Docket No. 50-346

License No. NPF-3

Serial No. 527

July 13, 1979

TOLEDO
EDISON

LOWELL E. ROE
Vice President
Facilities Development
(419) 259-5242

Director of Nuclear Reactor Regulation
Attention: Mr. Robert N. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Reid:

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 Specification, Appendix A,be revised to reflect the changes attached. The proposed changes include 1) addition of requirements on the interim anticipatory reactor trip system in sections 3.3.2.3, 4.3.2.3 and Table 3.3-15 and Table 4.3-15; 2) addition of requirements for auxiliary feedwater flow indications in section 4.7.1.2; 3) change to reactor coolant high pressure reactor trip setpoint in Table 2.2-1; 4)addition of a reactor coolant pressure temperature curve, Figure 3.4-5; 5) addition of requirements in sections 3.4.3 and 4.4.3 for the electromatic relief valve; 6) change to technical specification basis 2.2.1; and 7) change to SFAS actuation mode and actions of Tables 3.3-3 and 4.3-2.

This amendment request involves several changes of Class III type. It is therefore determined to be a Class IV amendment. Enclosed is $12,300.00 as required by 10CFR170.

The seven attachments identify each proposed change, its safety evaluation and schedule required to implement the change after NRC approval. Items 1-5 above fulfill the seven day requirements of your letter of July 6, 1979.

Yours very truly,

Lowell E Roe

LER:TJM

Attachments

368 138

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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 Technical Specifications, Appendix A to Facility Operating License No. NPF-3, together with the Safety Evaluation for the requested change. The proposed changes include 1) addition of requirements on the interim anticipatory reactor trip system in sections 3.3.2.3, 4.3.2.3 and Table 3.3-15 and Table 4.3-15; 2) addition of requirements for auxiliary feedwater flow indications in section 4.7.1.2; 3) change to reactor coolant high pressure reactor trip setpoint in Table 2.2-1; 4) addition of a reactor coolant pressure temperature curve, Figure 3.4-5; 5) addition of requirements in sections 3.4.3 and 4.4.3 for the electromatic relief valve; 6) change to technical specification basis 2.2.1; and 7) change to SFAS actuation mode and actions of Tables 3.3-3 and 4.3-2.

By Lowell E. Roe
Vice President, Facilities Development

Sworn to and subscribed before me this thirteenth day of July, 1979.

Linda L. Costell
Notary Public

LINDA L. COSTELL
Notary Public — State of Ohio
My Commission Expires Feb. 9, 1982

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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 Technical Specifications, Appendix A to Facility Operating License No. NPF-3, together with the Safety Evaluation for the requested change. The proposed changes include 1) addition of requirements on the interim anticipatory reactor trip system in sections 3.3.2.3, 4.3.2.3 and Table 3.3-15 and Table 4.3-15; 2) addition of requirements for auxiliary feedwater flow indications in section 4.7.1.2; 3) change to reactor coolant high pressure reactor trip setpoint in Table 2.2-1; 4) addition of a reactor coolant pressure temperature curve, Figure 3.4-5; 5) addition of requirements in sections 3.4.3 and 4.4.3 for the electromatic relief valve; 6) change to technical specification basis 2.2.1; and 7) change to SFAS actuation mode and actions of Tables 3.3-3 and 4.3-2.

By s/ Lowell E. Roe
Vice President, Facilities Development

Sworn to and subscribed before me this thirteenth day of July, 1979.

Linda L. Costell
Notary Public

LINDA L. COSTELL
Notary Public — State of Ohio
My Commission Expires Feb. 9, 1982

Docket No. 50-346
License No. NPF-3
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July 13, 1979

I Addition to Davis-Besse Nuclear Power Station Unit NO. 1 Technical Specifications, Appendix A of Technical Specification 3.3.2.3, Table 3.3-15 and Table 4.3-15 concerning the interim anticipatory reactor trip system. See proposed additions attached.

A. Time Required to Implement-

This change can be effective upon NRC issuance.

B. Reason for Change (Facility Change Request 79-283)

This is to identify limiting conditions for operation and surveillance requirements for the new interim anticipatory reactor trip system.

C. Safety Evaluation

This FCR calls for providing new technical specifications for "Limiting Condition for Operation" and surveillance requirements for the newly installed interim Anticipatory Reactor Trip System (ARTS).

