AmerGen

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An Exelon/British Energy Company

10 CFR 50.90

January 15, 2001

5928-00-20355

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE No. DPR-50
DOCKET No. 50-289
LICENSE AMENDMENT REQUEST NO. 303 -
PRESSURE-TEMPERATURE PROTECTIVE LIMITS

In accordance with 10 CFR 50.4(b)(1), enclosed is License Amendment Request No. 303.

The purpose of this License Amendment Request is to revise the TMI Unit 1 Technical Specifications to relocate the variable low RCS pressure-temperature core protection safety limits from the Technical Specifications to the existing TMI Unit 1 Core Operating Limits Report (COLR). The proposed change is in accordance with NRC Generic Letter (GL) 88-16 guidance with regard to placing cycle-dependent parameters into the COLR and the NRC-approved Topical Report BAW-10179 P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses." It is noted that NRC has previously issued a similar amendment for ANO-1 (Amendment No. 186, dated October 3, 1996).

The proposed change also reintroduces the variable low RCS pressure trip (VLPT) function to the Technical Specification with the actual setpoint maintained in the COLR.

Using the standards in 10 CFR 50.92, AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this License Amendment Request is provided to the designated official of the Commonwealth of

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Pennsylvania, Bureau of Radiation Protection, as well as the chief executive of the township and county in which the facility is located.

AmerGen requests that this license amendment application be approved by April 1, 2001 to support Cycle 14 core design analysis and reload report finalization, and to preclude the need for a cycle specific amendment request upon final development of the Cycle 14 limits.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

Very truly yours,

Mark E. Warner

Mark E. Warner
Vice President, TMI Unit 1

MEW/djd

Enclosure: 1) Safety Evaluation and No Significant Hazards Consideration
2) Affected TMI Unit 1 Technical Specification Pages

cc: H. J. Miller, Administrator, USNRC Region I
T. G. Colburn, USNRC Senior Project Manager, TMI Unit 1
J. D. Orr, USNRC Senior Resident Inspector, TMI Unit 1
File No. 00126
D. Allard, Director, Bureau of Radiation Protection -
PA Department of Environmental Resources
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 303

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF DAUPHIN)

This License Amendment Request is submitted in support of Licensee's request to change the Technical Specifications for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed marked up pages for the TMI Unit 1 Technical Specifications are also included. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

AmerGen Energy Company, LLC

By: Mark E. Warner
Vice President, TMI Unit 1

Sworn and Subscribed to before me

this 15th day of January 2001.

Linda C. Witter
Notary Public

Notarial Seal
Linda C. Witter, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Sept. 25, 2004
Member, Pennsylvania Association of Notaries

ENCLOSURE 1

TMI Unit 1 License Amendment Request No. 303

Safety Evaluation and No Significant Hazards Consideration

I. License Amendment Request No. 303

AmerGen Energy Company, LLC (AmerGen) requests that the following changed replacement pages be inserted into the existing Technical Specification:

Revised Technical Specification pages: vi, vii, 2-1, 2-2, 2-3, 2-4a, 2-4c, 2-7, 2-8, 2-10, 2-11, 3-29, 3-30, and 4-4

Marked up pages showing the requested changes are provided in Enclosure 2.

II. Reason for Change

The proposed change relocates Technical Specification Figures 2.1-1, 2.1-3, and 2.3-1 to the TMI Unit 1 Core Operating Limits Report (COLR). The initial implementation of the COLR for TMI Unit 1 was approved by the NRC as Amendment No. 150, dated July 6, 1989. The TMI Unit 1 COLR is established and controlled in accordance with NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988, which allowed licensees to remove cycle-specific parameters from Technical Specifications and place them in a COLR provided the limits are developed using NRC-approved methods. Changes planned for future TMI Unit 1 reload core designs (e.g., Mk-B12 fuel with fine mesh debris filters in Cycle 14 and implementation of Framatome Cogema Fuel's Statistical Core Design Methodology in Cycle 15) are expected to result in periodic revisions to the variable low reactor coolant system (RCS) pressure-temperature core protection safety limits. Therefore, AmerGen is requesting that the variable low RCS pressure-temperature core protection safety limits be considered cycle-specific parameters and relocated to the TMI Unit 1 COLR. TMI Unit 1 Technical Specification Section 6.9.5 provides existing controls for the COLR in accordance with GL 88-16.

