TOLEDO EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346

License No. NPF-3

Serial No. 1-268

May 17, 1982

PRINCIPAL STAFF			
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Mr. James G. Keppler, Director
United States Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

Please find attached a copy of Facility Change Request (FCR) 82-006 Revision A and associated Safety Evaluation for Davis-Besse Nuclear Power Station Unit No. 1 (DB-1). This FCR is to document the justification for continued power operation of DB-1 at 90% and at rated thermal power in light of the increase in Borated Water Storage Tank water temperature from 90°F value (maximum assumed in FSAR) to a value of 120°F during the period of January 12 through 14, 1982.

Very truly yours,

R P Crouse

RPC:GAB:lab

attachment

cc: DB-1 NRC Resident Inspector

MAY 20 1982

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DAVIS-BESSE NUCLEAR POWER STATION UNIT 1
FACILITY CHANGE REQUEST INITIATION
ED-6806

1. FCR NO. 82-006
2. ☐ ORIGINAL ☒ REV. A

3. SYSTEM	4. SUS NO.	5. COMPONENT/EQUIPMENT NO.
BWST, EMER. CORE COOLING SYSTEM	049A	BWST

6. PROPOSED CHANGE, TEST, EXPERIMENT OR LICENSE AMENDMENT

NOTE: THIS FCR REVISION SUPERSEDES THE ORIGINAL FCR.
NOTE: THE SAFETY EVALUATION ATTACHED TO THE ORIG. FCR IS ALSO SUPERSEDED BY THIS FCR.

1. THIS FCR IS TO DOCUMENT THE SAFETY EVALUATION FOR OPERATION OF DAVIS BESSE IN LIGHT OF A BWST WATER TEMP. INCREASE TO 120°F (BEYOND THE 90°F VALUE ASSUMED IN THE FSAR) IN CONJUNCTION WITH A POWER REDUCTION TO 90%. DURING THE PERIOD OF JANUARY 13-14 1982

6a. ☒ CHECK BOX IF ADDITIONAL SHEETS ARE ATTACHED

7. REASON FOR REQUEST

TO PROVIDE DOCUMENTATION FOR JUSTIFICATION OF POWER OPERATION AT 90% AND AT RATED THERMAL POWER WITH A BWST WATER TEMPERATURE UP TO APPROXIMATELY 120° F.

7a. ☐ CHECK BOX IF ADDITIONAL SHEETS ARE ATTACHED

8. LIST COMMITMENTS (NRC, LER, DVR, OTHER)	9. COMPLETION DUE DATE	10.
NONE	IMMEDIATE	☐ OUTAGE REQUIRED ☒ NON OUTAG ☐ REDUCED LOAD ☐ UNKNOWN

11. REQUESTED IMPLEMENTATION DATE	12. REASON FOR DATE
ASAP	REVISE LER / SUBMITTAL TO NRC

13. GROUP CODE	14. PRIORITY	15. REQUESTOR	SECTION	DATE
5		SUSHIL JAIN	PLANT NUC. SYSTEMS	2/12/82

16. GROUP CODE

	DESCRIPTION
1	Prevents or Limits Plant Power Level
2	NRC Commitment
3	ALARA, Potential Radiation Release/Concern
4	Industrial Security
5	LER/DVR Commitment
6	Improves Plant Availability
7	Safety Concern For Equipment/Personnel
8	Convenience
9	Other
0	Paperwork Change Only - No Physical Fieldwork Required

B) PRIORITY
"1" Special High Priority (Assigne only by Division Director o higher authority)
"3" Required High Priority
"5" Required but not immediately
"7" Highly desirable
"9" Desirable but can be done late

17. APPROVED FOR FURTHER ACTION

17a. SUPERVISOR	DATE	17b. STATION SUPT.	DATE	17c. NUCLEAR ENG. MGR.	DATE
F R Miller	3/5/82	T D Murray	3/8/82	C R Domeck / FHM	3/5/82

18. ENGINEERING ACTION

18a. LICENSE AMENDMENT REQUIRED
☐ Yes ☒ No

18b. ACCOUNTING CLASSIFICATION
☐ Construction ☐ Maintenance ☒ Operation

18c. AREA FUNCTION/JOB ORDER

18d. ENGINEERING ACTION COMPLETED BY SUSHIL JAIN DATE 3/4/82

PROPOSED CHANGE CONTINUED

2. THIS FCR ALSO DOCUMENTS THE SAFETY EVALUATION FOR OPERATION OF DAVIS BESSE AT RATED THERMAL POWER FOR A PERIOD FROM JANUARY 12, 1982 TO JANUARY 14, 1982 WITH A BWST WATER TEMPERATURE OF 120°F.