The new technical specifications are considered adequate to demonstrate and ensure the continued operability of ARTS. The surveillance requirements specified for this system ensure that the overall system functional capability is maintained and the periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

This is not an unreviewed safety question.

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TABLE 4.3-15

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. Turbine Trip	N.A.	N.A.	S/U*	N.A.
2. Steam Generator - Feedwater Differential Pressure - High **	S	R	M	1,2
3. Output Relay	N.A.	N.A.	M	1,2

* This Surveillance will be performed prior to each unit start up if it has not been performed within the last 31 days.

** This Surveillance requirement is satisfied through surveillance requirement 4.3.2.2.1.

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TABLE 3.3-15

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS OF TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Turbine Trip	1	1	1	1*	16
2. Steam Generator Feedwater - Differential Pressure - High	1	1	1	1,2	17
3. Output Relay	4	2	3	1,2	18

ACTION 16 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or reduce reactor power to less than 15% full power within the next 6 hours.

ACTION 17 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

a) The control rod drive trip breaker associated with the inoperable channel is placed in the tripped condition within one hour.

b) The Minimum Channels OPERABLE requirement is met; however, one additional control rod drive trip breaker associated with another channel may be tripped for up to 2 hours for surveillance testing per Specification 4.3.2.3, after reclosing the control rod drive trip breaker opened in a) above.

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* Applicable only above 15% reactor power.

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.3 The Anticipatory Reactor Trip System instrumentation channels of Table 3.3-15 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-15

SURVEILLANCE REQUIREMENTS

4.3.2.3 The Anticipatory Reactor Trip System shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the modes and at the frequencies shown in Table 4.3-15.

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II Changes to Davis-Besse Nuclear Power Station Unit No. 1 Technical Specifications - Appendix A, Section 4.7.1.2 - concerning auxiliary feedwater flow measurement. See proposed changes attached

A. Time Required to Implement

This change can be effective upon NRC issuance

B. Reason for Change (Facility Change Request 79-281)

This is to identify surveillance requirements for the new auxiliary feedwater flow indication.

C. Safety Evaluation

The proposed surveilliance on this instrumentation is a 31 day CHANNEL FUNCTIONAL TEST and an 18 month CHANNEL CALIBRATION as defined in Davis-Besse Unit 1 Technical Specifications. This surveillance frequency is considered adequate to demonstrate and ensure the continued operability of the subject flow instrumentation.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two inde... steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each steam turbine driven pump develops a differential pressure of $\geq$ 1070 psid on recirculation flow when the secondary steam supply pressure is greater than 800 psia, as measured on PI SP 12B for pump 1-1 and PI SP 12A for pump 1-2.

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

3. Performing a CHANNEL FUNCTIONAL TEST on the auxiliary feedwater flow instrumentation

b. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on an auxiliary feedwater actuation test signal.

2. Verifying that each pump starts automatically upon receipt of an auxiliary feedwater actuation test signal.

3. Performing a CHANNEL CALIBRATION on the auxiliary feedwater flow instrumentation.

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III Changes to Davis-Besse Nuclear Power Station, Unit 1 Technical Specifications. Appendix A, Table 2.2-1 concerning the reactor coolant high pressure reactor protection system trip setpoint.

A. Time Required to Implement

This change can be effective upon NRC issuance

B. Reason for Change (Facility Change Request 79-170)

The reactor coolant high pressure reactor protection system trip setpoint was reduced in order to decrease the number of transients that could open the pressurizer pilot operated relief valve. This change reduces the upper bound of the current technical specification to be consistent with the recently revised setpoint.

C. Safety Evaluation

This change provides for lowering the Reactor Coolant System (RCS) high pressure trip setpoint from 2351.4 psig to 2296.4 psig in the Reactor Protection System (RPS).

From the calculations performed by B&W it is evident that if the RPS high pressure trip setpoint is changed from 2351.4 psig to 2296.4 psig the peak pressurizer pressure will remain below 2345 psig during loss of feedwater transients. The severity of these transients is reduced by implementing the change proposed by this FCR as described below.