It is noted that the NRC has approved a similar Technical Specification change for Arkansas Nuclear One, Unit No. 1, Amendment No. 186, dated October 3, 1996.

The proposed change also adds the variable low RCS pressure trip (VLPT) function to the Technical Specifications with the actual setpoint maintained in the COLR. The VLPT was removed from the TMI Unit 1 Technical Specifications and approved by NRC in Amendment No. 142, dated July 18, 1988. However, future TMI Unit 1 reload core designs may cause the VLPT setpoint to become more limiting than the low RCS pressure and high RCS temperature RPS trip setpoints. Therefore, the proposed change will preclude the need for a future Technical Specification amendment. The VLPT instrumentation requirements are being reinstated with appropriate footnotes to identify applicability of the specification and surveillance requirements.

Technical Specification pages vi and 2-10 are editorially revised to correct the page number associated with Table 2.3-1 and to add the word "Table" in the page title, respectively. The last paragraph of Technical Specification page 2-7, second line, the word "setpoint" is editorially revised to indicate the plural "setpoints."

III. Safety Evaluation Justifying Change

The variable low RCS pressure-temperature protective limits define a locus of points for which the minimum steady-state departure from nucleate boiling ratio (DNBR) is greater than or equal to the DNBR analysis limit for the critical heat flux (CHF) correlation being used. These points are calculated for the maximum overpower condition and limiting reactor coolant pump operating configurations. The difference between the actual core outlet pressure and the indicated RCS pressure is accounted for in determining these points. As stated in the bases for Technical Specification 2.1, the minimum DNBR value during steady-state operation, normal operational transients and anticipated transients must be greater than the limits specified for the appropriate CHF correlation. This is ensured by maintaining RCS pressure and core outlet temperature within the variable low RCS pressure-temperature protective limits of existing Technical Specification Figures 2.1-1 and 2.1-3. It is noted that the safety limits (i.e., minimum DNBR limits) are not changing or being removed from the Technical Specifications; only the associated protective limits that ensure that the safety limits are met will be moved to the COLR and will be subject to cycle-specific changes. Therefore, the variable low RCS pressure-temperature protective limits of Figures 2.1-1 and 2.1-3 are being relocated to the COLR.

Prior to its removal from TMI Unit 1 Technical Specifications, the VLPT setpoint, in conjunction with the low RCS pressure and high RCS temperature trip setpoints, defined the worst-case combination of RCS pressure and core outlet temperature conditions that might exist at steady-state conditions to preclude departure from nucleate boiling for the limiting fuel pin in the core. The VLPT setpoint assured automatic enforcement of the core protection limits in the event the variable low RCS pressure-temperature protective limit was more restrictive than the low RCS pressure and high RCS core outlet temperature limits contained in existing Technical Specification Table 2.3-1 and Figure 2.3-1. The VLPT setpoint became non-limiting in TMI Unit 1 Cycle 7, when it was removed from Technical Specifications (Amendment No. 142) and has remained non-limiting since. Technical Specification Figure 2.3-1 is being relocated to the COLR to preclude an additional Technical Specification Change in the event that the VLPT setpoint becomes limiting in future reload designs. Technical Specification Table 2.3-1 is also being revised to include the VLPT function and will refer to the setpoint being maintained in the COLR. The other trip setpoints indicated on Technical Specification Figure 2.3.1 are unaffected and will continue to be maintained in Table 2.3-1. In addition to reinstating the VLPT function into the Technical Specifications, the corresponding pressure-temperature instrumentation requirements that were deleted from TMI Unit 1 Technical Specification Tables 3.5-1 and 4.1-1 by Amendment 149, dated April 27, 1989, are also being reinstated with appropriate footnotes to identify applicability of the specification and surveillance requirements.

Due to the introduction of the Mk-B12 fuel design in TMI Unit 1 Cycle 14 and changes expected for future reload designs, it is anticipated that the variable low RCS pressure-temperature protective limits and possibly the VLPT setpoint will be revised in order to assure that adequate margins of safety are maintained. These cycle-specific COLR limits and setpoints will be developed using the NRC-approved methods listed in Technical Specification 6.9.5.2. Technical Specifications will continue to require operation within both the core protective and operational limits, and the approved CHF correlation DNBR safety limit will continue to be met. Therefore, the proposed change to TMI-1 Technical Specifications does not adversely affect nuclear safety or safe plant operations.

