Union Electric

One Ameren Plaza
1901 Chouteau Avenue
PO Box 66149
St. Louis, MO 63166-6149
314.621.3222

December 3, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555-0001

Gentlemen: ULNRC-04158

AmerenUE

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
REVISION TO TECHNICAL SPECIFICATION 5.6.6 REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

References: 1) Letter dated April 2, 1998 from B.C. Westreich, NRC, to G.L. Randolph
2) Letter dated May 28, 1999 from J.N. Donohew, NRC, to G.L. Randolph

AmerenUE herewith transmits, an application for amendment to Facility Operating License No. NPF-30 for Callaway Plant, and the Callaway Plant Pressure and Temperature Limits Report (PTLR). The amendment application proposes revising Improved Technical Specification 5.6.6 to add references into the specification for the approval letter from the NRC, that approves the PTLR, and to WCAP-14040-NP-A, Revision 2. The PTLR is submitted per the guidance of Generic Letter 96-03 for NRC approval to allow the plant-specific application of the methodology used to calculate new plant heatup and cooldown, and cold overpressure protection curves.

Amendment No. 124, Reference 1, modified the Callaway Plant Technical Specifications. These modifications incorporated revised reactor coolant system (RCS) heatup and cooldown limit curves, and a revised Cold Overpressure Mitigation System (COMS) Power Operated Relief Valve (PORV) setpoint limit curve. These curves were calculated using the NRC-approved methods described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", and meet the requirements of 10 CFR 50 Appendices G and H.

Amendment No. 133, Reference 2, converted the Callaway Plant Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS). Subsequently, in discussions with the NRC staff, it was determined that the NRC had not specifically approved the methodology to allow changes to the limit curves without NRC approval as discussed in the guidance of Generic Letter

A001

PDR ADOCK

96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The guidance of Generic Letter 96-03 indicates that the licensee must (1) have a methodology approved by the NRC to reference in its Technical Specifications; (2) develop a report such as a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) or a similar document to contain the figures, values, parameters, and any explanation necessary; and (3) modify the applicable sections of the Technical Specifications accordingly.

The Enclosure to this letter provides a "DRAFT" PTLR. The enclosed PTLR is considered "DRAFT" based on incorporating a specific Reference to the document which provides NRC approval of the methodology. Attachments 1, 2, and 3 provide the Significant Hazards Evaluation, the Environmental Consideration, and the proposed changes to the Improved Technical Specifications. This change has been approved by the Callaway Onsite Review Committee and the Nuclear Safety Review Board.

Please note that we request that this amendment's issuance date be no later than April 1, 2000 and the date of implementation will coincide with the date of implementation of the ITS, which will be no later than April 30, 2000, as stated in the two license conditions. This implementation date is needed because of the revised COMS arming temperature of 275°F approved in Reference 3. The CTS contains an arming temperature of 368°F, which was lowered to 275°F via the response to NRC RAI Q3.4.11-3 on ITS Section 3.4. PTLR Figure 2.2-1 depicts the 275°F arming temperature, consistent with ITS 3.4.6, 3.4.7, 3.4.10, 3.4.11, and 3.4.12.

If you have any questions concerning this matter, please contact us.

Very truly yours,

Alan C Passwater

Alan C. Passwater
Manager, Corporate Nuclear Services

JMC/

Attachments:
1) Significant Hazards Evaluation
2) Environmental Consideration
3) Proposed Improved Technical Specification Changes

Enclosure

STATE OF MISSOURI)
)   S S
CITY OF ST. LOUIS)

Alan C. Passwater, of lawful age, being first duly sworn upon oath says that he is Manager, Corporate Nuclear Services for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Alan C Passwater
Alan C. Passwater
Manager, Corporate Nuclear Services

SUBSCRIBED and sworn to before me this 3Rd day of December, 1999.

Melissa L. Orr

MELISSA L. ORR
Notary Public - Notary Seal
STATE OF MISSOURI
City of St. Louis
My Commission Expires: June 23, 2003

cc: M. H. Fletcher
Professional Nuclear Consulting, Inc.
19041 Raines Drive
Derwood, MD 20855-2432

Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive
Suite 400
Arlington, TX 76011-8064

Senior Resident Inspector
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Jack Donohew (2)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
1 White Flint, North, Mail Stop OWFN 4D3
11555 Rockville Pike
Rockville, MD 20852-2738

Manager, Electric Department
Missouri Public Service Commission
P.O. Box 360
Jefferson City, MO 65102

Ron Kucera
Department of Natural Resources
P.O. Box 176
Jefferson City, MO 65102

Denny Buschbaum
TU Electric
P.O. Box 1002
Glen Rose, TX 76043

Pat Nugent
Pacific Gas & Electric
Regulatory Services
P.O. Box 56
Avila Beach, CA 93424

bcc: Phyllis Murdock/A160.761
/QA Record (CA-758)
E210.01
J. V. Laux
G. L. Randolph
R. J. Irwin
S. Gallagher
J. D. Blosser
A. C. Passwater
D. E. Shafer (2)
S. Wideman (WCNOC)
A. J. DiPerna, (Bechtel)
J. D. Schnack
NSRB (Melissa Orr)
J. M. Chapman
A140.0001(1203)

