TOLEDO
EDISON

RICHARD P. CROUSE
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Docket No. 50-346
License No. NPF-3
Serial No. 862
October 14, 1982

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:

Under separate cover, we are transmitting three (3) original and forty (40) conformed copies of an application for Amendment to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station Unit No. 1.

This application requests that the Davis-Besse Nuclear Power Station Unit 1 Technical Specifications, Appendix A, be revised to reflect the changes attached. The proposed changes include:

1. Changes in Section 3.4.3 and Bases
2. Changes in Bases B3/4.7.1.2
3. Changes in Table 3.3-4

The three attachments identify the proposed changes and their safety evaluations. Item 1 concerns the setpoints for pressurizer electromatic relief and code safety valves. Item 2 concerns a revision to the bases on Auxiliary Feedwater System. Item 3 concerns a change in trip setpoints for BWST level and Essential Bus Feeder Breaker Trip to account for instrument error.

This amendment requests two changes of Class III type. Item 2 changes in the bases are for update proposes only since the bases are not part of the technical specifications pursuant to 10CFR50.36(a). Therefore, it is determined to be a Class III amendment request and enclosed is a check for 8,000 as requested by 10CFR170.22.

Very truly yours,

R P Crouse

Attachment

Aool
Rec'd w/out
Check

cc: DB-1 NRC Resident Inspector

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NO. NPF-3

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

Enclosed are forty-three (43) copies of the requested changes to the Davis-Besse Nuclear Power Station Unit No. 1 Facility Operating License No. NPF-3, together with the Safety Evalution for the requested change.

The proposed changes include:

1. Changes in Section 3.4.3 and Bases
2. Changes in Bases B3/4.7.1.2
3. Changes in Table 3.3-4

By Terry D Murray
Station Superintendent

For R. P. Crouse
Vice President, Nuclear

Sworn and subscribed before me this 14th day of October, 1982.

Nora Lynn Flood
Notary Public

NORA LYNN FLOOD
Notary Public, State of Ohio
My Commission Expires Sept. 1, 1987

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Attachment 1

I. Changes to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications Section 3.4.3 and Bases.

A. Time required to Implement

This change is to be effective upon NRC approval and implementation by Toledo Edison.

B. Reason for Change (Facility Change Request 79-348 Rev. A)

To provide increased margin between the PORV and Code Safety valve setpoints. This increased margin is facilitated by the relocation of pressurizer code safety valves.

C. Safety Evaluation

See Attached

SAFETY EVALUATION

This amendment request proposes changes to the PORV and the Pressurizer Code Safety Valve setpoints in the Technical Specifications in light of the relocation of code safety valves to the top of the pressurizer.

The safety function of the pressurizer code safety valves is to provide RCS over pressure protection in accordance with the ASME code. PORV is only used to prevent the opening of the code safety valves.

The accident analysis assumes that the Pressurizer code safety valves open at 2500 psig + 3%. In the original configuration there was 35 feet of piping between the pressurizer and the safety valve. The full flow pressure drop was 65 psig in this piping. This piping and the pressure drop are being eliminated by the relocation of code safety valves, therefore, the Technical Specification code safety valve setpoint can be raised by 65 psi (from 2435 to 2500 psig). With a 1% tolerance on the setpoint, the Technical Specification requirement should be $\leq$2525 psig. This setpoint can accomodate more margins to avoid PORV from challenging the safety valves as shown in the following table.

