November 18, 1987

Docket 50-346
Serial No. DB-87-014

Mr. Donald C. Shelton
Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Shelton:

SUBJECT: CHANGE TO TECHNICAL SPECIFICATION BASIS 3/4.7.1.2 - REQUIRED AUXILIARY FEEDWATER FLOW (TAC 65068)

The staff has completed its review of your request to change Technical Specification Basis Section 3/4.7.1.2, as requested in your letter dated March 23, 1987 (No. 1360). This change to the Basis supports a reduction of the required auxiliary feedwater flow to 600 gpm from 800 gpm. Based on the staff's review of the analyses presented in your request and in supplemental responses to staff questions dated October 2, 1987 (No. 1426), and November 6, 1987 (No. 1445), we find your proposed change acceptable. Accordingly, a revised page B 3/4 7-2 incorporating your proposed change is enclosed.

A copy of the Safety Evaluation supporting the staff's conclusion is enclosed.

Sincerely,

original signed by

Albert W. De Agazio, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. TS Basis page B 3/4 7-2
2. Safety Evaluation

cc w/enclosures:
See next page

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% its design pressure of 1050 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 14,175,000 lbs/hr which is 120 percent of the total secondary steam flow of 11,760,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the High Flux channels. The reactor trip setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{Z} \times W$$

where:

SP = reduced Trip Setpoint in percent of RATED THERMAL POWER (Not to Exceed W)

V = maximum number of inoperable safety valves per steam generator

W = High Flux Trip Setpoint for four pump operation as specified in Table 2.2-1

X = Total relieving capacity of all safety valves per steam generator in lbs/hour, 7,087,500 lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour, 845,759 lbs/hour

Z = Required relieving capacity per steam generator in lbs/hr, 6,585,600 lbs/hr.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEMS

The OPERABILITY of the Auxiliary Feedwater Systems ensures that the Reactor Coolant System can be cooled down to less than 280°F from normal operating conditions in the event of a total loss of offsite power.

Each steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 600 gpm at a pressure of 1050 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 280°F where the Decay Heat Removal System may be placed in operation.

Following any modifications or repairs to the Auxiliary Feedwater System piping from the Condensate Storage Tank through auxiliary feed pumps to the steam generators that could affect the system's capability to deliver water to the steam generators, following extended cold shutdown, a flow path verification test shall be performed. This test may be conducted in MODES 4, 5 or 6 using auxiliary steam to drive the auxiliary feed pumps turbine to demonstrate that the flow path exists from the Condensate Storage Tank to the steam generators via auxiliary feed pumps.

3/4.7.1.3 CONDENSATE STORAGE FACILITIES

The OPERABILITY of the Condensate Storage Tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere and to cooldown the Reactor Coolant System to less than 280°F in the event of a total loss of offsite power or of the main feedwater system. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO THE CHANGE OF TECHNICAL SPECIFICATION BASES SECTION 3/4.7.1.2
THE TOLEDO EDISON COMPANY
DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1
DOCKET NO. 50-346

1.0 INTRODUCTION

In a letter from Donald C. Shelton (Toledo Edison Company) to the U. S. Nuclear Regulatory Commission (NRC) dated March 23, 1987, a change in Technical Specification (TS) Bases for Davis-Besse Nuclear Power Station, Unit 1 was requested. The proposed modification involves a decrease from 800 gpm to 600 gpm for the Auxiliary Feedwater (AFW) flow requirement in the Davis-Besse TS Bases, Section 3/4.7.1.2. The reason for the proposed change is to assure equal flow requirements on all three AFW pumps now that a faulty variable flow valve has been removed from one of the three pump lines. According to information provided by the licensee, such a change would maintain reactor core parameters within safety limits and impose no additional risk to the public based on analyses of the applicable bounding case.

2.0 EVALUATION

The bounding transient for a decrease in AFW flow is the loss of feedwater transient. In Davis-Besse's Final Safety Analysis Report (FSAR) Accident Analysis, this transient may be initiated by any of three failures: the abnormal closure of a feedwater valve, pump failure, or a feedwater line break (FWLB). Babcock & Wilcox has determined the most bounding of these to be the abnormal feedwater valve closure. The reactor coolant system (RCS) pressure surge due to the FWLB, occurs before AFW initiation takes place. Therefore the reduction in AFW flow is of less concern than for the valve malfunction case, which relies upon AFW for the recovery of steam generator (SG) heat transfer

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capability. Also, the reactor trip for the FWLB occurs significantly sooner than for the valve malfunction, making the feedwater valve malfunction case more limiting.

Assumptions made in the reanalysis of the loss of feedwater transient include:

* Initial reactor power at 102%
* No credit for PORV, pressurizer sprays, or makeup flows,
* 1.2 times ANS 5.1 (1979) decay heat curve,
* Availability of offsite power, and
* AFW feed to only one SG, due to single failure considerations, at a flowrate of 600 gpm.

In the reanalysis, the loss of feedwater results in a rise in RCS temperature and pressure leading to a reactor trip at the 2400 psia high pressure trip setpoint. The primary pressure peaks at approximately 2605 psig, which exceeds the system design limit of 2500 psig, but remains within the safety limit of 2750 psig (110% of design pressure) and is therefore acceptable.