ATTACHMENT 1

SIGNIFICANT HAZARDS EVALUATION

SIGNIFICANT HAZARDS EVALUATION

INTRODUCTION

The Improved Technical Specifications (ITS) conversion (Amendment # 133) removed the Reactor Coolant System (RCS) heatup and cooldown limit curves and the Cold Overpressure Mitigation System (COMS) Power Operated Relief Valve (PORV) setpoint limit curve from the Current Technical Specifications (CTS) and placed them in the Pressure and Temperature Limits Report (PTLR). This amendment application submits the Callaway Plant PTLR along with proposed changes to ITS 5.6.6 to incorporate references to the following: 1) the NRC letter approving the Callaway Plant PTLR, and 2) WCAP-14040-NP-A, Revision 2, as documents approved by the NRC which contain analytical methods used to determine the RCS pressure and temperature and COMS PORV limits. The information contained in the PTLR covers all of the Westinghouse and NRC issues that were resolved in Amendment No. 124.

The CTS heatup and cooldown, and COMS PORV limit curves were reviewed and approved by the NRC staff in Amendment # 124, dated April 2, 1998. These limit curves that are valid for 20 effective full power years were calculated using the methods described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," which was approved by the NRC staff in its generic Safety Evaluation dated October 16, 1995. Once the NRC staff has reviewed and approved the plant-specific application of the PTLR methodology, future use of this methodology to calculate new heatup and cooldown, and COMS PORV limits can be made without prior NRC staff approval.

50.92 EVALUATION

The proposed change to the Improved Technical Specifications does not involve a significant hazards consideration as discussed below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change submits the PTLR, which contains the relocated CTS heatup and cooldown, and COMS PORV limits and the methodology used to calculate them, and the added references into ITS 5.6.6. The proposed change is administrative in nature since it is a movement of information from the CTS to a licensee controlled document, and has prior NRC staff approval. The PTLR contains the limit curves and the ITS requires more restrictive actions to be taken when the limiting conditions for operation are not met than is currently required by the CTS. The heatup and cooldown, and COMS PORV limits within the PTLR will be implemented and controlled per Callaway Plant programs and procedures and changes to the PTLR will be performed per requirements of 10 CFR 50.59 to ensure that change to these limits in the future will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does this change create the possibility of a new or different kind of accident from any accident previously evaluated?

   As stated earlier, the movement of the heatup and cooldown, and COMS PORV limits from the CTS to the PTLR has no influence or impact, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operations will be altered as a result of this proposed change. The proposed change is administrative in nature since it is a movement of requirements from the CTS to a licensee controlled document , the PTLR, and the change adds references into the ITS incorporating the licensee controlled document. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not affect the acceptance criteria for an analyzed event. The margin of safety presently provided by the CTS remains unchanged. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protective functions. Therefore, the proposed change is administrative in nature and does not impact the operation of Callaway Plant in a manner that involves a reduction in a margin of safety.

CONCLUSION

Based upon the preceding information, it has been determined that the proposed change to the Improved Technical Specifications does not involve a significant increase in the probability or consequences of any accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92 (c) and does not involve a significant hazards consideration.

ATTACHMENT 2

ENVIRONMENTAL CONSIDERATION

ENVIRONMENTAL CONSIDERATION

This amendment application revises the requirements of Improved Technical Specification (ITS) 5.6.6 to add a reference to the NRC's approval letter for the Callaway Plant Pressure and Temperature Limits Report and WCAP-14040-NP-A, Revision 2.

The proposed amendment does not involve changes with respect to the use of facility components located within the restricted area, as defined in 10 CFR 20. It is an administrative change that adds references into the ITS. AmerenUE has determined that the proposed amendment does not involve:

1) A significant hazards consideration, as discussed in Attachment 1 of this amendment application;

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite;

3) A significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION CHANGES

IMPROVED TECHNICAL SPECIFICATIONS

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).

2. WCAP-10216-P-A, REV. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

3. WCAP-10266-P-A, REV. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).

4. NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report' (TAC NO 77268)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

criticality,

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."

and COMS PORV

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

NRC letter dated [date], ["title of letter,"] and

1. ~~The analytical method used to determine the RCS pressure and temperature limits was developed in accordance with: 10 CFR 50, Appendix G and H, Regulatory Guide 1.99, Revision 2, NUREG-0800, Standard Review Plan, Section 5.3.2, Branch Technical Position MTEB 5-2, ASME B & PV Code, Section III, Appendix G, ASME B & PV Code, Section XI, Appendix A, WCAP-14040-NP-A, Section 2.2, and~~

2. ~~Cold overpressure mitigation system limits were developed in accordance with: NUREG-0800, Standard Review Plan, Section 5.2.2, Branch Technical Position RSB 5-2, 10 CFR 50, Appendix G and H, Regulatory Guide 1.99, Revision 2, Branch Technical Position MTEB 5-2, WCAP-14040-NP-A, Section 2.2.~~

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January, 1996.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used.

(continued)

ULNRC-04158

ENCLOSURE

PRESSURE AND TEMPERATURE LIMITS REPORT

CALLAWAY PLANT

PRESSURE AND TEMPERATURE LIMITS REPORT

Revision 0

PRESSURE AND TEMPERATURE LIMITS REPORT

Table of Contents

List of Figures

List of Tables

PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This PTLR for Callaway Plant has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6. The TS addressed in this report are listed below:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

2.0 Operating Limits

The parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. The limits were developed in accordance with the NRC-approved methodology specified in Specification 5.6.6 (Ref. 1). NRC approval of this methodology was received in Amendment No. [ ], (Ref. 3).