	Prior to Moving Pressurizer Code Safety Valves	After Moving Pressurizer Code Safety Valves	Proposed New Technical Specification Trip Setpoints
Code Safety Valve Setpoint (psig)	2435 + 1%	2435 + 1%	2500 + 1%
Reactor Coolant System pressure with full flow through the code safety valves (psig)	2500 + 1%	2435 + 1%	2500 + 1%
PORV open setpoint (psig)	2390	2390	2390
Margin between code safety valve and PORV open setpoints (psig)	45	45	110

Based on the above, it is concluded that the above change provides increased margin between the PORV and Code Safety Valve Setpoints. At the same time, the safety function of the code safety valves is not degraded. Pursuant to the above, this is not an unreviewed safety question.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM

SAFETY VALVES AND ELECTROMATIC RELIEF VALVE - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of ≤ 2525 psig* ~~2435 PSIG ± 1%.*~~ When not isolated, the pressurizer electromatic relief valve shall have a trip setpoint of ≥ 2390 PSIG and an allowable value of ≥ ~~2385.5~~ 2390 PSIG.**

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 For the pressurizer code safety valves, there are no additional Surveillance Requirements other than those required by Specification 4.0.5. For the pressurizer electromatic relief valve a channel calibration check shall be performed every 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** Allowable value for channel calibration check.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced.

RC Pressure - Low, High and Pressure Temperature

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC High Pressure setpoint is reached before the High Flux Trip Setpoint. The trip setpoint for RC High Pressure, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC High Pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, ~~2435~~ psig. The RC High Pressure trip also backs up the High Flux trip. ≤2525

The RC Low Pressure, 1985 psig, and RC Pressure-Temperature ($12.60\ T_{out}\ °F-5660$) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the Flux - Δ Flux-Flow trip the High Flux/Number of Reactor Coolant Pumps On trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

DELETION OF THIS PAGE PREVIOUSLY
PROPOSED BY LETTER
Serial No. 669 Date 12/26/80

SEE REVISED SECTION
3.4.2 ATTACHED HERETO

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2435 PSIG ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE DHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SAFETY VALVES - SHUTDOWN

CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 669 Date 12/26/80

LIMITING CONDITION FOR OPERATION

3.4.2 Decay Heat Removal System relief valve DH-4849 shall be OPERABLE with a lift setting of $\leq$ 330 PSIG* and isolation valves DH-11 and DH-12 open and control power to their valve operators removed.

APPLICABILITY: MODES 4 and 5.

ACTION:

A. With DH-4849 not OPERABLE:

1. Make the valve OPERABLE within eight hours; or

2. a. Within next one hour, disable the capability of both high pressure injection (HPI) pumps to inject water into the reactor coolant system; and

   b. Within next eight hours:

      1. Disable the automatic transfer of makeup pump suction to the borated water storage tank on low makeup tank level; and

      2. Reduce makeup tank level to $\leq$ 73 inches and reduce reactor coolant system pressure and pressurizer level within the acceptable region on Figure 3.4.2-a (in MODE 4) and 3.4.2-b (in MODE 5).

B. With DH-11 or DH-12 closed, open DH-21 and DH-23 within one hour.

C. With the control power not removed from DH-11 and DH-12, remove the power to the valve operators at the Motor Control Centers within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2. Decay Heat Removal System relief valve DH-4849 shall be determined OPERABLE:

a. per the surveillance requirements of Specification 4.0.5.

b. at least once per 24 hours by verifying either:

   1. isolation valves DH-11 and DH-12 open with control power removed from their valve operators; or

   2. valves DH-21 and DH-23 open.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

• This sheet was submitted in previous request Serial # 669 dated 12/26/80

• No change on this sheet is requested in this Submittal

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one loop, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE, ensuring that the DNBR will be maintained above 1.30 at the maximum possible THERMAL POWER for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR equal to 22%, whichever is more restrictive.

In MODES 3, 4 and 5, a single reactor coolant loop or DHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two DHR loops to be OPERABLE.

Natural circulation flow or the operation of one DHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capacity of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 336,000 lbs. per hour of saturated steam at the valve's setpoint.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

The relief capacity of the decay heat removal system relief valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that this relief valve is not OPERABLE, reactor coolant system pressure, pressurizer level and make up water inventory is limited and the capability of the high pressure injection system to inject water into the reactor coolant system is disabled to ensure operation within reactor coolant system pressure - temperature limits.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

CHANGES PREVIOUSLY PROPOSED BY LETTER Serial No 669 Date 12/26/80

For a RPS high pressure trip setpoint of 2300 psig, the maximum overshoot of the Reactor Coolant System pressure for a loss of feedwater (LOFW) event would be 2350 psig. Also, the LOFW is the maximum over-pressure anticipated transient. The string inaccuracies and drift for the RPS high pressure trip are 15.29 psi, or 16 psi conservatively. The maximum pressure peak for an anticipated transient is then 2366 psig.

