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BGE

A Member of the Constellation Energy Group

November 18, 1999

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1; Docket Nos. 50-317
License Amendment Request: Revision to Technical Specification P-T Curves

Pursuant to 10 CFR 50.90, Baltimore Gas and Electric Company (BGE) hereby requests an amendment to Operating License No. DPR-53 to incorporate the changes described below into the Technical Specifications for Calvert Cliffs Unit 1.

DESCRIPTION

The proposed amendment revises the Unit 1 Heatup Curve (Technical Specification Figure 3.4.3-1), Unit 1 Cooldown Curve (Technical Specification Figure 3.4.3-2), and Unit 1 Maximum Power-Operated Relief Valve (PORV) Opening Pressure vs Temperature Curve (Technical Specification Figure 3.4.12-1) to change fluence level from 2.61×10^{19} n/cm^2 to 4.49×10^{19} n/cm^2 (E>1MeV). This change reflects the new actual fluence level for which these curves are valid, and is necessary to extend the applicability of the curves for Unit 1 operation.

BACKGROUND

Appendix G to 10 CFR Part 50 requires the establishment of pressure/temperature (P-T) limits for material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. It requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the methods of analysis and the required margins of safety of the applicable section of the American Society of Mechanical Engineers (ASME) Code. Accordingly, the Calvert Cliffs analytical procedure for developing reactor vessel beltline P-T limits utilizes the methods of linear elastic fracture mechanics and the guidance found in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

Calvert Cliffs Technical Specification Figures 3.4.3-1 and 3.4.3-2 contains P-T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. Each P-T limit curve defines an acceptable region for normal operation. The Low Temperature Overpressure Protection (LTOP) System controls Reactor Coolant System (RCS) pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P-T limits. The

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reactor vessel is the limiting RCPB component for demonstrating such protection. Technical Specification 3.4.3 provides the allowable combinations for operational P-T during cooldown, shutdown, and heatup to keep from violating the 10 CFR Part 50, Appendix G requirements during the LTOP modes.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures. Reactor Coolant System pressure, therefore, is maintained low at low temperatures, and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shut down; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P-T limits by a significant amount could cause brittle cracking of the reactor vessel. Technical Specification 3.4.3 requires administrative control of RCS P-T during heatup and cooldown to prevent exceeding the P-T limits. Technical Specification 3.4.12 provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability. The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the curves in Technical Specification Figure 3.4.12-1.

REQUESTED CHANGES

Revise Unit 1 Technical Specification Figures 3.4.3-1, 3.4.3-2, and 3.4.12-1 as shown in the marked-up Technical Specification pages in Attachment (1).

SAFETY ANALYSIS

The current fluence-based 10 CFR Part 50, Appendix G, P-T limits for Calvert Cliffs Unit 1 was approved by the NRC on March 15, 1994 (Reference a). The fluence level corresponds to the pressurized thermal shock (PTS) screening criteria defined in 10 CFR 50.61 for the critical elements. Methods described in Regulatory Guide 1.99, Revision 2, are used to predict the embrittlement effect of neutron irradiation on reactor vessel materials. Regulatory Guide 1.99 defines embrittlement effect in terms of adjusted reference temperatures (ART), which is the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin term.

At the time of the current P-T limit approval, the PTS critical elements for the Unit 1 pressure vessel were the intermediate shell axial Welds 2-203 A, B, and C (Reference b). For these welds, the PTS screening criteria of 270°F was projected to be exceeded at fluence level 2.61×10^{19} n/cm^2, prior to the expiration of the current 40-year operating license. Since then a number of developments have taken place that significantly changed the previously projected PTS values. Among these developments was the NRC approval of use of Duke Power Company's McGuire Unit 1 reactor vessel surveillance results for reactor vessel fracture toughness analyses of Calvert Cliffs Unit 1 Weld Seams 2-203-A, B, and C (Reference c). Use of the McGuire Unit 1 reactor vessel surveillance results changed the chemistry factor for Welds 2-203 A, B, and C significantly; and as a result, these welds are no longer the PTS critical elements. The PTS critical elements for Calvert Cliffs Unit 1 are now the axial Welds 3-203-A, B, and C (References d, e, and f). The PTS projection for these welds will remain below screening criteria of 270°F for a period exceeding 20 years beyond the current 40-year operating license (References e and f). The proposed revision takes advantage of the material properties of the new PTS critical elements (Welds 3-203-A, B, and C) to validate the current P-T curves for a higher fluence value.