The methodology listed in WCAP-14040-NP-A, Revision 2 was used with one exception:

a) ASME Code Case N-514 was used.

The revised P/T Limit curves account for a requirement of 10 CFR 50, Appendix G, that the temperature of the closure head flange and vessel flange regions must be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure (3107 psig).

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup of 100°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2.1-1 and 2.1-2.

2.2 Cold Overpressure Mitigation System (COMS) Setpoints (LCO 3.4.12)

The pressurizer power-operated relief valves (PORVs) shall each have lift settings in accordance with Figure 2.2-1. The (COMS) arming temperature is 275°F. These lift setpoints have been developed using the NRC approved methodologies specified in Technical Specification 5.6.6.

2.2 (continued)

The maximum allowed PORV setpoint for COMS is derived by analysis which models the performance of the COMS assuming limiting mass and heat input transients with incorporation of 10% relaxation of the Appendix G limits in accordance with ASME Code Case N-514. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) pressure and temperature instrumentation uncertainties; (2) single failure of one PORV; and (3) effects of reactor coolant pump operation.

To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications place limitations on the number of safety injection pumps and centrifugal charging pumps that are capable of injecting, unisolating accumulators, and starting reactor coolant pumps during the appropriate COMS MODES. These limitations are outlined in TS LCO 3.4.6, LCO 3.4.7, and LCO 3.4.12.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2708-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 100.4°F
3/4T, 84.2 °F

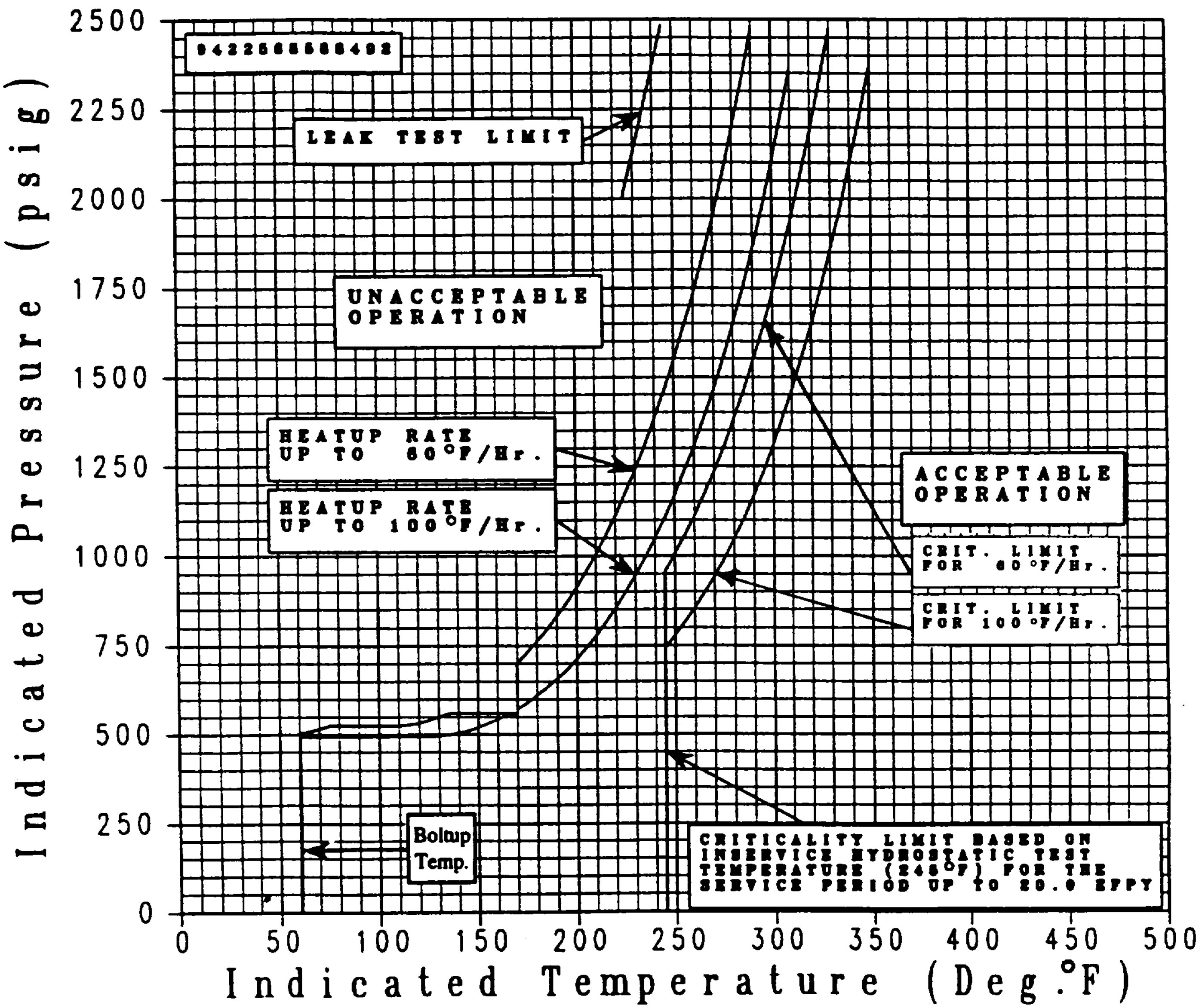

FIGURE 2.1-1 Callaway Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for the First 20 EFPY (With Margins for Instrumentation Errors)
Includes Vessel flange requirements of 170°F and 561 psig per 10CFR50, Appendix G.