In the case of loss of main feedwater the Steam and Feedwater Rupture Control System (SFRCS) isolates both steam generators (SG) on the feedwater and steam side, and auxiliary feedwater is initiated to the SGs. Because of the insufficient heat transfer to the secondary side of the SGs caused by loss of feedwater, the reactor is tripped on high RCS pressure. It is estimated that with the present RPS high pressure trip setpoint the reactor will trip approximately 20 seconds after a loss of feedwater from 100% power as compared to 13 seconds with the lower RPS high pressure trip setpoint. Therefore, with the lower setpoint there is less energy (corresponding to the seven second time difference) contained in the RCS.

On the secondary side the steam pressure builds up because of main steam line isolation and the code safety valves lift (on the secondary side) to relieve the pressure. Since the energy content of the RCS is lower for the lower RPS trip setpoint, the secondary side code safety valves lift for a shorter duration. Consequently, when the secondary system code safety valves reseat, the secondary side level in the steam generators is higher for the lower RPS trip setpoint as compared to that which would be obtained with the present (higher) RPS trip setpoint. Also, when the level in the steam generator is brought to normal through auxiliary feedwater operation for the case of the lower RPS trip setpoint, a smaller amount of cold auxiliary feedwater will be added to the steam generators, thereby reducing the net volumetric contraction of the RCS.

C. Safety Evaluation (Continued)

This will result in a higher minimum pressurizer level during such a transient with the lower RPS trip setpoint.

It should be noted that the RCS temperature and pressure remain the same after the secondary system code safety valves have reseated regardless of the higher or lower RPS high pressure trip setpoint. This assumes that the pressurizer electromatic relief valve does not actuate during the transient. With the increase in the relief setpoint of the electromatic relief valve to 2400 psig, actuation of the relief valve will be eliminated during a loss of feedwater transient. Also, with the increase in electromatic relief valve actuation setpoint the amount of reactor coolant lost to the pressurizer quench tank on lifting of the electromatic relief valve will be eliminated, leading to higher reactor coolant inventory in the RCS to maintain a higher pressurizer level.

In summary, the proposed reduction in the RPS high pressure trip setpoint does not degrade the safety of the plant and does not invalidate any of the safety analyses presented in the Davis-Besse Unit 1 FSAR or in the safety evaluation submitted to the NRC on December 22, 1978 (Serial No. 475). The possibility of an accident or a malfunction of a different type than any evaluated in the FSAR is not created. Also, the margin of safety as defined in the bases for technical specification is not reduced. Pursuant to the above, the proposed change does not involve an unreviewed safety question.

Because 2300 psig in Table 2.2-1 is less than existing trip setpoint in license of 2355 psig, the setpoint change can be made prior to receiving the license amendment.

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. High Flux	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.6% of RATED THERMAL POWER with four pumps operating#
	$\leq$ 80.7% of RATED THERMAL POWER with three pumps operating	$\leq$ 80.8% of RATED THERMAL POWER with three pumps operating#
	$\leq$ 53.0% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 53.1% of RATED THERMAL POWER with one pump operating in each loop#
3. RC High Temperature	$\leq$ 619°F	$\leq$ 619.08°F#
4. Flux - Δ Flux-Flow(1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2.#
5. RC Low Pressure(1)	$\geq$ 1985 psig	$\geq$ 1984.0 psig* $\geq$ 1976.5 psig**
6. RC High Pressure	$\leq$ 2300 psig ~~$\leq$ 2355 psig~~	$\leq$ 2301.0 psig* $\leq$ 2308.5 psig** ~~$\leq$ 2356.0 psig*~~ ~~$\leq$ 2363.5 psig**~~
7. RC Pressure-Temperature(1)	$\geq$ (16.25 T_{out} °F - 7873) psig	$\geq$ (16.25 T_{out} °F - 7873.64) psig#

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IV Change to Davis-Besse Nuclear Power Station, Unit 1 Technical Specifications, Appendix A - Section 3.4.9.3, Figure 3.4-5 and Bases section 3/4 4.9 - concerning reactor coolant pressure and temperature limitations during emergency conditions - see attached proposed changes.

A. Time Required to Implement

This change can be effective upon NRC issuance.

B. Reason for Change (Facility Change Request 79-287)

This change is proposed as a result of discussions with staff members of Nuclear Reactor Regulation and Office of Inspection and Enforcement.