IV. No Significant Hazards Consideration

AmerGen has determined that this License Amendment Request poses no significant hazards consideration as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The removal of the cycle-dependent variable low RCS pressure-temperature protective limits from Technical Specifications and placing them in the COLR is an administrative change only and has no impact on plant safety. The proposed change does not affect the safety analyses, physical design, or operation of the plant. Technical Specifications will continue to require operation within the core protective and operational limits for each reload cycle as calculated by NRC-approved reload design methods. The appropriate actions required if limits are exceeded will remain in the Technical Specifications. The reload report presents the results of cycle-specific evaluations of accident analyses and transients addressed in the TMI Unit 1 Updated Safety Analysis Report (UFSAR). The cycle-specific changes in fuel cycle design and the corresponding COLR limits are controlled in accordance with TMI Unit 1 Technical Specification Section 6.9.5 and 10 CFR 50.59.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change is administrative in nature. No change to the design configuration or method of operation of the plant is made by this proposed change, and therefore, no new transient initiator has been created. Technical Specifications will continue to require operation within the required core protective and operating limits and appropriate actions will be taken if the limits are exceeded.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

   Existing Technical Specification operability and surveillance requirements are not reduced by the proposed change to relocate the variable low RCS pressure-temperature protective limits and the VLPT setpoint to the COLR. The proposed changes are administrative in nature and do not relate to or modify the safety margins defined in and maintained by the Technical Specifications. The cycle-specific COLR limits for future reload designs will continue to be developed based on NRC-approved methods. The addition of the VLPT function to the Technical Specifications ensures appropriate actions and surveillance requirements are implemented should the VLPT setpoint become limiting in future reload designs.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

V. <u>Information Supporting an Environmental Assessment</u>

An environmental assessment is not required for the proposed change since the proposed change conforms to the criteria for "actions eligible for categorical exclusion" as specified in 10 CFR 51.22(c)(9). The proposed change will have no impact on the environment. The proposed change does not involve significant hazards as discussed in the preceding section. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released off-site.

In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

VI. Implementation

AmerGen requests that the amendment authorizing this change become effective upon issuance and implemented within 30 days to allow the appropriate COLR revision to be issued.

ENCLOSURE 2

Affected TMI Unit 1 Technical Specification Pages

LIST OF TABLES

LIST OF FIGURES

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, axial power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in ~~Figure 2.1-1.~~ the Variable Low RCS Pressure - Temperature Core Protection Safety Limits as specified in the COLR. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and axial power imbalance (power in the top half of core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the protective limit as defined by the locus of points (solid line) for the specified flow set forth in the Axial Power Imbalance Protective Limits given in the Core Operating Limits Report (COLR). If the actual-reactor-thermal-power/axial-power-imbalance point is above the line for the specified flow, the protective limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed, departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in excessive cladding temperature and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Reference 1) and BWC (Reference 2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark BZ type fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits.

The ~~curve presented in Figure 2.1-1 represents~~ Variable Low RCS Pressure-Temperature Core Protection Safety Limit curves presented in the COLR represent the conditions at which the minimum allowable DNBR or greater is predicted for the limiting combination of thermal power and number of operating reactor coolant pumps. These ~~This~~ curves ~~is~~ are based on the nuclear power peaking factors given in Reference 3 and the COLR which define the reference design peaking condition in the core for operation at the maximum overpower. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The Axial Power Imbalance Protective Limits curves in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing:

a. The DNBR limit produced by a total nuclear power peaking factor consisting of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

b. The maximum allowable local linear heat rate that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the axial power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of the Axial Power Imbalance Protective Limits given in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

Variable Low RCS Pressure - Temperature Core Protection Safety Limit for four reactor coolant pumps operating

The ~~curve of Figure 2.1-1~~ is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations ~~shown in Figure 2.1-3.~~ The ~~curves of Figure 2.1-3~~ represent the conditions at which the DNBR limit is predicted 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 is equal to 22 percent, (BAW-2), or 26 percent (BWC) whichever condition is more restrictive.

as specified in the COLR.