TABLE 2.1-1 Callaway Plant Heatup Limits at 20 EFPY With Margins for Instrumentation Errors									
60°F/hr		60°F/hr Crit. Limit		100°F/hr		100°F/hr Crit. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	245	0	60	0	245	0	225	2000
60	500	245	544	60	494	245	546	245	2485
65	507	245	533	65	494	245	531		
70	516	245	527	70	494	245	519		
75	524	245	524	75	494	245	509		
80	524	245	524	80	494	245	502		
85	524	245	527	85	494	245	497		
90	524	245	533	90	494	245	494		
95	524	245	541	95	494	245	494		
100	524	245	551	100	494	245	495		
105	524	245	563	105	494	245	499		
110	524	245	577	110	494	245	504		
115	527	245	593	115	494	245	511		
120	533	245	611	120	494	245	520		
125	541	245	631	125	494	245	531		
130	551	245	653	130	495	245	543		
135	561	245	677	135	499	245	557		
140	561	245	703	140	504	245	574		
145	561	245	732	145	511	245	592		
150	561	245	763	150	520	245	612		
155	561	245	796	155	531	245	634		
160	561	245	832	160	543	245	658		
165	561	245	871	165	557	245	685		
170	561	245	912	170	561	245	714		
170	703	245	957	170	574	245	745		
175	732	250	1006	175	592	250	779		
180	763	255	1058	180	612	255	816		
185	796	260	1113	185	634	260	856		
190	832	265	1173	190	658	265	899		
195	871	270	1238	195	685	270	946		
200	912	275	1306	200	714	275	996		
205	957	280	1381	205	745	280	1049		
210	1006	285	1460	210	779	285	1107		
215	1058	290	1544	215	816	290	1169		
220	1113	295	1636	220	856	295	1236		
225	1173	300	1732	225	899	300	1307		
230	1238	305	1837	230	946	305	1384		
235	1306	310	1948	235	996	310	1467		
240	1381	315	2066	240	1049	315	1555		
245	1460	320	2193	245	1107	320	1649		
250	1544	325	2328	250	1169	325	1750		
255	1636	330	2472	255	1236	330	1858		
260	1732			260	1307	335	1973		
265	1837			265	1384	340	2096		
270	1948			270	1467	345	2227		
275	2066			275	1555	350	2367		
280	2193			280	1649				
285	2328			285	1750				
290	2472			290	1858				
				295	1973				
				300	2096				

TABLE 2.1-1 Callaway Plant Heatup Limits at 20 EFPY With Margins for Instrumentation Errors									
60°F/hr		60°F/hr Crit. Limit		100°F/hr		100°F/hr Crit. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
				305	2227				
				310	2367				

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2708-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 100.4°F
3/4T, 84.2 °F