The current Part 50 Appendix G P-T limits for the reactor vessel beltline region were based on the ART values of 241.4°F and 181.0°F for the 1/4T and 3/4T locations, respectively. These ART values were predicted using Regulatory Guide 1.99, Revision 2, and the material properties for the then critical element, Welds 2-203-A, B, and C. As mentioned above, the fluence value used was 2.61×10^{19} n/cm², which corresponds to the screen criterion temperature limit of 270°F for these welds. Using the material properties of the new PTS critical elements, Welds 3-203-A, B, and C, a fluence value of 4.49×10^{19} n/cm² can now be justified for the Unit 1 Technical Specification P-T curves. As shown in Table 1 below, this combination results in a 1/4T ART of 229.3°F and 3/4T ART of 181°F, which are equal to or less than the values of 241.4°F and 181°F that were used to construct Unit 1 Technical Specification P-T curves [Figures 3.4.3.1, 3.4.3-2, and 3.4.12-1] (Reference a).

Table 1

Calvert Cliffs Unit 1 ART Calculation Using RG 1.99, Revision 2 Methodologies

P-T Limit Fluence Level in n/cm²	PTS Critical Element	Chemistry Factor Used for ART Calculation	Initial RT_{NDT} in deg-F	Margin in deg-F	1/4T Fluence in n/cm²	3/4T Fluence in n/cm²	1/4T ART in deg-F	3/4T ART in deg-F
Current Value 2.61×10^{19}	Old 2-203-A,B,C	210	-50	56	1.55×10^{19}	5.51×10^{18}	241.4	181.0
Proposed Value 4.49×10^{19}	New 3-203-A,B,C	174	-56	65.5	2.68×10^{19}	9.51×10^{18}	229.3	181.0

For Welds 3-203-A, B, and C, the PTS screening criteria of 270°F will be exceeded at a fluence level of 8.80×10^{19} n/cm². This fluence level will not be reached since the current projected end-of-life fluence values are 3.50×10^{19} n/cm² and 4.95×10^{19} n/cm², for 32 and 48 effective full power years (EFPYs), respectively (Reference f). With the proposed new fluence level of 4.49×10^{19} n/cm², the current P-T curves and LTOP setpoints will remain valid well beyond the current 40-year licensed period (32 EFPY).

DETERMINATION OF SIGNIFICANT HAZARDS

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

1. *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   In accordance with 10 CFR Part 50, Appendix G, the Calvert Cliffs pressure/temperature (P-T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials were developed using the methods of linear elastic fracture mechanics and the guidance found in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix G. The Calvert Cliffs (P-T) limits are based on fluence level. The fluence level corresponds to the pressurized thermal shock (PTS) screening criteria defined in 10 CFR 50.61 for the critical elements. Methods described in the Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, are used to predict the embrittlement effect of neutron irradiation on reactor vessel materials. Regulatory Guide 1.99 defines embrittlement effect in terms of adjusted reference temperatures (ART), which depends on the material property of the PTS critical element.

The proposed higher fluence level for the Technical Specification P-T limits was made possible by the identification of a new 10 CFR 50.61 critical element for fracture toughness requirements for protection against PTS events. The material properties of the new critical element resulted in an increase in fluence level from 2.61×10^{19} n/cm^2 to 4.49×10^{19} n/cm^2 for the ART values calculated using the material properties of the old PTS critical element. The P-T limits analysis remain well within the conservative acceptance limits of the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. Hence, with the new higher fluence level, the 10 CFR Part 50, Appendix G, requirement for adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests, for the reactor coolant pressure boundary materials, is maintained.

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

2. *Would not create the possibility of a new or different type of accident from any accident previously evaluated.*

The implementation of the proposed revision has no significant effect on either the configuration of the plant, or the manner in which it is operated.

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

3. *Would not involve a significant reduction in a margin of safety.*

As discussed above, the P-T limits analysis remain well within the conservative acceptance limits of the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. Hence, with the new higher fluence level, the 10 CFR Part 50, Appendix G, requirement for adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests, for the reactor coolant pressure boundary materials, is maintained.

Therefore, this proposed modification does not significantly reduce the margin of safety.

ENVIRONMENTAL ASSESSMENT

We have determined that operation with the proposed amendment will not result in any significant change in the types or significant increases in the amounts of any effluents that may be released offsite, and no significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

SAFETY COMMITTEE REVIEW

The Plant Operations and Safety Review Committee and the Offsite Safety Review Committee have reviewed this proposed amendment and concur that operation with the proposed amendment will not result in an undue risk to the health and safety of the public.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy Division, Baltimore Gas and Electric Company (BGE), and that I am duly authorized to execute and file this License Amendment Request on behalf of BGE. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other BGE employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Charles H. Cruse

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 18th day of November, 1999.