THE ATTACHED SAFETY EVALUATION IS NOT INTENDED TO BE USED FOR BWST WATER TEMPERATURE EXCURSIONS FOR DATES OTHER THAN MENTIONED ABOVE.

.S-BESSE NUCLEAR POWER STATION UNIT 1
FACILITY CHANGE REQUEST IMPLEMENTATION SUPPLEMENT
FOR ENGINEERING ACTION AND APPROVAL

19.FCR NO. 82-006	REV. A	SUPP. 1

20. ADDITION/CHANGE TO WORK PACKAGE AND REASON (Attach Additional Sheets If Necessary)

SEE LETTER NO. BT- N/A FOR WORK PACKAGE

21. ALARA REVIEW REQUIRED
☐ Yes ☒ No

22. CHANGES IN FACILITY AS DESCRIBED IN FSAR
☐ No ☒ Yes (If Yes, Safety Evaluation Required)

23. QA REQUIRED
☐ No ☒ Yes (If Yes, Reason) ☒ NSR ☐ FP ☐ PICA ☐ Q CORE DRILL ☐ ASME ☐ ISI ☐ SEISMIC ☐ OTHER ____

24. SAFETY REVIEW

a. Yes ☒ No ☐ NSR change in the facility.

b. Yes ☐ No ☒ NSR changes in the procedures as described in the FSAR.

c. Yes ☐ No ☒ NSR test or experiment not described in the FSAR.

d. Yes ☐ No ☒ A change in the Technical Specifications or Operating License.

e. NOTE: If any of the above sections are marked YES, check one box below and go to Block 25.

☐ Previous written Safety Evaluation in Supplement No. ____ applies.

☒ Written Safety Evaluation attached ~~or~~ ~~referenced~~.

☐ PICA and/or Q Core Drill ONLY.

25. SAFETY EVALUATION

a. Yes ☐ No ☒ The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased.

b. Yes ☐ No ☒ A possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created.

c. Yes ☐ No ☒ The margin of safety as defined in the bases for any Technical Specification is reduced.

d. Yes ☐ No ☒ An analysis is attached if any of the above is marked YES.

e. NOTE: If any of the above sections are YES, an 'UNREVIEWED SAFETY QUESTION' is involved, and the change may not be made without NRC concurrence.

26. SECURITY REVIEW

☐ Yes ☒ No Decrease in the effectiveness of the Industrial Secuirty Plan.

NOTE: If the Security Review indicates YES, then the change may not be made without NRC concurrence.

27. ENGINEERING SUMMARY OF PLANS AND PRECAUTIONS

- ACTIONS UNDER ORIGINAL FCR HAVE BEEN COMPLETED.
- FCR CAN BE CLOSED OUT FOLLOWING REQUIRED REVIEW AND APPROVALS

28. PREPARED BY	DATE	29. RESPONSIBLE AREA/ORGANIZATION OR INDIVIDUAL FOR IMPLEMENTATION
SUSHIL JAIN	2/12/82	Station Superintendent ~~General~~

30. REVIEW AND APPROVAL FOR IMPLEMENTATION

30a. ENGINEERING	DATE	30b. SRB CHAIRMAN	DATE	30c. STATION SUPT.	DATE
F R Miller	3/5/82	[signature]	3/9/82	[signature]	3/9/82

