TOLEDO EDISON

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Docket No. 50-346
License No. NPF-3
Serial No. 717

May 22, 1981

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stolz:

This is in response to your letter dated April 2, 1981 (Log No. 687) relating to the Auxiliary Feedwater System reliability analysis evaluation for Davis-Besse Nuclear Power Station Unit 1. Enclosure 1 to your letter listed six items requiring Toledo Edison response. Attachment 1 to this letter summarizes our response to items 1 through 4 and 6. Attachment 2 provides our response to item 5.

Very truly yours,

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Attachments

cc:
DB-1 NRC Resident Inspector

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Attachment 1 to Toledo Edison letter to the NRC on
Auxiliary Feedwater System Reliability Analysis Evaluations

Item 1. Technical Specification Administrative Control of Manual Valves - Lock and Verify Position

The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications.

Response: All manual valves in the suction and discharge of the auxiliary feedwater (AFW) pumps at Davis-Besse 1 (DB-1) are locked and administratively controlled per the existing administrative procedure AD 1839.02, "Operation and Control of Locked Valves". This procedure also requires independent reverification of restoration of a valve position if the position was changed. In addition, PT 5186.01, "Locked Valve Verification Periodic Test" ensures, on a monthly basis, that these valves are locked in their correct position. The above controls provide adequate assurance that the manual valves are not inadvertently positioned to interrupt AFW flow to the steam generators. As a previous commitment (Toledo Edison Letter, Serial No. 1-56 dated April 11, 1979) and an approved station procedure, these controls are fully auditable and subject to inspection and enforcement action, if not complied with. Consequently, no changes to plant Technical Specification surveillance requirements are necessary.

Item 2. Local Manual Realignment of Valves

The Davis-Besse plant requires local manual realignment of valves to conduct periodic tests on one AFW system train and has only one remaining AFW train available for operation, therefore, the licensee should propose Technical Specifications to provide that a dedicated invididual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system from the test mode to its operational alignment.

Response: The valves that need manual realignment to conduct periodic tests on one AFW system train and could affect availability of the train are AF21 for train 1 (AF22 for train 2), AF23 and AF50 (or AF51). These valves are in series. See FSAR figures 10-5 and 10-6.

The testing is conducted by recirculating the AFW flow to the condensate storage tanks. The surveillance test ST 5071.01

"AFW System Monthly Test" requires that in modes 1, 2 and 3 an operator (located by the AFP to be tested) be in direct communication with the control room when AF23 is open. If the affected train is required to be operated as demanded by the Steam and Feedwater Rupture Control System, the control room immediately instructs the operator to close AF23. Since this valve is in series with the other above mentioned valves, closure of this valve makes the train available for feeding the steam generators by closing the path for AFW flow diversion. Thus this item is already covered by existing surveillance test requirements. It should be noted that only one train of AFWS is tested at a time. The redundant 100% capacity train is available for feeding AFW to the steam generators if needed. The above provides adequate resolution of your concern. No additional Technical Specifications are therefore proposed.

Item 3: AFW System Flow Path Verification

The licensee should confirm flow path availability of an AFW system train that has been out of service to perform periodic testing or maintenance as follows:

(1) Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

(2) The licensee should propose Technical Specifications to assure that, prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

Response: (1) The manual valves in the AFWS are controlled under AD 1839.02 (as stated in response to item 1 above) and are restored to their required position following periodic testing or maintenance. This procedure also requires independent reverification of any valve, whose position was changed during surveillance testing or maintenance. Following performance of maintenance on any AFW train causing inoperability of the entire train (e.g. maintenance on the pump, turbine or associated controls), the operability of entire train is demonstrated by performance of any one of the AFWS surveillance tests (ST 5071.01, 5071.03 or 5071.04). Following performance of maintenance on a portion of an AFW train (e.g. valve repacking, torque/limit switch repair), the surveillance test is performed to demonstrate the operability of the affected equipment only. All of the above surveillance tests ensure the restoration and independent verification of the affected valve: to their required position. Therefore Toledo Edison is in compliance with this item.