Variable Low RCS Pressure - Temperature Core Protection Safety Limits in the COLR

The maximum thermal power for each reactor coolant pump operating condition (four pump, three pump and one pump in each loop) given in the COLR is due to a power level trip produced by the flux-flow ratio multiplied by the minimum flow rate for the given pump combination plus the maximum calibration and instrumentation error.

the Variable Low RCS Pressure - Temperature Core Protection Safety Limits specified in the COLR

Using a local quality limit of 22 percent (BAW-2), or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 3 of ~~Figure 2.1-3~~ is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or BWC correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

combination of operating reactor coolant pumps of the Variable Low RCS Pressure - Temperature Core Protection Safety Limits specified in the COLR,

For each ~~curve of Figure 2.1-3~~, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2), or 26 percent (BWC) for the particular reactor coolant pump ~~situation~~. ~~Curve 1~~ is more restrictive than any other reactor coolant pump ~~situation~~ because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curves.

combination

The Variable Low RCS Pressure - Temperature Core Protection Safety Limit for four reactor coolant pumps operating

REFERENCES

(1) UFSAR, Section 3.2.3.1.1 - "Fuel Assembly Heat Transfer Design"

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, Babcock & Wilcox, Lynchburg, Virginia, April 1985

(3) UFSAR, Section 3.2.3.1.1.3 - "Nuclear Power Factors"

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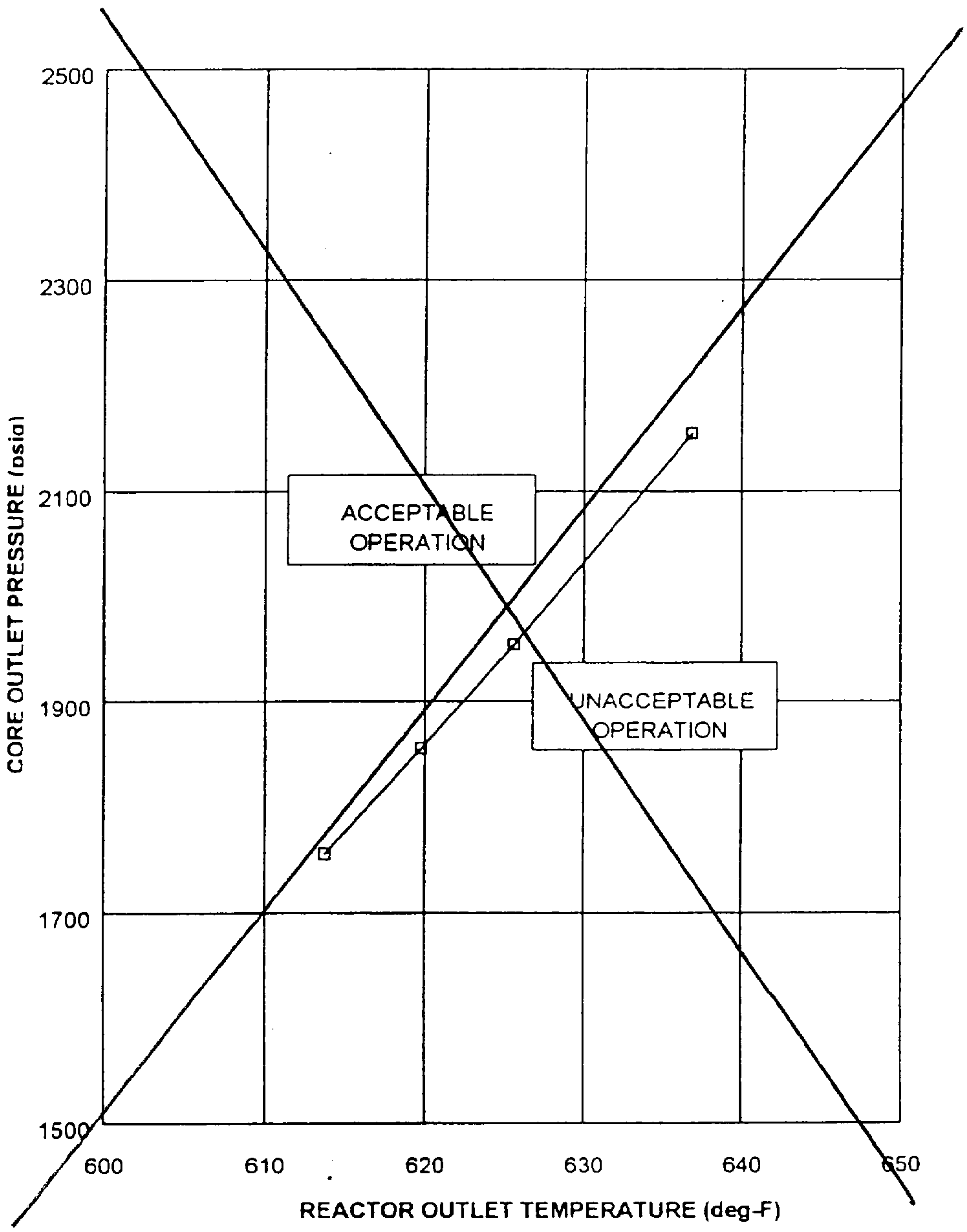