C. Safety Evaluation

See Attached

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Page One of Two SAFETY EVALUATION

In certain small loss of coolant accidents, there may be no forced or natural circulation of the reactor coolant, and water will be added to the reactor coolant system by high pressure injection pumps and/or by the makeup pumps. Under these emergency/faulted conditions, care must be taken to assure that pressure and temperature conditions at the reactor pressure vessel are maintained to avoid conditions which could lead to propagation of flaws by brittle fracture.

During the situations of concern, it is assumed that normal RCS loop circulation may be interrupted. Under these conditions, the RCS loop temperature sensors in the hot and cold legs cannot be relied upon as accurate indications of temperature conditions at the reactor vessel wall. Specifically, with interrupted flow, the cold leg reactor coolant system temperature indicator, which is the normal point of reference for management of brittle fracture limits, cannot be relied upon to reflect the reactor vessel conditions accurately. This is due to the fact that the ECCS injection point is down-stream of the RTD. However, it can be shown for all conditions in which the reactor vessel is filled with water at least to the level of the inlet nozzles that the expected mixed average temperature in the reactor vessel downcomer will be not less than 150°F colder than the temperature at the core exit. This is true because the outlet plenum vent valves in Davis-Besse Unit 1 will open in response to development of pressure differences resulting from thermal temperature differences between the reactor vessel outlet and inlet and permit the recirculation of steam or water through the vent valves and mixing with the cold water being injected from the high pressure injection system prior to entering the downcomer. Given the 150°F temperature limit, the operator may read an appropriate sampling of core exit thermocouples and infer from his measurement that downcomer temperature will be not more than 150°F colder than the temperature indicated by the core exit thermocouples.

With the information available to him, the operator should take action if required, by reducing the injection rate of high pressure injection water and/or makeup water to avoid exceeding appropriate pressure and temperature limits. Appendix G of 10CFR50 states that this Appendix is "to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests,. . . ." Consequently, under the emergency/faulted conditions described above, the normal pressure-temperature limits do not apply. The modified pressure-temperature curve in Figure 3.4-5 was prepared to reflect the margins allowed under faulted conditions. The curve in Figure 3.4-5 still contains adequate margins to the real limits because a number of conservatisms were used in the calculation. The use of a modified pressure-temperature curve is in accordance with paragraph 2.2.5 page 5.3.2-17 of the NRC Standard Review Plan. The pressure and temperature limits in Figure 3.4-5 were derived by removing the factor of two conservatism from allowable pressure which is prescribed by Appendix G. In all other respects, the curve has been calculated to meet the prescription of Appendix G for the Davis-Besse Unit 1 reactor vessel.

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The pressure temperature line for saturated water has been plotted in Figure 3.4-5 to indicate that this line is well below the acceptable limit curve for avoidance of reactor vessel brittle fracture concerns during small LOCA events. Because the saturation line is well away from the brittle fracture limit, it is evident that significantly subcooled pressure and temperature conditions, i.e., >>50F, must be reached at the core outlet before the brittle fracture limits are approached. Having established these conditions, the operator may throttle high pressure injection to maintain the reactor coolant system pressure below the allowable pressure temperature curve and still be confident that adequate core cooling is being maintained.

The change does not constitute an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in FSAR, has not been increased.
2. The possibility of an accident or malfunction of a different type other than any evaluated previously in the FSAR has not been created.
3. The margin of safety as defined in the basis for any technical specification has not been reduced.

R Lund
7-11-79
CR Domeck /4M
7/11/79

POOR ORIGINAL

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM EMERGENCY/FAULTED OPERATION

LIMITING CONDITION FOR OPERATION

3.4.9.3 In the emergency/faulted condition that there is no forced or natural circulation in the reactor coolant system and there is high pressure injection and/or makeup addition, the Reactor Coolant System temperature and pressure shall be limited in accordance with the limit line shown on Figure 3.4-5. Under the above emergency/faulted conditions, Figure 3.4-3 will not apply.

APPLICABILITY: Modes 3, 4, and 5

ACTION: With the above limit exceeded, throttle the high pressure injection flow and/or makeup flow so that the pressure and temperature are within the acceptable limits within 30 minutes.

SURVEILLANCE REQUIREMENTS

4.4.9.3 The Reactor Coolant System temperature and pressure shall be determined to be within the acceptable limits at least once per 30 minutes during the emergency condition.