31. IF LICENSE AMENDMENT, APPROVED BY CNRB

MTG ____ DATE ____

Safety Evaluation for FCR 82-006 Revision A

This safety evaluation is performed to document the justification for continued power operation in light of the increase in Borated Water Storage Tank (BWST) water temperature beyond the value assumed in the accident analyses of Davis-Besse Nuclear Power Station Unit 1 (DB-1) Final Safety Analysis Report (FSAR). The safety function of the BWST is to provide a supply of borated water for injection by the ECCS in the event of a LOCA. Specifically, this safety evaluation addresses the following two modes of operation at Davis-Besse Nuclear Power Station:

1. Operation at 90% rated thermal power with a BWST water temperature of 120°F for the period of January 13 through January 14, 1982. Section I below and attachment 1 hereto constitute the safety evaluation for this mode of operation.

2. Operation at rated thermal power with a BWST water temperature of 120°F for the period of January 12 through January 14, 1982. Section II below and attachment 2 hereto provide the safety evaluation for this mode of Operation.

Background

On Wednesday, January 13, 1982, Nuclear Engineering was advised by the Station personnel that BWST water temperatures as high as 120°F had been experienced on January 12 and 13, 1982. The high temperature was caused by an attempt to thaw out the frozen BWST level transmitter sensing lines. Following an investigation into the FSAR accident analyses, it was determined that the Emergency Core Cooling System (ECCS) analysis assumes a BWST water temperature of 90°F.

I. Operation at 90% Power with a BWST Water Temperature of 120°F

Subsequent to the preceding identification, Nuclear Engineering contacted the B&W ECCS analysis personnel and discussed possible resolution of the elevated BWST water temperatures. B&W indicated that the most significant contribution to core heat removal will be through the high pressure injection (HPI) pumps. For the elevated BWST water temperature condition, the low pressure injection (LPI) system is of lesser significance since it is primarily used to maintain water level above the core and the excess water inventory is discharged through the break to the containment emergency sump. As a result, effects of high BWST water temperature were considered only for the HPI pumps at first. In accordance with B&W discussions, Nuclear Engineering performed a brief calculation to determine the maximum safe level of power opreation at which a BWST water temperature of 120°F will provide adequate emergency core cooling capability. This is described in the following paragraphs.

A. Reactor Coolant System Heat Removal

If the BWST water temperature is raised beyond 90°F, the total heat removal capacity of the ECCS is correspondingly reduced. To quantify this effect, enthalpy of water at 90° (h_{90}, 58.06 Btu/lbm) and 120°F (h_{120}, 88.00 Btu/lbm) was determined. It was then assumed that

following a LOCA, the Reactor Coolant System (RCS) reaches saturation conditions at 1065 psia (552°F). This corresponds to the steam generator secondary side saturation conditions and assumes that the RCS cold leg temperature follows the steam generator saturation temperature. The enthalpy of water (reactor coolant) at this temperature (h_{sat}) was determined to be 552.4 Btu/lbm. To determine the relative change in the heat removal capability a ratio R was calculated as follows:

$$R = \frac{h_{sat} - h_{120}}{h_{sat} \quad h_{90}} = \frac{552.4 - 88}{552.4 - 58} = 0.939$$

This implies that the 120°F BWST water is capable of removing approximately 94% of that decay heat which is produced by operating at 100% rated thermal power. It is noted that the decay heat produced in the core is proportional to the steady state level of power operation. Based on prudent engineering judgment, Nuclear Engineering immediately instructed Station management to conservatively limit power level to 90% of rated thermal power until such time that BWST water temperature is restored to within 90°F. The above calculation possesses additional conservatisms as indicated below:

1. Latent heat of evaporation leading to formation of steam in the RCS has been neglected.

2. It is assumed that all the heat transfer is through the BWST water. No credit has been taken for the cooling via steam discharge through the postulated break.

3. No credit has been taken for heat removal through the steam generator.

Subsequent to the above limitation on power level, Nuclear Engineering further investigated the ECCS pump net positive suction head requirements and the containment energy removal effects. This is summarized in the following paragraphs. Nuclear Engineering also requested B&W to further investigate this matter in light of DB-1 ECCS analysis. The results of this investigation are summarized in Attachment 1.