~~CORE PROTECTION SAFETY LIMIT~~
~~TMI-1~~
Figure 2.1-1

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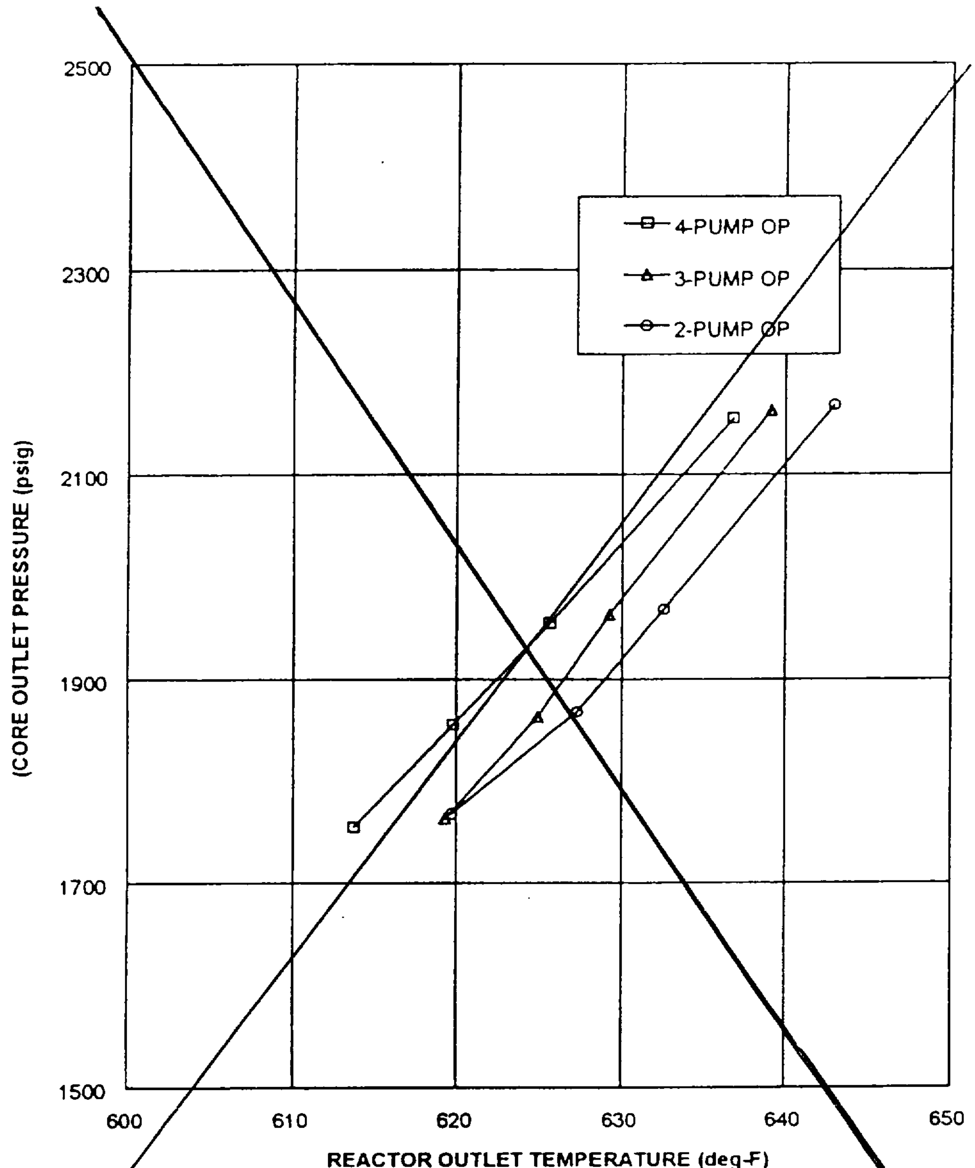