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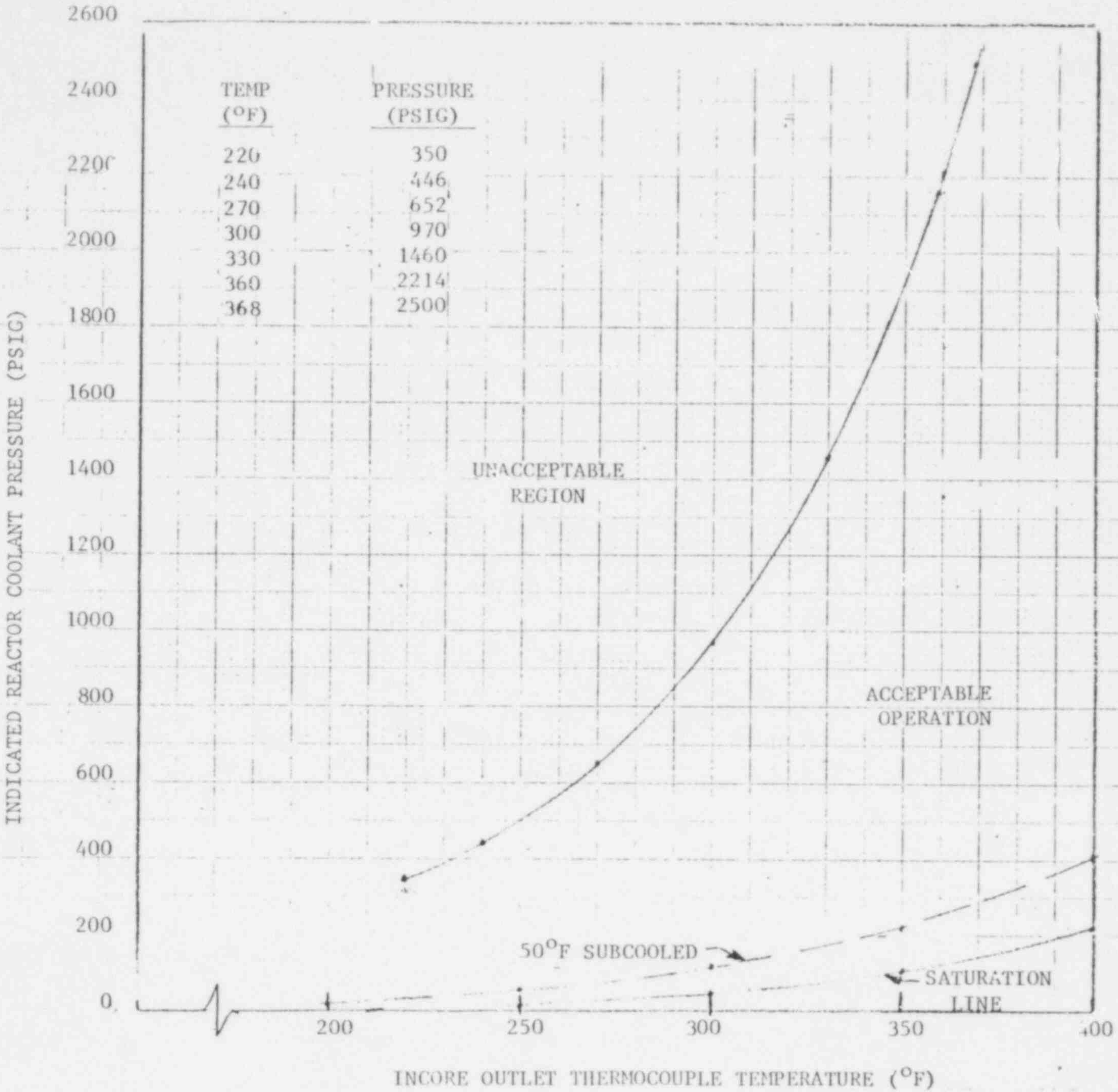

Figure 3.4-5 Reactor Coolant System Emergency Pressure/Temperature Limit Curve. Applicable for two effective full power years after June, 1979.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The unirradiated impact properties and residual elements of the beltline region materials are listed in Bases Table 4-1. The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents. Bases Figure 4-1 illustrates the calculated peak neutron fluence, at several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules as a function of exposure time.