The inaccuracies and drift for the string that controls the electromatic relief valve for the pressurizer are 16.75 psi, or 17 psi conservatively. Included in this value is an inaccuracy of 4 psi and a drift of 7.5 psi for the transmitter. The 4 psi and 7.5 psi were combined by taking the square root of the sum of the squares, giving 8.5 psi. Subtracting 4 psi from 8.5 psi gives a value of 4.5 psi that is attributable to only the drift. The 8.5 psi was then added to inaccuracy and drift values for other components in the string to obtain a total of 16.75 psi.

The allowable value of $\geq$2385.5 psig is obtained by subtracting 4.5 psi due to the drift from the trip setpoint of $\geq$2390 psig. The minimum lift pressure for the pressurizer electromatic relief valve is then (2400 - 10 - 17) psig = 2373 psig. Consequently, the resultant margin between the maximum pressure peak of 2366 psig and minimum lift pressure of 2373 psig for the pressurizer electromatic relief valve following an anticipated transient is 7 psi.

Thus, a 2300 psig RPS high pressure trip setpoint and the above values for the pressurizer electromatic relief valve will avoid actuation of the pressurizer electromatic relief valve during anticipated transients.

THIS PAGE TO BE
REPLACED BY
ATTACHED BASES.

The pressurizer code safety valves must be set such that the peak Reactor Coolant System pressure does not exceed 110% of design system pressure (2500 psig) or, 2750 psig. The control rod group withdrawal accident will result in the most limiting high pressure in the RCS. The analysis assumes RPS high pressure trip at 2300 psig and the code safety valves open at 2500 psig. The tolerance on the RPS instrument accuracy is 30 psi and, it is +3% for the code safety valve settings. The pressurizer electromatic relief valve was assumed not to open for this transient. The resulting system peak pressure was calculated to be 2716 psig. Therefore, the code safety valve setpoint is conservatively set at $\leq$ 2525 psig which is the maximum pressure of 2500 psig + 1% for tolerance.

The pressurizer electromatic relief valve should be set such that it will open before the code safety valves are opened. However, it should not open on any anticipated transients. Loss of Feedwater (LOFW) was identified as the limiting anticipated transient for RCS pressure. The analysis assumes RPS high pressure trip at 2300 psig; with 30 psi for instrument errors, the resulting peak RCS pressure is calculated to be 2380 psig. This includes a 50 psig pressure overshoot on a LOFW transient.

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Attachment 2

I. Changes to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications, Bases B3/4.7.1.2.

A. Time required to Implement

This change is to be effective upon NRC approval.

B. Reason for Change (Facility Change Request 82-138)

To correct the discrepancy between the FSAR and the Technical Specification.

C. Safety Evaluation

See Attached

SAFETY EVALUATION

This revision to the Technical Specification Bases Section 3/4.7.1.2, page B3/4.7-2 relates to the pumping capacity of Auxiliary Feedwater Pumps. The following safety evaluation constitutes the review of the change pursuant to the 10 CFR 50.59.

The safety function of the Auxiliary Feedwater System is to provide feedwater to the steam generators for the removal of reactor decay heat in the absence of main feedwater and to promote natural circulation of the Reactor Coolant System in the event of a loss of all four reactor coolant pumps.