Meanwhile, on the secondary side, the SG level drops. Ten inches is the SG low level setpoint which actuates AFW with a 40 second delay for flow delivery. The analysis shows dryout of the SG at the same time as AFW flow initiation begins. However, the dryout condition exists for a relatively short duration (see Figure A). The dryout condition is tolerable due to the Babcock & Wilcox SG design which includes AFW injection directly on the SG tubes for immediate recovery of heat transfer capability. The RCS hot leg reacts with a maximum temperature of approximately 615° F which lies below the 635° F Reactor Core Safety Limit. The consequences of SG dryout in this Babcock & Wilcox designed plant is acceptable, mainly due to the once through SG design.

Pressurizer level goes offscale high but maintains a bubble approximately four vertical feet from the top of the pressurizer. The pressurizer thereby avoids filling completely and is able to control primary pressure conditions.

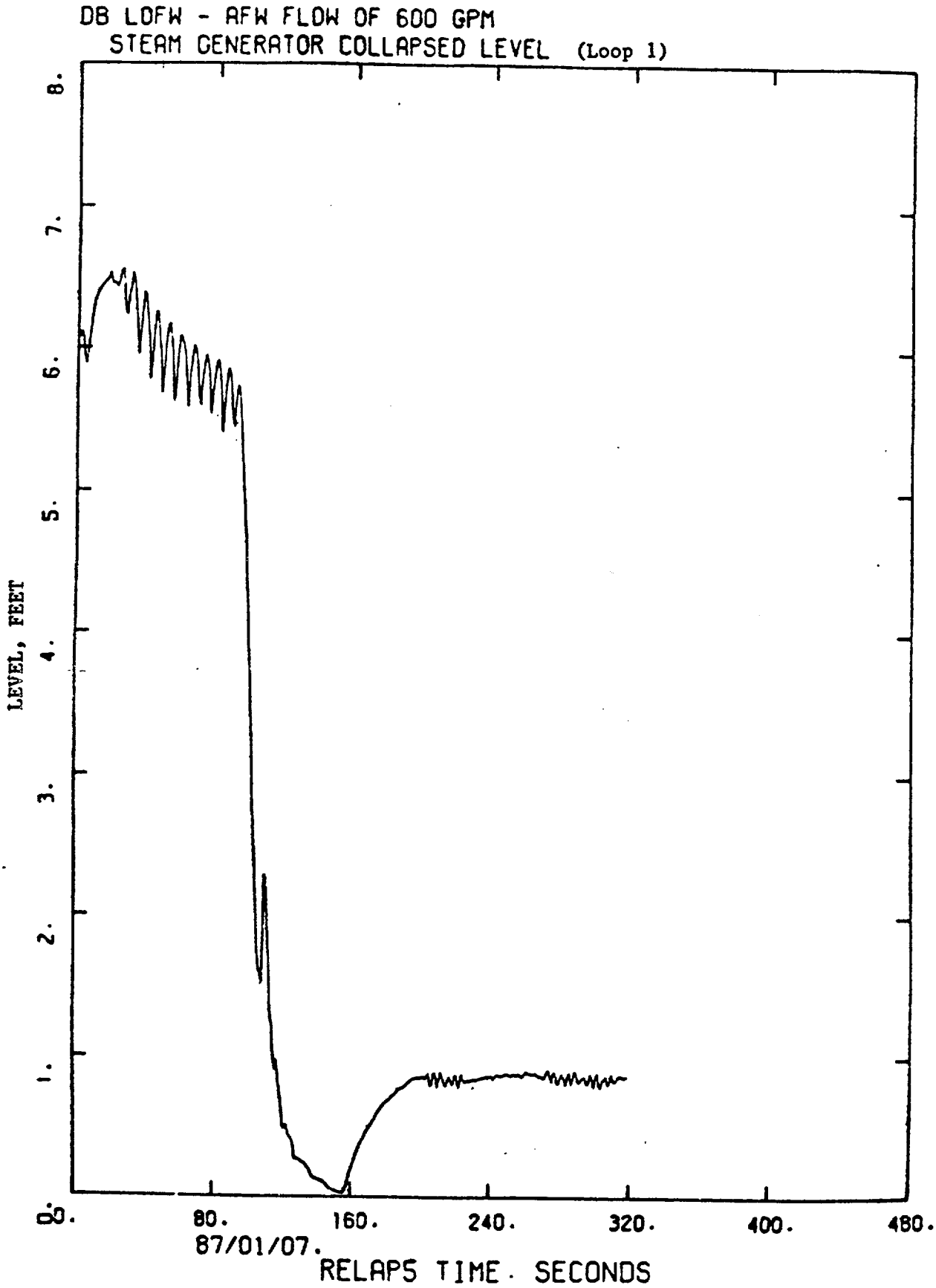
The RELAP5 computer code was used to model the bounding loss of feedwater transient. RELAP5 has not been approved by the NRC staff for safety analyses for the Davis-Besse plant. In this particular application however the code was found to give conservative results. The reactor trip time of 14 seconds calculated by the code is longer than the actual expected reactor trip time. With the conservative reactor trip time, SG inventory and heat transfer is underestimated resulting in a conservative RCS temperature rise and pressurizer insurge. Because the results of the use of RELAP5 are conservative in this application, we find it acceptable for use in this licensing action.

3.0 CONCLUSION

Because the licensee evaluated the bounding transient associated with the proposed AFW flow decrease using a conservative computer model, and because the results are within the safety limits of the TS, we find the proposed change acceptable.

Dated: November 18, 1987

FIGURE A


DB LOFW - AFW FLOW OF 600 GPM
STEAM GENERATOR COLLAPSED LEVEL (Loop 1)
LEVEL, FEET
0.
1.
2.
3.
4.
5.
6.
7.
8.
0.
80.
160.
240.
320.
400.
480.
87/01/07.
RELAP5 TIME SECONDS