Bases Figure 4-2 illustrates the design curves for predicting the radiation-induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. The adjusted RT_{NDT}'s of the beltline region materials at the end of the fifth full power year are listed in Bases Table 4-1. The adjusted RT_{NDT}'s are given for the 1/4T and 3/4T (T is wall thickness) vessel wall locations. The assumed RT_{NDT} of the closure head region is 40°F and the outlet nozzle steel forgings is 60°F.

During cooldown at the higher temperatures, the limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are segments D-E and D-F of the limit lines on Figures 3.4-2 and 3.4-4, respectively. These limits will not require adjustments due to the neutron fluences.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure-temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

~~All~~ The pressure-temperature limit curves in Figures 3.4-2, 3.4-3 and 3.4-4 are applicable up to the fifth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, 3.4-3 and 3.4-4.

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BASES

The limitations to prevent non-ductile failure during emergency/faulted operation when there is no forced or natural circulation in the reactor coolant system and there is high pressure injection and/or makeup addition established. Figure 3.4-5 takes into consideration that the reactor coolant system loop temperature sensors in the hot and cold legs cannot be relied upon as accurate indications of temperature conditions at the reactor vessel wall. The pressure/temperature limitations for this transient are given in Figure 3.4-5. The temperature scale in the curve has been shifted upward by 150°F to conservatively account for the temperature difference between the expected mixed average temperature in the reactor vessel downcomer and the temperature at the core exit. The pressure/temperature limits in Figure 3.4-5 were derived by removing the factor of two conservatism from allowable pressure which is prescribed in Appendix G. In other respects, the curve has been calculated to meet the prescription of Appendix G.

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V Change to Davis-Besse Nuclear Power Station, Unit 1 Technical Specifications, Appendix A - Section 3.4.3; 4.4.3 concerning pressurizer, electromatic relief valve setpoints and the associated Bases section - see proposed change attached.

A. Time Required to Implement

This change can be effective upon NRC issuance.

B. Reason for change (Facility Change Request 79-282)

This change adds the associated setpoints for the pressurizer electromatic relief valve.

C. Safety Evaluation

See attached

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SAFETY EVALUATION

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 to 2350 psig. Also, the LOFW in 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 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 ≥ 2385.5 psig is obtained by subtracting 4.5 psi due to the drift from the trip setpoint of ≥ 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.

The above values for the pressurizer electromatic relief valve in conjunction with a 2300 psig RPS high pressure trip setpoint will avoid actuation of the pressurizer electromatic relief valve during anticipated transients. All safety analyses for Davis-Besse Unit 1 assume that the vent capacity of the pressurizer electromatic relief valve will not be available; thus, these analyses are unchanged by an increase in its setpoint.

The change does not constitute an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in FSAR, has not been increased.
2. The possibility of an accident or malfunction of a different type other than any evaluated previously in the FSAR has not been created.
3. The margin of safety as defined in the basis for any technical specification has not been reduced.

R Lund
CR Domeck/FM
7/11/79

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SAFETY VALVES ~~OPERATING~~ 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 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 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 ~~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.

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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 to 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 ≥ 2385.5 psig is obtained by subtracting 4.5 psi due to the drift from the trip setpoint of ≥ 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 237[illegible] [illegible]ig 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.

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VI Change to Davis-Besse Nuclear Power Station, Unit 1, Technical Specifications, Appendix A - Bases Section 2.2.1 (page B2-6) concerning revision to the basis of reactor coolant high pressure reactor trip (subject of Attachment III) and the reactor coolant pressure-temperature trip setpoints - see proposed change attached.

A. Time to Implement

This change can be effective upon NRC issuance.

B. Reason for Change (Facility Change Request 79-174)

This change is to update the bases section of the technical specifications to be consistent with current plant conditions.

C. Safety Evaluation

This change calls for making the following changes to the Davis-Besse Unit 1 Technical Specifications:

1) On page B2-6, Reactor Coolant System (RCS) pressure-temperature trip setpoint should be changed from (13.01 T_{out} ^{o}F -5973) psig to (16.25 T_{out} ^{o}F - 7873) psig. This change was made to Table 2.2-1 of the technical specifications through license amendment 11 (dated June 16, 1978) following the removal of burnable poison rod assemblies and orifice rod assemblies from the core.

2) Through attachment, the Reactor Protection System (RPS) high pressure trip setpoint is being changed from 2355 to 2300 psig.

Safety evaluations have already been performed concluding that these are not unreviewed safety questions.