WITNESS my Hand and Notarial Seal: Denise A. Smith
Notary Public

My Commission Expires: 2/1/2002
Date

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Attachment: (1) Technical Specification Marked-up Pages

REFERENCES:

(a) Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated March 15, 1994, Issuance of Amendment for Calvert Cliffs Nuclear Power Plant, Unit No. 1(TAC No. M87690)

(b) Letter from Mr. G. C. Creel (BGE) to NRC Document Control Desk, dated December 13, 1991, Response to the 1991 Pressurized Thermal Shock Rule

(c) Letter from Mr. M. L. Boyle (NRC) to Mr. R. E. Denton (BGE), dated July 29, 1994, Request For Approval To Use Plant Specific Data For Reactor Vessel Fracture Toughness Analysis, Calvert Cliffs Nuclear Power Plant, Unit No. 1 (TAC No. M88316)

(d) Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated July 21, 1995, Request for Approval of Updated Values of Pressurized Thermal Shock (PTS) Reference Temperatures (RT_{PTS}) Values (10 CFR 50.61)

(e) Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated January 2, 1996, Updated Values for Pressurized Thermal Shock Reference Temperatures – Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M93230 and M93231)

(f) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated July 1, 1998, Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity at Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. MA0532 and MA0533)

cc:
R. S. Fleishman, Esquire
J. E. Silberg, Esquire
S. R. Peterson, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR
J. H. Walter, PSC

ATTACHMENT (1)