(2) Toledo Edison believes that requirements of a flow test prior to plant startup following an extended cold shutdown is restrictive and is not required, keeping in view the already stringent surveillance requirements and other administrative controls on the AFW System. We believe that a verification of valve positions and pump operability is adequate to assure the operability of the train. In addition, performance of a flow test has an adverse impact on the steam generator water chemistry. This may require additional time to restore the steam generator chemistry within acceptable limits. Therefore, such a flow test is not desirable.

Item 4: Flow Blockage by Plugged Strainers

The licensee should assure that there are no temporary strainers in place in the AFW piping system that may cause flow blockage if plugged. Operating experience at several plants has shown this to be a potential common cause failure mechanism which could fail the entire AFWS. The suction strainers between the condensate storage tank and the pumps are an example.

Response: There are no temporary strainers in the AFW suction piping at the present time that may cause flow blockage when plugged.

Item 5: Design Basis for AFW System Flow Requirements

The licensee is requested to provide the AFWS flow design basis information required in Enclosure 2 for the Davis-Besse 1 design basis transients and accident conditions.

Response: The design basis information required in Enclosure 2 of your letter for DB-1 design basis transients and accident conditions is provided in Attachment 2 to this letter.

Item 6: Diversity in the Motive Power for the AFWS Pumps

We are concerned with the dependency of both AFWS pumps on steam from the main steam lines. Other PWRs are known to have a similar configuration (e.g., Calvert Cliffs); however, because of the more rapid dry-out of the steam system in B&W plants, such a steam dependency is of more concern in Davis-Besse. The licensee should state plans for providing a third AFWS train which will utilize a pump powered from a source other than steam. A schedule of implementation should be provided.

Response: Following our meeting with your staff on March 5, 1981, we have initiated a detailed probabilistic risk assessment study on the AFW system. This study considers the Pre-TMI-2 DB-1 AFWS, Post TMI-2 AFWS and the Post TMI-2 AFWS with a third source of AFW consisting of the existing electric motor driven startup feed pump discharging into the AFW nozzles. This study is expected to be complete in late July 1981, and will develop the probabilities for system unavailabilities for the above configurations. This study will also identify dominant failure contributors to system unavailability. Based on the conclusions and recommendations of this study, upgrades to the existing AFW system and/or the startup feedwater pump will be planned. We will be willing to discuss our plans and schedule with you at the time that this study is completed.

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Attachment 2 to Toledo Edison Letter to the NRC on Auxiliary Feedwater System Reliability Analysis Evaluation

Item 1.a.

Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1) Loss of Main Feed (LMFW)
2) LMFW w/loss of offsite AC power
3) LMFW w/loss of onsite and offsite AC power
4) Plant cooldown
5) Turbine trip with and without bypass
6) Main steam isolation valve closure
7) Main feed line break
8) Main steam line break
9) Small break LOCA
10) Other transient or accident conditions not listed above

Response to Item 1.a.

The original design of the Auxiliary Feedwater System (AFWS) established a requirement for a minimum flow sufficient to remove heat load equal to about 5% full power. This flow rate was not based on any specific transient. However, where appropriate, this flow rate is used as part of the transient analysis for the accidents considered in the FSAR. Table 1 contains a list of those transients considered in the FSAR along with their acceptance criteria. A value of 800 gpm has been determined to be an acceptable flow for the AFW system taking into account a single failure. This value has been shown to be acceptable for a loss of main feedwater (LMFW) event from 112% full power (including instrument error allowance). This event requires the maximum heat removal capacity (including RC pump heat) for the AFW system, therefore, is considered the design basis event. For the above parameters and assuming 1.2 times the ANS 5.1 decay heat, analysis showed that the primary acceptance criteria were met (with a reactor trip on high RC pressure). These criteria are:

- RCS peak pressure less than 110%
- DNBR greater than minimum allowable
- Site boundary doses less than 10CFR100 limits

In addition to meeting these criteria, with a minimum AFW flow of 800 gpm, the pressurizer did not go solid.