B. Net Positive Suction Head Considerations

Another effect of the increase in BWST water temperature is to degrade the available net positive suction head (NPSH) for the ECCS pumps. For the low pressure injection and containment spray pumps, operation without cavitation at higher temperature has been demonstrated acceptable in FSAR (Section 6.3 2.14) when taking suction from the containment emergency sump. This bounds the 120°F suction water temperature from the BWST. In addition, calculations performed for a 30° increase in water temperature (90° to 120°F) indicate that this increase results in a loss of approximately 2.5' of available NPSH. This is acceptable in light of the ample margin existing between the required and available NPSH (see FSAR Section 6.3.2.14).

C. Effects on Containment Vessel Integrity

With an increase in BWST water temperature, the adverse impact on the containment vessel integrity is caused by two factors:

1. Containment energy removal effects owing to the higher temperature containment spray water.

2. Increase in blow down energy released to the containment through the break. This is based on the fact that the higher temperature ECCS water will eventually be released to the containment resulting in higher energy releases than those assumed for 90°F BWST water.

There is no effect of the higher temperature containment spray water on the containment temperature and pressure peak since the peak is reached before the containment sprays have an effect.

The containment energy release from the reactor coolant system, however, affects the containment pressure peak. Per the attached B&W analysis (see Section IV of B&W analysis, attachment 1) with operation at 90% power, the net effect of added energy to containment spray and reduced energy release from the break (due to reduction in operating power) is a decrease in containment pressure and temperature response. Also, the peak containment temperature and pressure are lower than those provided in the FSAR. Based on the above, it is concluded that containment vessel integrity will not be adversely affected during a LOCA by operation at 90% power with a BWST water temperature of 120°F.

II. Operation at 100% Power with BWST Water Temperature of 120°F

Per the request of Toledo Edison, B&W performed an additional analysis to evaluate 100% full power operation in light of a BWST water temperature of 120°F. This analysis is included in attachment 2 hereto, and is based on the worst case kw/ft actually observed during the operation between 1/12/82 to 1/19/82.

A. Net Positive Suction Head Considerations

The results of Section I.B. remain valid for this case.

B. Mass and Energy Release to Containment

As described in Section I.C. above, the higher temperature containment spray water does not have an adverse effect on the post-accident containment temperature and pressure peaks. However, analysis performed by B&W (see Section IV attachment 2) indicate an increase in the containment energy content by 1.7%. Most conservatively, this will raise the peak containment pressure from 36.95 psig to a value less than 37.55 psig (increase of less than 0.6 psi) which provides adequate margin from the maximum design pressure of 40 psig. Based on the above, it is our judgment that the containment vessel integrity

would not have been adversely affected by operating at 100% power with a BWST water temperature of 120°F.

It is therefore concluded that with 120°F BWST water, 100% power and actual observed lower peaking conditions, the existing large break LOCA analyses bound the peak clad temperature. For small breaks, a decreased reactor vessel inventory will result. However, substantial margin to 10 CFR 50.46 peak clad temperature would have been retained.

It is emphasized that the above determinations possess additional conservatism in that:

1. ECCS analysis requires only one train of HPI, LPI and containment spray pump to be operable. During the time period of interest above, both trains of ECCS were operable. This substantially augments the containment and core energy removal capability available during this time.

2. The ECCS analysis assumes that the core produces 1.2 times the ANS decay heat, whereas in reality the twenty percent excess decay heat is not produced.

Pursuant to the above it is concluded that:

1. The increase in BWST water temperature to 120°F with a corresponding decrease in rated thermal power to 90% (for the period of January 13-14, 1982) did not degrade the safety function of the BWST and the emergency core cooling capability of the ECCS is not compromised.

2. The consequences of operating at 100% rated thermal power with a BWST water temperature of 120°F (for the period of January 12-14, 1982) were bounded by the existing LOCA analyses in light of the low local operating peaks that were observed when the plant was in this configuration. Also, the containment integrity considerations are not significantly altered.

Subsequently, the two modes of operation described above do not involve an unreviewed safety question.

pk b/4

Sushil Jain
3/4/82

Frederick R Miller
3/5/82