2.19 for operating B&W plants are as follows:

- (a) The intent of Item II.K.2.19 was accomplished;
- (b) Results provided by CRAFT-2 were similar to those provided by the more detailed TRAP-2 program. However, both codes showed poor agreement when compared with the test data.
- (c) The poor agreement of the code prediction with test data has been attributed to the fact that the Crystal River data was not adequate for assessing thermal-hydraulic codes; and
- (d) A more rigorous assessment of the B&W small break LOCA model is being performed under TMI Action Item II.K.3.30. Further code assessment under TMI Action Item II.K.2.19 is therefore unnecessary.

We require that the applicant confirm the applicability of the above referenced analysis to Midland.

II.K.3.1 INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to assure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

DISCUSSION AND EVALUATION

The Midland plants will be protected against the possibility of a stuck open PORV by two upstream Class 1E block valves. The PORV and block valves are Class A as defined by Regulatory Guide 1.26 and as such are designed fabricated, tested, installed and certified by the requirements of the ASME Boiler and Pressure vessel code Section III for Class 1 valves. The block valves are designed to close automatically upon coincidence of low reactor system pressure and the PORV open. The automatic closure feature is manually overridden to provide low temperature overprotection as discussed in Section 5.2.2 of this SER. Although the applicant has committed to meet the requirements of Item II.K.3.1, we require additional information on the block valve setpoint, independence of closure signals and power supplies and the preoperational testing program in order to complete our review. We will report our conclusion in a Supplement to this SER.

II.K.3.2 PORV AND SAFETY VALVE REPORTING REQUIREMENTS

Position:

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

This requirement shall be met before issuance of a full-power license.

DISCUSSION AND EVALUATION

The applicant has agreed to meet these reporting requirements which have been incorporated in the plant Technical Specifications. The NRC staff review of Technical Specifications is contained in Chapter 16 of this SER.

II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

DISCUSSION AND EVALUATION

The applicant proposes to trip the reactor coolant pumps automatically on a low reactor coolant system pressure signal. The design of the automatic trip system is being determined in part from analytical benchmark calculations of the LOFT L3-6 small break test. These calculations are described in the B&W topical report "B&W's Best Estimate Prediction of the LOFT L3-6 Nuclear Small Break Test using the CRAFT-2 Computer Code, Document No. 12-1124993, March 1981". The report is currently under review by the NRC staff. We will report on the resolution of this issue in a supplement to this SER.

II.K.3.30 REVISED SMALL-BREAK LOSS-OF-COOLANT ACCIDENT METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Changes to Previous Requirements and Guidance

The change requirement (1) allows for justification of acceptability of present small-break LOCA models by comparison with test data, and (2) requests each licensee to outline scope and schedule for model revision or comparison with test data by late fall 1980. The original requirement did not allow provision for showing acceptability of present models by comparison with plant data.

II.K.3.31 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR PART 50.46

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in Item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

DISCUSSION AND EVALUATION (II.K.3.30 and II.K.3.31)

Although Consumers Power Company believes that the current small-break LOCA model, discussed in Section 6.3 of this SER, is in full compliance with the commission's regulations, they are participating with other owners of B&W reactors in a verification program to provide improvement to certain features of the model and to pro-

vide comparison with experimental data including LOFT and operating plant transients. The staff has had several meetings with the owners of B&W reactors on this issue and we believe that reasonable progress is being made toward its resolution.

The NRC will determine the need for new small-break LOCA calculations for Midland under II.K.3.31 at the conclusion of the model review in progress under II.K.3.30.