Per the existing Technical Specification Bases Section B3/4.7.1.2, the required capability of the auxiliary feedwater pumps to deliver feedwater flow to remove decay heat and reduce the reactor coolant system temperature to less than 280°F is given to be 850 gpm at 1035 psig. As per FSAR Vol. 9, Section 15, Page 15.1.7-1 and per B&W analysis submitted to the NRC in Toledo Edison Serial No. 717 dated May 22, 1981, a flow rate of 800 gpm is determined to be acceptable to meet the required safety function of the Auxiliary Feedwater System. Therefore, the numbers in the bases need to be revised accordingly. Since a capacity of 800 gpm at 1050 psig has been demonstrated to be acceptable in the FSAR and Toledo Edison submittal of May 22, 1981 this change does not degrade the safety function of the Auxiliary Feedwater System. Therefore this is not an unreviewed safety question.

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 ~~850~~ 800 gpm at a pressure of ~~1035~~ 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 into operation.

3/4.7.1.3 CONDENSATE STORAGE FACILITIES

The OPERABILITY of the condensate storage tank with the and to 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

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Attachment 3

I. Changes to Davis-Besse Nuclear Power Station Unit 1, Appendix A Technical Specifications Table 3.3-4.

   A. Time required to Implement

      This change is to be effective upon NRC approval.

   B. Reason for Change (Facility Change Request 82-116)

      To account for instrument error in the trip setpoint for the BWST level and Essential Bus Feeder Breaker Trip.

   C. Safety Evaluation

      See Attached

SAFETY EVALUATION

This amendment request proposes that the lower end of the Technical Specification Borated Water Storage Tank (BWST) low level interlock trip setpoint be changed from 91.5 inches to 89.5 inches with a corresponding change in the Technical Specification allowable value. The request also proposes a change to the Essential Bus Feeder Breaker C1 (D1) trip setpoint from 7±1.5 seconds to $\leq$7.8 seconds in the Technical Specifications.

The safety function of the BWST low level interlock is to provide a permissive output to enable the manual transfer of decay heat (DH) and containment spray (CS) pump suction from the BWST to the containment emergency sump on low level in the BWST. This trip is also used to prevent a premature manual transfer and to protect these pumps from cavitation for lack of proper net positive suction head. The safety function of the C1 (D1) essential bus feeder breaker trip setpoint is to provide a time delay setpoint of the undervoltage relays (set at 90%) on the 4.16 KV safety related buses which actuate the diesel generators in the event of loss of offsite power. It also ensures acceptable voltages on the safety related 480V buses for continuous and emergency operations.

Change In BWST Low Level Setpoint

At present, the suctions to the DH and CS pumps are manually transferred from the BWST to the Containment Emergency Sump on a Safety Features Actuation System (SFAS) trip of incident level 5 (BWST low level). The lower end of the Technical Specifications for the BWST low level interlock trip setpoint is 91.5 inches. However, the existing setpoint range for the BWST low level interlock allowed by the Technical Specifications does not permit adequate margin for errors inherent in the instrumentation. The following table shows the adequacy of the BWST levels required for this suction transfer to be successfully performed as a manual transfer in light of the proposed setpoint change.

Table 1

Description	BWST Level (Inches)	BWST Volume (Gallons)
Develop Minimum Level to Transfer Suction to Containment Emergency Sump		
1. Accident Analysis minimum level to start the transfer per the original analysis	36	
2. Instrument string inaccuracy and drift	13.5	
3. Lowest safe indicated reading to start control room operator action to transfer	49.5	
Develop Minimum Contained Volume		
1. Highest (lowest) indicated reading to Interlock trip can occur (Technical Specification trip limits)	100.5 (89.5)	
2. Instrument string inaccuracy and drift	± 13.5	
3. Highest (lowest) actual level that Interlock trip can occur (this volume may not be available for the decay heat or containment spray pump)	114 (76)	
4. Instrument String drift	± 1.2	
5. Highest (lowest) allowable interlock trip	101.7 (88.3)	
6. 360,000 gallons required to be added for Emergency Core Cooling System (ECCS) analysis	334.3	360,000
7. Lowest safe indicated level for ECCS Analysis in Modes 1, 2, 3, & 4	448.3	482,778 (minimum volume presently required by Technical Specification)