RC Pumps	Reactor Coolant Flow (lbs/hr)	Power	Pumps Operating (Type of Limit)
4	134.02×10^6	112%	Four Pumps (DNBR Limit)
3	See COLR	See COLR	Three Pumps (Quality Limit)
2	See COLR	See COLR	One Pump in Each Loop (Quality Limit)

~~CORE PROTECTION SAFETY BASES~~
~~TMI-1~~
FIGURE 2.1-3

Amendment No. ~~50, 126, 142, 167, 184~~, 214 the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imblance in the COLR are produced.

b. Pump Monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) 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.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in ~~Figure~~ Table 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained below the safety limit (2750 psig) for any design transient (Reference 2). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated overpressurization transients (Reference 3). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

The low pressure (1800 psig) and variable low pressure (11.75 T_{out}-5103) trip setpoints were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (References 4, 5, and 6). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. Application of the B&W

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Variable Low RCS Pressure – Temperature Core Protection Safety Limits specified in ~~the~~ COLR

crossflow model, can ~~resulted~~ result in safety limits (see ~~Figures 2.1-1 and 2.1-3~~) outside the acceptable operating region formed by the low pressure, high pressure, and high temperature trip setpoints ~~(see Figure 2.3-1) which justifies the removal of~~ the variable low pressure trip~~s~~ is unnecessary and its removal is justified.

(see the RCS Pressure-Temperature Protective Maximum Allowable setpoints specified in the COLR). In such circumstances,

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618.8F) shown in ~~Figure~~ Table 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RPS is 520° to 620°F. The trip setpoint of the channel is 618.8F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1.2°F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip set point of 620°F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS (5)

	Four Reactor Coolant Pumps Operating (Nominal Operating) Power - 100%	Three Reactor Coolant Pumps Operating (Nominal Operating) Power - 75%	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, max. % of rated power	105.1	105.1	105.1	5.0(2)
2. Nuclear power based on flow (1) and imblance max. of rated power	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (4) on pump monitors max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2355	2355	2355	1720(3)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Reactor coolant temp. F., max.	618.8	618.8	618.8	618.8
7. High Reactor Building pressure, psig max.	4	4	4	4
8. Variable low reactor coolant system pressure, psig min.				Bypassed

(1) Reactor coolant system flow, %.
(2) Administratively controlled reduction set during reactor shutdown.
(3) Automatically set when other segments of the RPS (as specified) are bypassed.
(4) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.
(5) Trip settings limits are limits on the setpoint side of the protection system bistable connectors.

Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints Figure in COLR

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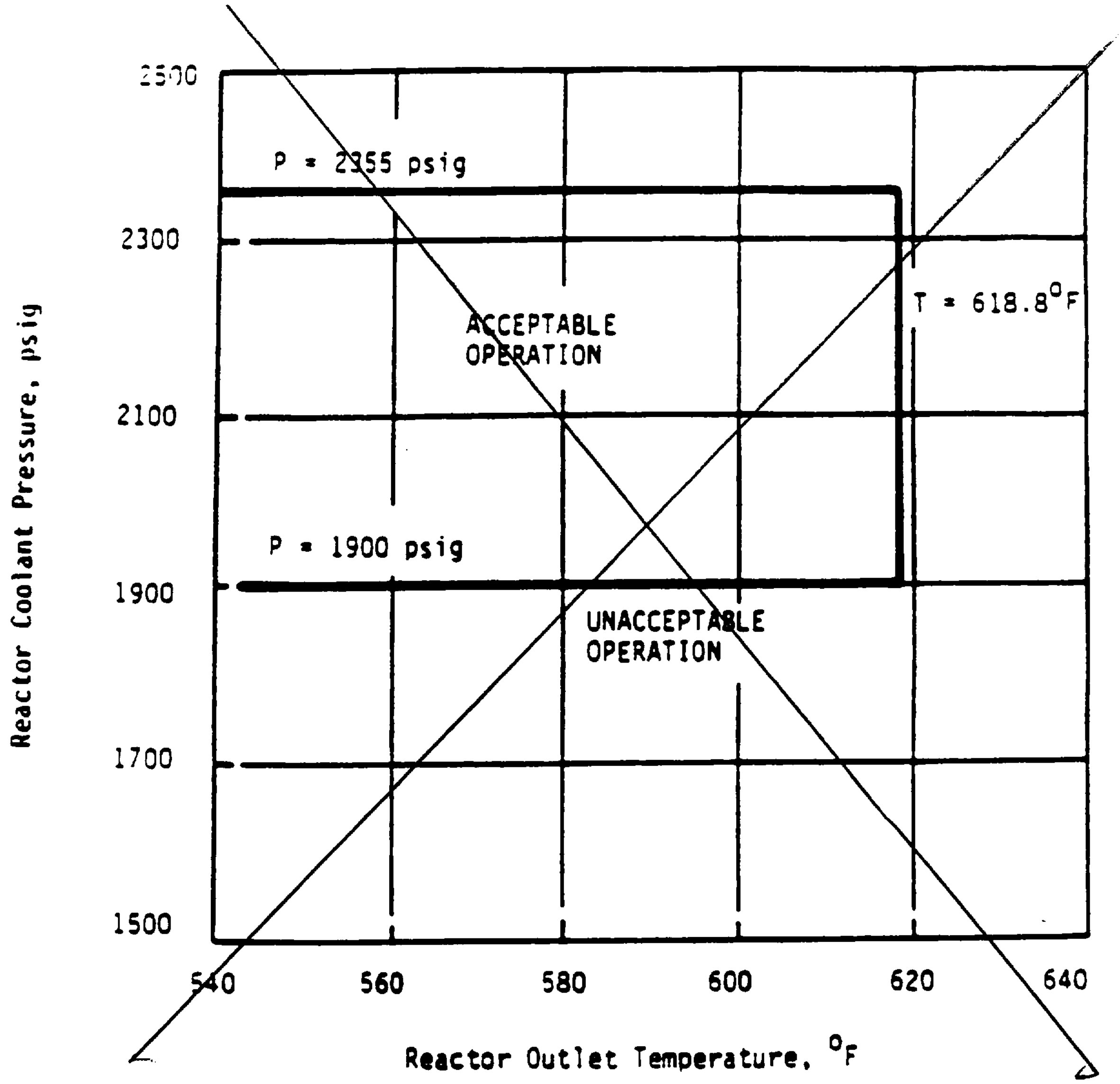

~~TMI-1~~
~~PROTECTION SYSTEM MAXIMUM~~
~~ALLOWABLE SETPOINTS~~

Figure 2.3-1

Amendment No. 13, 17, 28, 39, 45,
78, 126, 135, ~~142~~, 167

TABLE 3.5-1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. <u>Reactor Protection System</u>			
1. Manual pushbutton	1	0	(a)
2. Power range instrument channel	2	1	(a)
3. Intermediate range instrument channels	1	0	(a) (b)
4. Source range instrument channels	1	0	(a) (c)
5. Reactor coolant temperature instrument channels	2	1	(a)
6. ~~(Deleted)~~ Pressure-Temperature Instrument channels	*2	*1	- * - (a)(d)
7. Flux/imbalance/flow	2	1	(a)
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	(a)
b. Low reactor coolant pressure instrument channels	2	1	(a)

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
A. <u>Reactor Protection System (cont'd)</u>			
9. Power/number of pumps instrument channels	2	1	(a)
10. High reactor building pressure channels	2	1	(a)

(a) Restore the conditions of Column (A) and Column (B) within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours.

(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, intermediate range instrumentation is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, or 2 of 4 power range instrument channels are greater than 10 percent full power, source range instrumentation is not required.

(d) This specification applies only when the variable low RCS pressure trip (VLPT) setpoint is specified in the COLR.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	F	
11. ~~(Deleted)~~ Reactor Coolant Pressure – Temperature Comparator	S(2)	M(2) X	R(2) X	(2) This specification applies only when the variable low RCS pressure trip (VLPT) setpoint is specified in the COLR.
12. Pump Flux Comparator	S	M	R	
13. High Reactor Building Pressure Channel	S	M	F	
14. High Pressure Injection Logic Channels	NA	Q	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F.
16. Low Pressure Injection Logic Channel	NA	Q	NA	
17. Lower Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	