TECHNICAL SPECIFICATION MARKED-UP PAGES

3.4.3-3

3.4.3-4

3.4.12-6

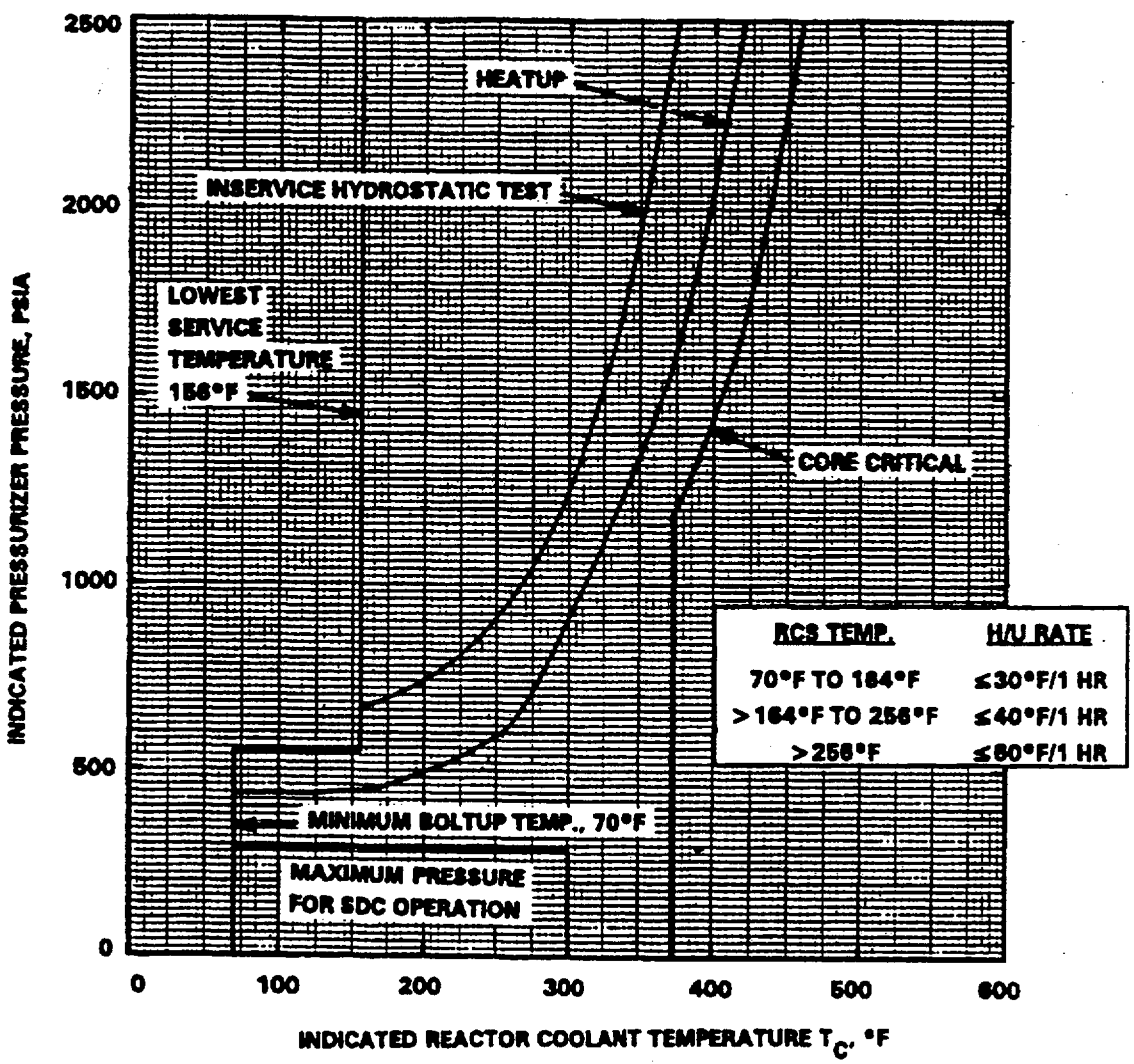

Figure 3.4.3-1
Calvert Cliffs Unit 1 Heatup Curve, for Fluence ≤ ~~2.61~~ 4.49 x10[19] n/cm[2]
Reactor Coolant System Pressure Temperature Limits

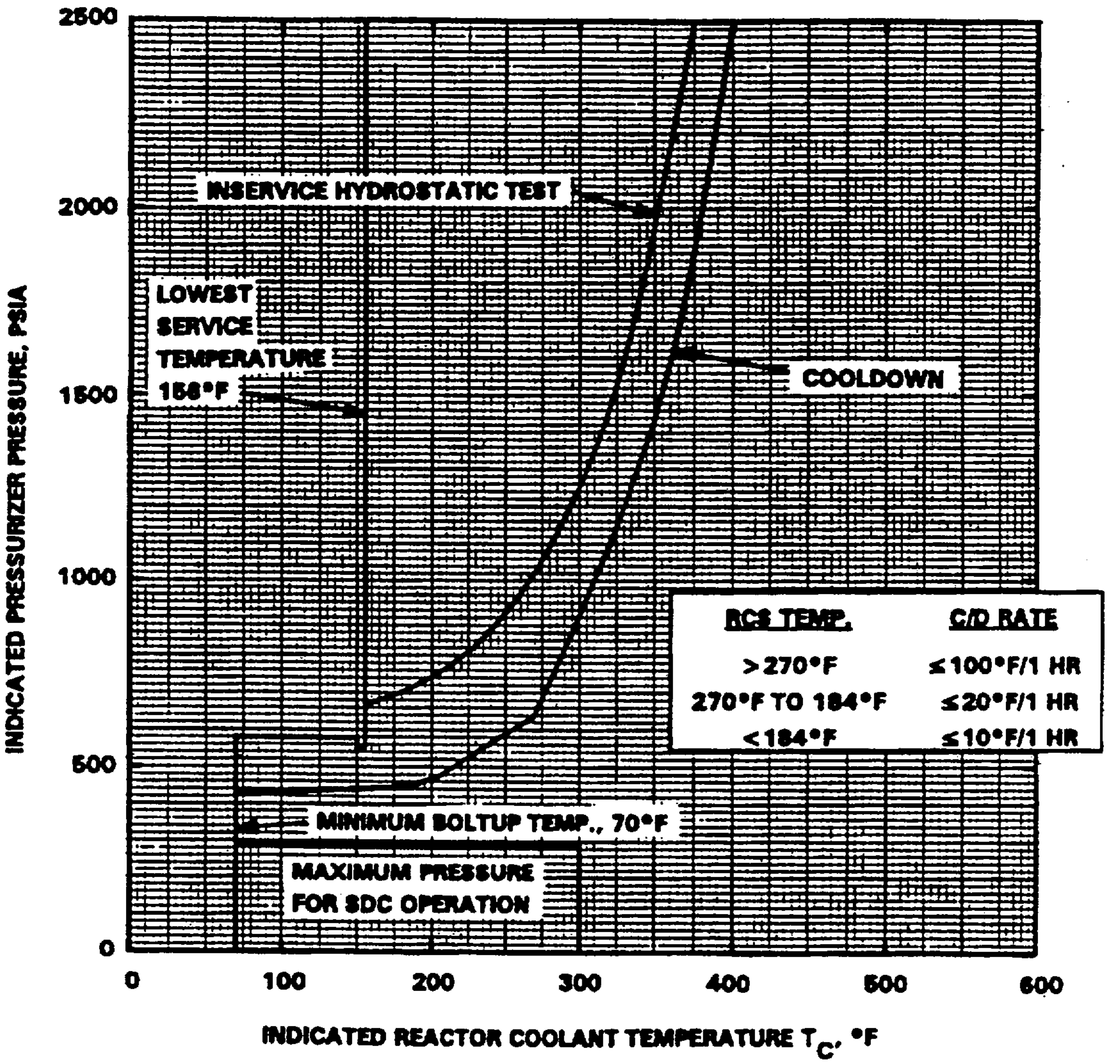

Figure 3.4.3-2
Calvert Cliffs Unit 1 Cooldown Curve, for Fluence ≤ ~~2.61~~ 4.49 x10^{19} n/cm^2
Reactor Coolant System Pressure Temperature Limits