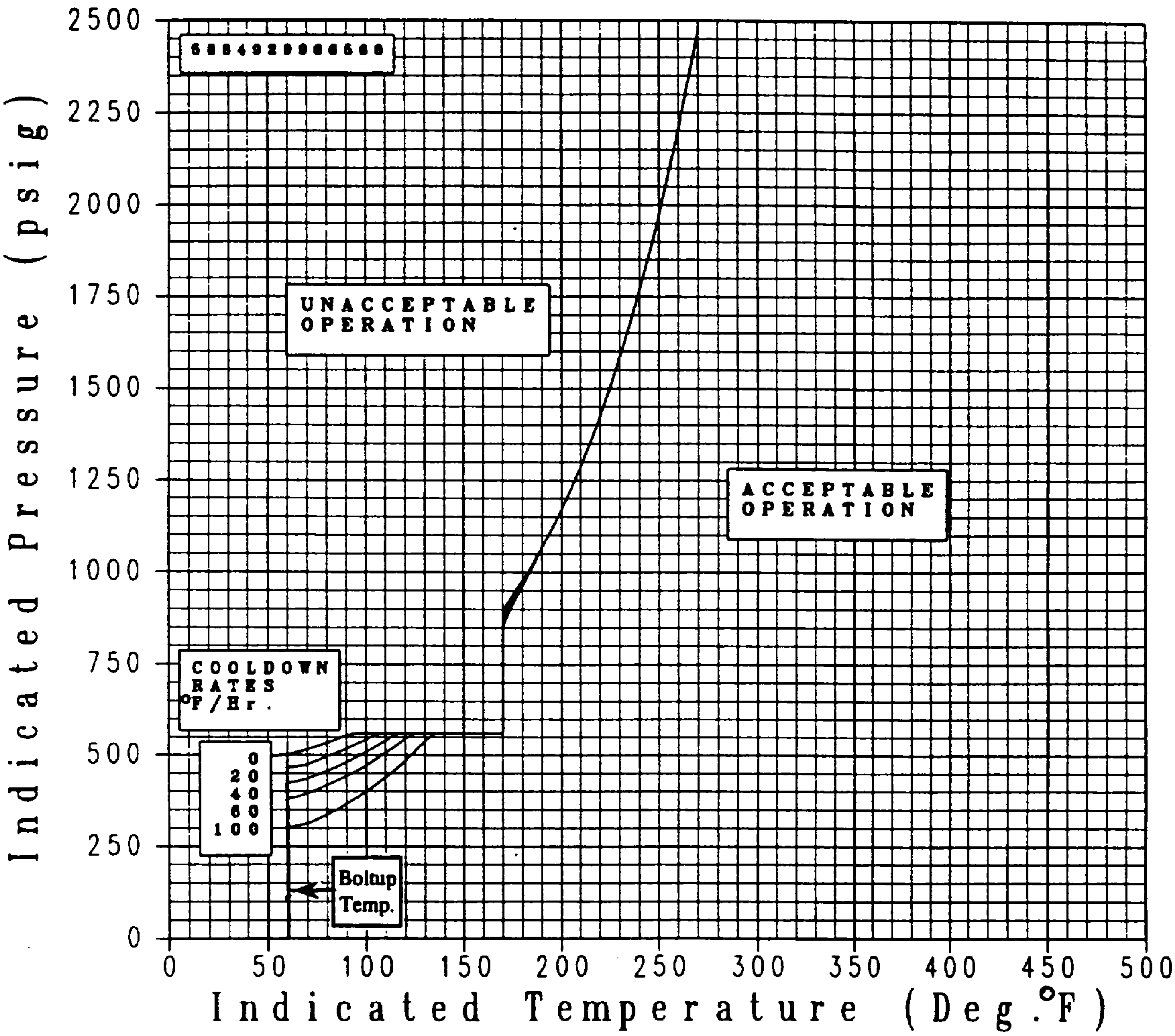

FIGURE 2.1-2 Callaway Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for the First 20 EFPY (With Margins for Instrumentation Errors)
Includes Vessel flange requirements of 170°F and 561 psig per 10CFR50, Appendix G.

TABLE 2.1-2 Callaway Plant Cooldown Limits at 20 EFPY With Margins for Instrumentation Errors									
Steady State		20°F/hr		40°F/hr		60°F/hr		100°F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0	60	0
60	500	60	468	60	427	60	386	60	302
65	507	65	468	65	427	65	386	65	302
70	516	70	476	70	436	70	396	70	313
75	525	75	486	75	446	75	406	75	325
80	534	80	496	80	457	80	418	80	338
85	544	85	506	85	468	85	430	85	351
90	555	90	518	90	481	90	443	90	366
95	561	95	531	95	494	95	457	95	382
100	561	100	544	100	508	100	472	100	400
105	561	105	559	105	524	105	489	105	419
110	561	110	561	110	540	110	506	110	439
115	561	115	561	115	558	115	526	115	461
120	561	120	561	120	561	120	546	120	485
125	561	125	561	125	561	125	561	125	511
130	561	130	561	130	561	130	561	130	539
135	561	135	561	135	561	135	561	135	561
140	561	140	561	140	561	140	561	140	561
145	561	145	561	145	561	145	561	145	561
150	561	150	561	150	561	150	561	150	561
155	561	155	561	155	561	155	561	155	561
160	561	160	561	160	561	160	561	160	561
165	561	165	561	165	561	165	561	165	561
170	561	170	561	170	561	170	561	170	561
170	899	170	885	170	873	170	864	170	856
175	937	175	925	175	917	175	911	175	910
180	977	180	969	180	963	180	961	180	969
185	1020	185	1015	185	1014	185	1016		
190	1066	190	1065						
195	1116								
200	1170								
205	1228								
210	1289								
215	1355								
220	1426								
225	1503								
230	1584								
235	1672								
240	1766								
245	1866								
250	1973								
255	2088								
260	2211								
265	2342								
270	2482								