Accidents 7, 8, 12 and 13 of Table 1 specifically require AFW for mitigation. As reported in the FSAR, the results are acceptable with an 800 gpm AFW flow. The other accidents listed in Table 1 do not require AFW for mitigation although the availability of the AFWS is assumed.

The other events listed in the questions but not included in Table 1 are discussed below.

Loss of Onsite and Offsite AC Power - This event is not a design basis for Davis-Besse 1 since it requires a failure of both emergency diesel generators. However, one train of AFWS is capable of supplying AFW to the steam generator even with loss of both onsite and offsite AC power.

Plant Cooldown - Plant cooldown with AFW and with a loss of offsite power is a controlled event with decay heat levels equal to or lower than the loss of feedwater event identified as the design basis event. The design basis event bounds this case for the AFW flow required.

Turbine Trip With and Without Bypass - This event does not affect the AFWS unless MFW fails. In which case, the loss of MFW event previously addressed would bound the AFWS design.

Main Steam Isolation Valve Closure - Again, this event does not directly affect the AFWS unless MFW is lost as discussed above.

Small Break LOCA - The AFW criteria assumed for this event are described in the B&W report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 FA Plant, Volume 3." This report was submitted to the NRC on May 22, 1979 with a letter serial No. 506 and demonstrated that an AFW flow of 800 gpm for Davis-Besse will not lead to the violation of the acceptance criteria.

Item 1.b.

Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating events identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCI, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cooldown the primary system

Response to Item 1.b.

The design basis event for sizing the AFWS is the Loss of Feedwater event discussed in the response to Item 1.a. The acceptance criteria for the other transients which assume the availability of AFW are given in Table 1.

The acceptance criteria for these accidents include RCS pressure limits, ensuring RC pressure boundary integrity, fuel limits and offsite dose limits. The RCS cooling rate is not an acceptance criterion for accident analyses. An overcooling event that drains the pressurizer is not desirable, however, it does not violate any of the accident analysis acceptance criteria.

Maintaining a minimum steam generator level is not an acceptance criterion for accident analyses. It is desirable that the reactor be tripped and AFW initiated prior to steam generator dryout, but this is not required in order to obtain acceptable results. After AFW has been initiated, the high injection point in the steam generator reduces system dependence on a specific level for adequate heat transfer. The steam generator level control is set low for decay heat removal with forced circulation and for natural circulation without a small break LOCA. The level is set high for small break LOCA event. For a detailed analysis of this dual level control, see Serial No. 475 dated 12/22/78 and Serial No. 471 dated 12/11/78.

TABLE 1

	Accident Description	FSAR Section	Acceptance Criteria
1.	Startup Accident	15.2.1	A,B
2.	Uncontrolled Control Rod Assembly Group Withdrawal at Power	15.2.2	A,B
3.	Control Rod Assembly Misalignment	15.2.3	A,B
4.	Makeup and Purification Malfunction	15.2.4	A,B,C
5.	Loss of Forced Reactor Coolant Flow	15.2.5	D,E
6.	Reactor Coolant Pump Startup Accident	15.2.6	A,B
7.	Loss of Normal Feedwater Due to Closure of Feedwater Valve, Pump Failure, or a Feedwater Line Break	15.2.8	B,E
8.	Loss of All AC Power (Station Blackout)	15.2.9	B,E
9.	Excessive Heat Removal Due to Feedwater System Malfunction	15.2.10	B,E
10.	Steam Generator Tube Rupture	15.4.2	F,G
11.	Control Rod Assembly Ejection Accident	15.4.3	F,G
12.	Steam Line Break	15.4.4	E,F,G
13.	Loss of Coolant Accident	15.4.6	F

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KEY	ACCEPTANCE CRITERIA
A	Reactor Thermal Power Less than 112% of Rated Power
B	Reactor Coolant Pressure Less than Code Pressure Limits (110% of Design Pressure)
C	Minimum Shutdown Margin of 1% k/k during Refueling Conditions
D	Minimum DNB Ratio Greater Than 1.3
E	Fuel Cladding Temperature Less than 2200°F
F	Resultant Doses Less than 10CFR 100 Limits
G	No Loss of Reactor Coolant Pressure Boundary Integrity

Item 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above.