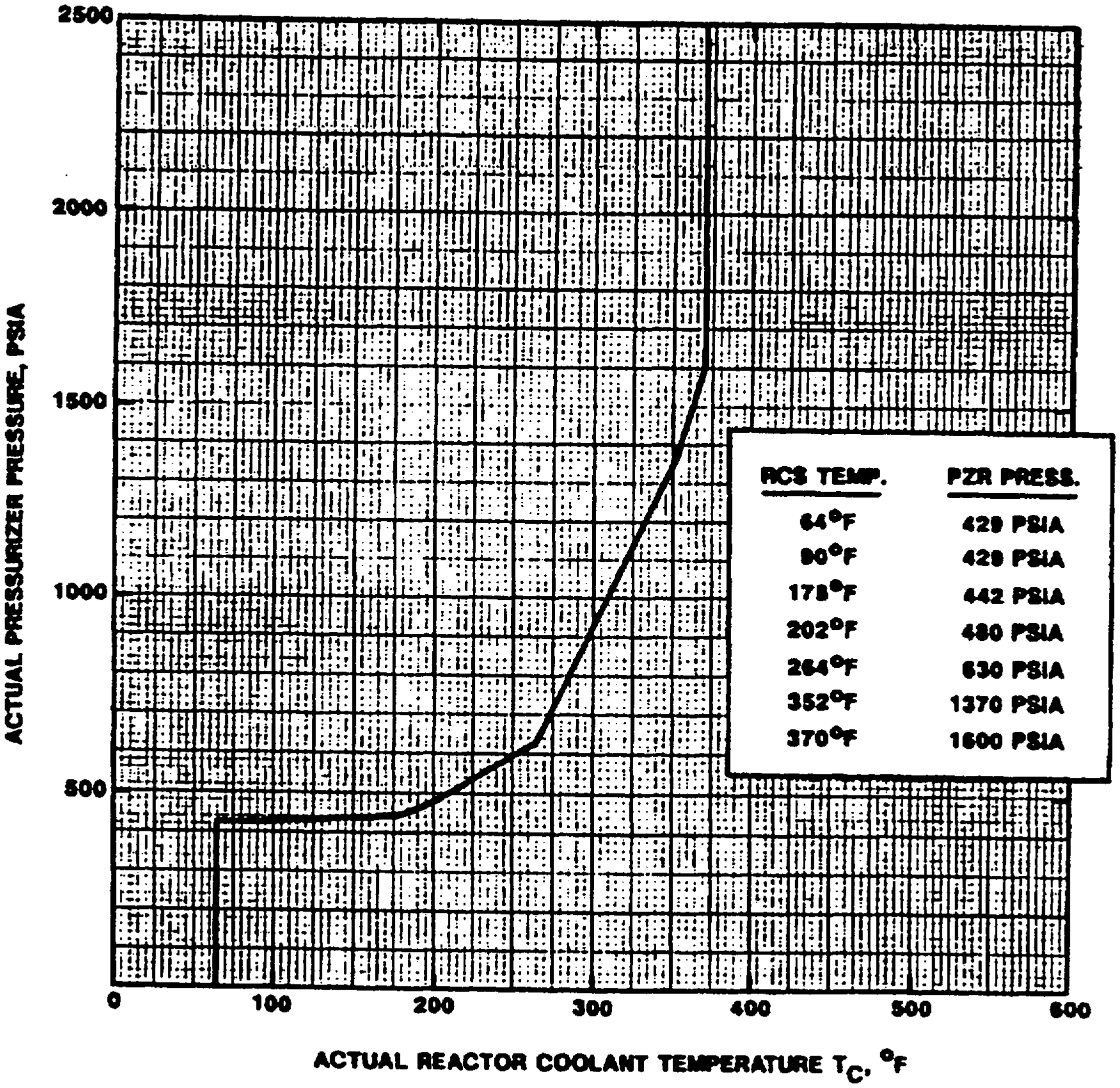

Figure 3.4-12-1
Calvert Cliffs Unit 1, for Fluence ≤ ~~2.61~~ 4.49 x10^{19} n/cm^2
Maximum PORV Opening Pressure vs Temperature