As shown in the above table, a control room operator will manually perform this transfer when the safety grade level indicators in the control room indicate between 49.5 and 100(89.5) inches of BWST level. This will give a control room operator about 4 minutes to make the transfer safely. At maximum flow from BWST this will occur at approximately 23 minutes. During this time, the control room operator is procedurally instructed to monitor the BWST level indicators to initiate timely manual transfer. As observed from the above calculation table, the minimum level requirements of BWST will be met if the transfer is initiated within about 4 minutes after the indicated BWST level drops to the interlock setpoint. This assumes the worst case condition of a BWST level draw-down of 11.3 inches per minute.

A control room operator will manually perform this transfer about 23 minutes after the initial SFAS trip that started all high pressure injection, low pressure injection and containment spray pumps assuming their maximum flow. The accident analysis requires 360,000 gallons to be added for ECCS analysis when in Modes 1, 2, 3, & 4.

Change In Undervoltage Relay Time Delay Setpoint

Each safety related 4.16 KV bus C1 (D1) contains four relays which sense bus voltage and initiate tripping of the bus (C1 or D1) feeder breakers on undervoltage. The 90% setting is the minimum permissible operating voltage of the safety related 4.16 KV buses that ensures acceptable voltages on the safety related 480V buses for continuous and emergency operation. The maximum allowable operating delay for these "90% voltage" relays is 9 seconds as assumed in the accident analysis. In determining the Technical Specification trip setpoint for the time delay, the maximum error and drift inherent in the relay must be subtracted from the 9 seconds to ensure that the accident analysis value is not exceeded. The tolerance in the time delay is ±10% (of setpoint) with an additional ±5% (of setpoint) for drift. Therefore, the maximum error including drift would be ±15% of setpoint. Therefore, the setpoint should be $\leq$7.8 seconds in the Technical Specification as shown in Table 2 below. By making these changes the accident analysis value of 9 seconds will not be exceeded and the safety function of the protective relay will be enhanced.

Table 2

Description	Trip Setpoint
1. Accident Analysis Value	9 sec.
2. Technical Specification Value	$\leq$7.8 sec.
3. Maximum Error including drift (15% of setting)	0.15x7.8 = 1.17

These changes in the Technical Specifications will provide the same safety function as is provided by the present setpoint as discussed above. No adverse environment will be created by the change and the safety functions noted earlier will not be affected. Pursuant to the above it is concluded that no unreviewed safety question is involved.

TABLE 3.3-4

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
INSTRUMENT STRINGS		
a. Containment Radiation	< 2 x Background at RATED THERMAL POWER	< 2 x Background at, RATED THERMAL POWER'
b. Containment Pressure - High	≤ 18.4 psia	≤ 18.52 psia'
c. Containment Pressure - High-High	≤ 38.4 psia	≤ 38.52 psia'
d. RCS Pressure - Low	≥ 1620.75 psig	≥ 1615.75 psig'
e. RCS Pressure - Low-Low	≥ 420.75 psig	≥ 415.75 psig'
f. BWST Level	≥ ~~91.5~~ 89.5 and ≤ 100.5 in. H_2O	≥ ~~90.3~~ 88.3 and ≤ 101.7 in. H_2O #
SEQUENCE LOGIC CHANNELS		
a. Essential Bus Feeder Breaker Trip (90%)	≥ 3744 volts for ~~7 ± 1.5 sec~~ ≤ 7.8 sec	≥ 3558 volts for ~~7 ± 1.5 sec~~' ≤ 7.8 sec
b. Diesel Generator Start, Load Shed on Essential Bus (59%)	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec	≥ 2071 and ≤ 2450 volts for 0.5 ± 0.1 sec'
INTERLOCK CHANNELS		
a. Decay Heat Isolation Valve and Pressurizer Heater	< 438 psig	< 443 psig'*

' Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

* Referenced to the centerline of DH11 and DH12