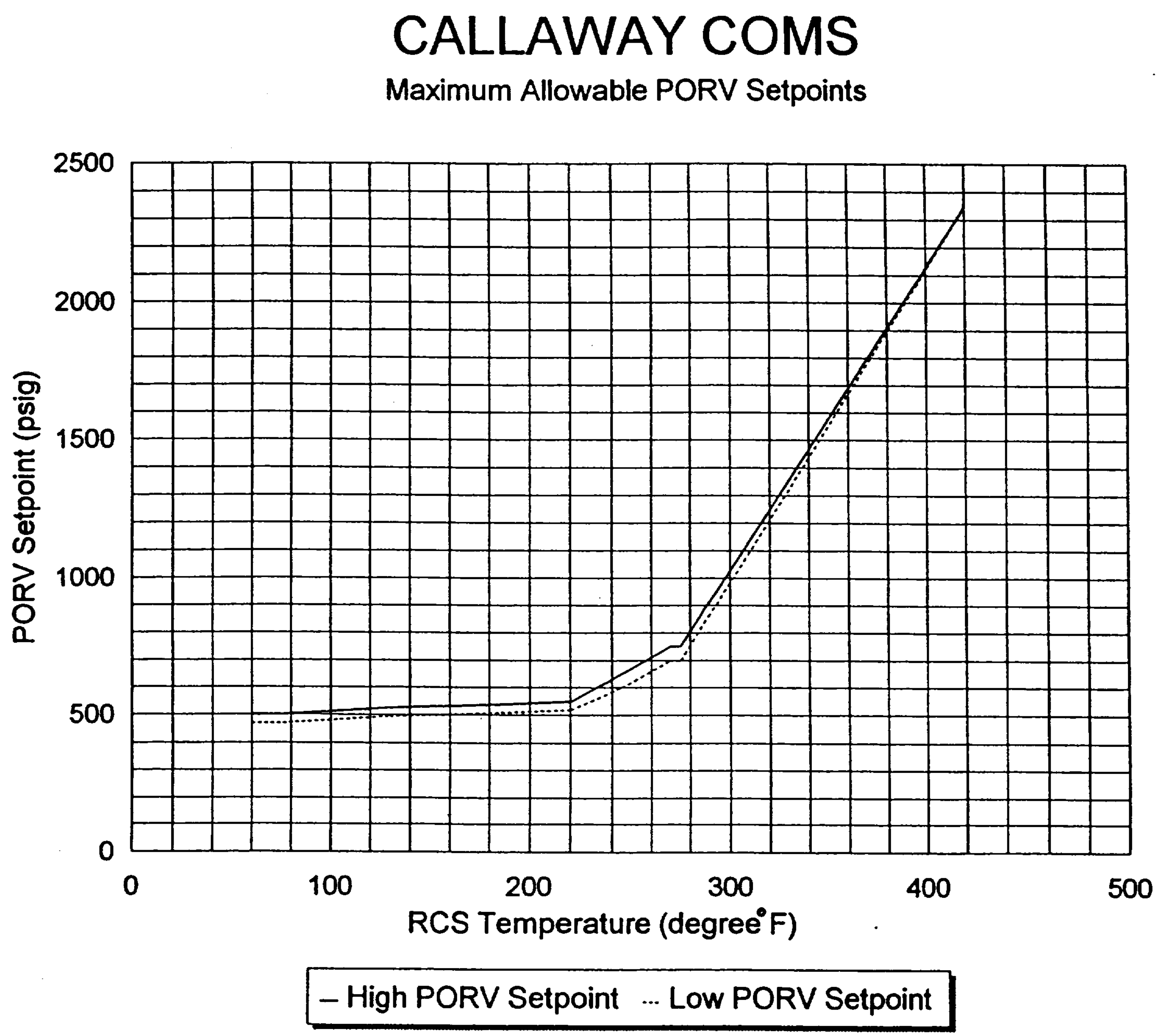

FIGURE 2.2-1 Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System

TABLE 2.2-1
CALLAWAY PLANT COMS MAXIMUM ALLOWABLE PORV SETPOINTS AT 20 EFPY

	Maximum Allowable Function Generator Setpoints (Breakpoints)		
Breakpoint Number	**Temperature – RCS (°F)**	**High Setpoint (psig)**	**Low Setpoint (psig)**
1	60	501	471
2	76	501	471
3	130	525	495
4	170	535	505
5	220	550	520
6	245	650	600
7	270	750	700
8	275	750	700
9	420	2350	2350

NOTE: Setpoints assume that 0 reactor coolant pumps are running for T<100°F and that 4 reactor coolant pumps are in operation for T ≥100°F.

3.0 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance program is in compliance with Appendix H to 10 CFR 50, entitled "Reactor Vessel Material Surveillance Program Requirements" and Section 5.3 of the Callaway Final Safety Analysis Report. The surveillance capsule withdrawal schedule is presented in FSAR Table 5.3-10. The surveillance capsule reports are as follows:

1. WCAP-11374, Revision 1, June 1987, "Analysis of Capsule U from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program."
2. WCAP-12946, June 1991, "Analysis of Capsule Y from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program."
3. WCAP-14895, July 1997, "Analysis of Capsule V from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program."

4.0 Reactor Vessel Surveillance Data Credibility

Regulatory Guide 1.99, Revision 2, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date three surveillance capsules have been removed and analyzed from the Callaway Plant reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with the discussion of Regulatory Guide 1.99, Revision 2, there are five requirements that must be met for the surveillance data to be judged credible.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Callaway Plant reactor vessel surveillance data and determine if the Callaway Plant surveillance data is credible.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," as follows:

> "the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

The Callaway Plant reactor vessel consists of the following beltline region materials:

- Intermediate shell plate R2707-1,
- Intermediate shell plate R2707-2,
- Intermediate shell plate R2707-3,
- Lower shell plate R2708-1,
- Lower shell plate R2708-2,
- Lower shell plate R2708-3, and
- Intermediate shell longitudinal weld seams, lower shell longitudinal weld seams, and a intermediate to lower shell circumferential weld seam All vessel beltline weld seams were fabricated with weld wire heat number 90077. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate and lower shell longitudinal weld seams were fabricated with Flux Type 0091 Lot Number 0842.

The Callaway Plant surveillance program utilizes longitudinal and transverse test specimens from lower shell plate R2708-1. The surveillance weld metal was fabricated with weld wire heat number 90077, Flux Type 124 Lot Number 1061.