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LIMITING SAFETY SYSTEM SETTINGS

BASES

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.

Δ 2300 psig

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, ~~2355 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.

The RC Low Pressure, 1985 psig, and RC Pressure-Temperature ~~($13.01 T_{out}\,°F - 5973$)~~ psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.32 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 1.32 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.

REPLACE THE MATHEMATICAL EXPRESSION WITH Δ $(16.25\ T_{out}\,°F - 7873)$

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Docket No. 50-346
License No. NPF-3
Serial No. 527
July 13, 1979

VII Change to Davis-Besse Nuclear Power Station, Unit 1 Technical Specifications - Appendix A Tables 3.3-3 and 4.3-2 concerning SFAS incident level 5 actuation - see proposed change attached

A. Time Required to Implement

This change could be effective after regional verification of equipment modification. This would be expected during the first planned outage of sufficient duration eight months after NRC technical specification approval.

B. Reason for Change (Facility Change Request 79-171)

This change is proposed to reduce the possibility of a false SFAS incident level 5 trip in appropriately transferring decay heat and containment spray suctions to the containment emergency sump for recirculation.

C. Safety Evaluation

See attached

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SAFETY EVALUATION

At the present time Technical Specification 3.3.2.1 (Safety Features Actuation System), Table 3.3-3 requires that an inoperable Borated Water Storage Tank (BWST) Level-low Instrument String or an inoperable Incident Level 5, Containment Sump Recirculation Output Logic, be placed in the tripped condition within one hour. This subjects the unit to a false trip of Containment Sump Recirculation (Incident Level 5) if an additional active failure is postulated. This false trip would produce a more serious safety consequence than a failure to trip, as it would transfer the suction of both trains of decay heat (DH) and containment spray (CS) pumps to the dry Containment Emergency Sump.

This request for a Technical Specification change proposes that the action for the BWST Level-low Instrument String be changed from 9 to a new 15 (attached). The action for Incident Level 5, Containment Sump Recirculation Output Logic, will also be changed from 10 to 15. Also, Mode 4 should be eliminated from the applicable modes for the Incident Level 5, Containment Sump Recirculation Output Logic, since the Instrument String for the BWST Level-low is not required in Mode 4 and there is no Incident Level 5 Manual Trip. The change from Action 9 to the new Action 15 will allow the BWST Level-low Instrument String to be bypassed instead of tripped if it becomes inoperable and the Incident Level 5, Containment Sump Recirculation Output Logic, to be blocked if it is inoperable.

If a BWST Level-low Instrument String is bypassed, the other three redundant BWST Level-low Instrument Strings can cause a 2/3 trip of SFAS Incident Level 5, which will cause the automatic transfer of the DH and CS pumps suctions from the BWST to the Containment Emergency Sump. If an Incident Level 5, Containment Sump Recirculation Output Logic is blocked, then one train of DH and CS pumps will not be automatically transferred from the BWST to the Containment Emergency Sump on an Incident Level 5 trip.

The operator will then have to manually transfer this train of DH and CS pumps when the redundant train is automatically transferred. The following alarms and indications in the Control Room would indicate to the operators when to make this manual transfer:

1. Control Room indicating lights and status board would indicate that the Containment Sump Recirculation Output Logic was blocked.

2. Annunciator and computer alarms would come on when SFAS started the HPI, DH and CS pumps.

3. There would be at least three out of four BWST level indications in the Control Room.

4. Annunciator and computer alarms would come on when the unaffected train was automatically transferred to the Containment Sump.

5. The operator would then manually transfer the affected train as follows:

   a. Stop HPI, DH and CS pumps in the train not transferred.

   b. Close BWST outlet valve (DH7A or B)

   c. Open containment emergency sump outlet valve (DH9A or B)

   d. Restart DH and CS pumps.

   e. Place HPI and DH pumps in "Piggy Back" mode if necessary.