Response to Item 2

As discussed in Response to Item 1.a., the design basis event which verifies the AFWS design flowrate is loss of main feedwater. The analysis assumptions for this event are listed below. Corresponding technical justification where not specifically listed, is based on licensing requirements and prudent engineering judgment at the time of the analysis. The information is not provided for the other events identified in Item 1.a. and Table 1 because the LMFW event is the most limiting.

The LMFW analysis used for this response has not been done for Davis-Besse although it has been done for the Sacramento Municipal Utility District's Rancho Seco plant. These two plants are very similar in design and performance, therefore, e results for the Rancho Seco analysis are also applicable to Davis-Besse, and the numbers given below are taken from that analysis. A comparison of the applicable parameters was made for these two palnts. The results of the comparison demonstrated that the Davis-Besse AFW system is adequate for cooling following a LMFW transient.

a) Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   - 112% full power (including instrument error allowance)

b) Time delay from the initiating event to reactor trip.

   - The reactor will trip on high reactor coolant pressure approximately 14 to 15 seconds after the loss of main feedwater event.

c) Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator.

   - The AFWS is initiated by a low steam generator level signal from the SFRCS. It is assumed that the time delay between receiving the initiate signal and full AFW flow to the steam generators is 40 seconds for a case without loss of offsite power. This is a total delay of approximately 55 seconds from the loss of main feedwater event.

d) Minimum steam generator water level when initiating event occurs.

- Steam generator inventory rather than water level is used as an input to this analysis.

e) Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

- The initial steam generator inventory is dependent on power level. For this case, a liquid inventory of 39,600 lbm per steam generator was used. The depletion rate of the inventory before initiation of AFW averages about 248 lbm/sec. Following initiation of AFW, there is not enough level to determine the depletion rate (since AFW is initiated on low steam generator level). Following complete depletion of the liquid inventory, the entire AFW flow is vaporized until decay heat plus RC pump heat drops below the capability of the AFW system. At that time, steam generator inventory begins increasing again. The decay heat used in this calculation was 1.2 times the ANS 5.1 decay heat.

f) Maximum pressure at which steam is released from the steam generator(s) and against which the AFW pump must develop sufficient head.

- The peak steam pressure occurs shortly after AFWS initiation and is about 1075 psig. Soon after this peak, however, the steam pressure is controlled by the first bank of steam safety valves to a pressure of 1050 psig.

g) Minimum number of steam generators that must receive AFW flow.

- This analysis was run assuming both steam generators were available, however, the heat load can be removed with one AFW pump and one OTSG. See FSAR Section 15.2.2 for details.

h) RC flow condition - continued use of RC pumps or natural circulation.

- Continued operation of RC pumps was used for this analysis.

i) Maximum AFW inlet temperature.

- An inlet temperature of 120°F was used.

j) Following a postulated steam or feedline break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also, identify credit for primary system heat removal due to blowdown.

- FSAR Sections 15.4.4 and 15.2.8 contain the details of the assumptions used in the main steam line and main feedwater line break analysis, respectively. The time delay assumed (to isolate the isolable steam or feedline break and direct AFW flow to intact steam generator at full flow) is 40 seconds from the AFW initiating signal assuming no loss of offsite power.

k) Volume and maximum temperature of water in main feedlines between steam generator(s) and AFWS connection to main feedline.

- There are no piping connections between the AFWS to the main feed line at Davis-Besse 1.

l) Operating condition of steam generator normal blowdown following initiating event.

- Davis-Besse steam generators do not have a blowdown system at the present time.

m) Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

- A heat capacity of 1.256×10^6 BTU/^{o}F is used for calculating the volume of feedwater required to cool the RCS to decay heat system parameters.

n) Time at hot standby and time to cool down the RCS to DHR System cut-in temperature to size AFW water source inventory.

- The condensate storage tank is sized to accommodate the plant at hot shutdown for thirteen hours followed by a six hour cooldown to 280^{o}F as reported in Section 9.2.7.2 of the FSAR.

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