At the time when the surveillance program was selected it was believed that copper and phosphorus were the elements most important to embrittlement of reactor vessel steels. Since all plate materials had approximately the same content of copper and phosphorus, lower shell plate R2708-1 was chosen for the surveillance program since it had the highest initial RT_{NDT} and the lowest initial upper shelf energy of the plate material. In addition, the current pressurized thermal shock (PTS) evaluation shows that if surveillance data is not used, lower shell plate R2708-1 is the plate that is predicted to have the highest embrittlement rate.

Per Regulatory Guide 1.99, Revision 2, "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. Since the surveillance weld metal was made with the same weld wire heat as all of the vessel beltline weld seams, it is representative of the limiting beltline weld metal.

Based on the above discussion, the Callaway Plant surveillance materials are those judged most likely to be controlled with regard to radiation embrittlement and the Callaway Plant surveillance program meets this criteria.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper shelf energy unambiguously.

Plots of Charpy energy versus temperature for the unirradiated and irradiated condition are presented in WCAP-14895, July 1997, "Analysis of Capsule V from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program."

Based on engineering judgement, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper shelf energy of the Callaway Plant surveillance materials unambiguously. Hence, theCallaway Plant surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of the ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the plate.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2.

Table 4.0-1
Callaway Plant Surveillance Capsule Data

Material	Capsule	F[1]	FF[2] (x)	ΔRT_{NDT} (y)	FFxΔRT_{NDT} (xy)	FF^2 (x^2)
Lower Shell Plate R2708-1 (Longitudinal)	U	0.3342	0.698	-7.33	-5.12	0.487
	Y	1.237	1.059	25.15	26.63	1.121
	V	2.359	1.232	16.45	20.27	1.518
Lower Shell Plate R2708-1 (Transverse)	U	0.3342	0.698	25.86	18.05	0.487
	Y	1.237	1.059	46.39	49.13	1.121
	V	2.359	1.232	44.82	55.22	1.518
		$\sum^{n}_{i=1}$	5.978	151.34	164.18	6.252
Weld Metal[3]	U	0.3342	0.698	68.53	47.83	0.487
	Y	1.237	1.059	36.92	39.10	1.121
	V	2.359	1.232	48.21	59.39	1.518
		$\sum^{n}_{i=1}$	2.989	153.66	146.32	3.126

(1) F = Calculated Fluence (10^{19} n/cm^2, E > 1.0 MeV). These values were re-evaluated as part of the capsule V analysis. (See Section 6 of WCAP-14895.)

(2) FF = Fluence Factor = $F^{(0.28 - 0.1 * \log F)}$

(3) ΔRT_{NDT} values do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values.

Per the 27th Edition of the CRC Standard Mathematical Tables (page 497), for a straight line fit by the method of least squares, the values b_o and b_1 are obtained by solving the normal equations

$$nb_o + b_1\sum x_i = \sum y_i \quad \text{and}$$

$$b_o\sum x_i + b_1\sum x_i^2 = \sum x_i y_i$$

These equations can be re-written as follows (b_o = a and b_1 = b):

$$\sum_{i=1}^{n} y_i = an + b\sum_{i=1}^{n} x_i \quad \text{and}$$

$$\sum_{i=1}^{n} x_i y_i = a\sum_{i=1}^{n} x_i + b\sum_{i=1}^{n} x_i^2$$

Lower shell plate R2708-1:

Based on the data provided in Table 4.0-1 these equations become:

$$151.34 = 3a + 5.978b \quad \text{and}$$

$$164.18 = 5.978a + 6.252b$$

Thus, b = 24.273 and a = 2.078, and the equation of the straight line which provides the best fit in the sense of least squares is:

$$Y' = 24.273\,(X) + 2.078$$

The scatter in predicting a value Y corresponding to a given X value is:

$$e = Y - Y'$$

Table 4.0-2 Callaway Plant Lower Shell Plate R2708-1			
FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
0.698	-7.33	19.0	-26.3
1.059	25.15	27.8	-2.65
1.232	16.45	32.0	-15.6
0.698	25.86	19.0	6.86
1.059	46.39	27.8	18.6
1.232	44.82	32.0	12.8

The scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 (Table 4.0-2) is not less than 17°F. However, the scatter is less than twice this value. Hence, this criterion is met for the surveillance data of lower shell plate R2708-1.

Weld Metal:

Based on the data provided in Table 4.0-1 the equations become:

$$153.66 = 3a + 2.989b \quad \text{and}$$

$$146.32 = 2.989a + 3.126b$$

Thus, b = -45.8 and a = 96.852, and the equation of the straight line which provides the best fit in the sense of the least squares is:

$$Y' = -45.8\ (X) + 96.852$$

The scatter in predicting a value of Y corresponding to a given X value is:

$$e = Y - Y'$$

Table 4.0-3 Callaway Plant Surveillance Weld Metal

FF	ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT} (°F)	Scatter of ΔRT_{NDT} (°F)
0.698	68.53	64.9	3.6
1.059	36.92	48.3	-11.4
1.232	48.21	40.4	7.8

The scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 (Table 4.0-3) is less than 28°F. Therefore, this criterion is met for the surveillance weld material data.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the neutron pads and thevessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vesselbeltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material.

The Callaway Plant surveillance program does not contain correlation monitor material. Therefore, this criterion is not applicable to the Callaway Plant surveillance program.