This manual transfer would have to be made about 23 minutes after the initial SFAS trip that started all HPI, DH and CS pumps. This change will improve the safety of the unit by reducing the possibility of a false trip of SFAS Incident Level 5, which would improperly transfer the suction of the HPI, DH and CS pumps to a dry Containment Emergency Sump. It is not an unreviewed safety issue.

dh c/3-4

Sushil Jain
6/18/79

CRDomeck/FM 7/11/79

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Technical Specification Change previously requested by Toledo Edison letter of 3/23/79

TABLE 3.3-3

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF UNITS	UNITS TO TRIP	MINIMUM UNITS OPERABLE	APPLICABLE MODES	ACTION
1. INSTRUMENT STRINGS					
a. Containment Radiation - High	4	2	3	~~All MODES~~ ****	~~9#~~ 13#
b. Containment Pressure - High	4	2	3	1, 2, 3	9#
c. Containment Pressure - High-High	4	2	3	1, 2, 3	9#
d. RCS Pressure - Low	4	2	3	1, 2, 3*	9#
e. RCS Pressure - Low-Low	4	2	3	1, 2, 3**	9#
f. BWST Level - Low	4	2	3	1, 2, 3	~~9#~~ 15
2. OUTPUT LOGIC					
a. Incident Level #1: Containment Isolation	2	1	2	All MODES	10
b. Incident Level #2: High Pressure Injection and Starting Diesel Generators	2	1	2	1, 2, 3, 4	10
c. Incident Level #3: Low Pressure Injection	2	1	2	1, 2, 3, 4	10
d. Incident Level #4: Containment Spray	2	1	2	1, 2, 3, 4	10
e. Incident Level #5: Containment Sump Recirculation	2	1	2	1, 2, 3, ~~4~~	~~10~~ 15

TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 1800 psig. Bypass shall be automatically removed when RCS pressure exceeds 1800 psig.

** Trip function may be bypassed in this MODE with RCS pressure below 600 psig. Bypass shall be automatically removed when RCS pressure exceeds 600 psig.

*** One must be in SFAS Channels #1 or #3, the other must be in Channels #2 or #4.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE functional units one less than the Total Number of Units operation may proceed provided both of the following conditions are satisfied:

a. The inoperable functional unit is placed in the tripped condition within one hour.

b. The Minimum Units OPERABLE requirement is met; however, one additional functional unit may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 10 - With any component in the Output Logic inoperable, trip the associated components within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 11 - With the number of OPERABLE Units one less than the Total Number of Units, restore the inoperable functional unit to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 12 - a. With less than the Mimimum Units OPERABLE and reactor coolant pressure $\geq$ 413 psig, both Decay Heat Isolation Valves (DH11 and DH12) shall be verified closed.

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b. With Less than the Minimum Units OPERABLE and reactor coolant pressure < 413 psig operation may continue; however, the functional unit shall be OPERABLE prior to increasing reactor coolant pressure above 413 psig.

ACTION 15- (see next page)

ACTION 15 - With the number of OPERABLE units one less than the total number of units, bypass the inoperable functional unit or block the inoperable output logic; and restore the inoperable functional unit or output logic to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours. With the number of OPERABLE units one less than the total number of units, one additional functional unit may be bypassed or blocked for up to two hours for surveillance testing per Specification 4.3.2.1.1 The following combinations of two output logics may be blocked simultaneously for surveillance testing:

1. Channels 1 and 3
2. Channels 2 and 4

No other combinations may be blocked simultaneously.

dh d/5

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TABLE 4.3-2

SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. INSTRUMENT STRINGS				
a. Containment Radiation - High	S	R	M	All MODES
b. Containment Pressure - High	S	R	M(2)	1, 2, 3
c. Containment Pressure - High-High	S	R	M(2)	1, 2, 3
d. RCS Pressure - Low	S	R	M	1, 2, 3
e. RCS Pressure - Low-Low	S	R	M	1, 2, 3
f. BWST Level - Low	S	R	M	1, 2, 3
2. OUTPUT LOGIC				
a. Incident Level #1: Containment Isolation	S	R	M	All MODES
Incident Level #2: High Pressure Injection and Starting Diesel Generators	S	R	M	1, 2, 3, 4
Incident Level #3: Low Pressure Injection	S	R	M	1, 2, 3, 4
Incident Level #4: Containment Spray	S	R	M	1, 2, 3, 4
e. Incident Level #5: Containment Sump Recirculation	S	R	M	1, 2, 3, Ø
3. MANUAL ACTUATION				
a. SFAS (Except Containment Spray and Emergency Sump Recirculation)	NA	NA	M(1)	All MODES
b. Containment Spray	NA	NA	M(1)	1, 2, 3
4. SEQUENCE LOGIC CHANNELS	S	NA	M	1, 2, 3, 4

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