Based on the preceding positive responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B, the Callaway Plant surveillance data is credible.

5.0 Supplemental Data Tables

Table 5.0-1 Comparison of Callaway Plant Surveillance Material 30-ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions.

Table 5.0-2 Calculation of Chemistry Factors Using Surveillance Capsule Data.

Table 5.0-3 Provides the unirradiated reactor vessel toughness data. The bolt-up temperature is also included in this Table.

Table 5.0-4 Provides a summary of the pressure vessel neutron fluence values at 20 EFPY used for the calculation of ART values.

Table 5.0-5 Provides a summary of the adjusted reference temperature (ARTs) for reactor vessel beltline materials at the ¼-T and ¾-T locations for 20 EFPY.

Table 5.0-6 Shows the calculation of the ART at 20 EFPY for the limiting reactor vessel material (lower shell plate R-2708-3).

Table 5.0-7 Provides RT_{PTS} values for 35 EFPY.

6.0 References

1. Technical Specification 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

2. NRC letter dated [ ], [title of letter]

3. License Amendment No. [ ], dated [ ], from [ ] to [ ].

4. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January, 1996.

Table 5.0-1
Comparison of Callaway Plant Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Materials	Capsule	Fluence (n/cm2, E>1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%) [c]
Lower Shell Plate R2708-1 (Longitudinal)	U	3.342×10^{18}	30.7	-7.33	14.5	0
	Y	1.237×10^{19}	46.6	25.15	20	6
	V	2.359×10^{19}	54.2	16.45	23	0
Lower Shell Plate R2708-1 (Transverse)	U	3.342×10^{18}	30.7	25.86	14.5	11
	Y	1.237×10^{19}	46.6	46.39	20	13
	V	2.359×10^{19}	54.2	44.82	23	3
Weld Metal	U	3.342×10^{18}	22.2	68.53	14.5	11
	Y	1.237×10^{19}	33.7	36.92	20	14
	V	2.359×10^{19}	39.2	48.21	23	8
HAZ Metal	U	3.342×10^{18}	--	65.93	--	0
	Y	1.237×10^{19}	--	56.38	--	14
	V	2.359×10^{19}	--	56.1	--	0

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1.

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

Table 5.0-2
Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	$F^{(1)}$	$FF^{(2)}$	$\Delta RT_{NDT}{}^{(3)}$	$FF \times \Delta RT_{NDT}$	FF^2
Lower Shell Plate R2708-1 (Longitudinal)	U	0.3342	0.698	-7.33	-5.12	0.487
	Y	1.237	1.059	25.15	26.63	1.121
	V	2.359	1.232	16.45	20.27	1.518
Lower Shell Plate R2708-1 (Transverse)	U	0.3342	0.698	25.86	18.05	0.487
	Y	1.237	1.059	46.39	49.13	1.121
	V	2.359	1.232	44.82	55.22	1.518
				SUM	164.18°F	6.252
	$CF_{R2708\text{-}1} = \Sigma(FF \times \Delta RT_{NDT}) \div \Sigma(FF^2) = (164.18°F) \div (6.252) = 26.3°F$					
Vessel Weld Metal Based on Surveillance Program Weld Metal Results(4)	U	0.3342	0.698	64.01	44.68	0.487
	Y	1.237	1.059	34.48	36.51	1.121
	V	2.359	1.232	45.03	55.48	1.518
				SUM	136.67°F	3.126
	$CF_{weld} = \Sigma(FF \times \Delta RT_{NDT}) \div \Sigma(FF^2) = (136.67°F) \div (3.126) = 43.7°F$					

(1) F = Calculated Fluence (10^{19} n/cm^2, E > 1.0 MeV). All updated fluence values were taken from the Capsule V analysis (Table 6-12 of WCAP 14895).

(2) FF = Fluence Factor = $F^{(0.28 - 0.1 * \log F)}$

(3) ΔRT_{NDT} values were obtained from the Capsule V analysis.

(4) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio of 0.934 ($CF_{vessel\ weld} \div CF_{surv.\ weld}$ = 29.7 ÷ 31.8 = 0.934).

TABLE 5.0-3			
Reactor Vessel Beltline Material Unirradiated Toughness Properties			
Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}[a]**
Closure Head Flange R2704-1	--	--	30°F[c]
Vessel Flange R2701-1	--	--	40°F[c]
Intermediate Shell Plate R2707-1	0.05	0.58	40°F
Intermediate Shell Plate R2707-2	0.06	0.61	10°F
Intermediate Shell Plate R2707-3	0.06	0.62	-10°F
Lower Shell Plate R2708-1	0.07	0.58	50°F
Lower Shell Plate R2708-2	0.06	0.57	10°F
Lower Shell Plate R2708-3	0.08	0.62	20°F
Intermediate and Lower Shell Longitudinal Weld Seams[b]	0.04	0.06	-60°F
Intermediate to Lower Shell Circumferential Weld Seam [b]	0.04	0.06	-60°F

(a) The initial RT_{NDT} values for the plates and welds are based on measured data (WCAP-12948).

(b) All vessel beltline weld seams were fabricated with weld wire heat number 90077. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate and lower shell longitudinal weld seams were fabricated with Flux Type 0091 Lot 0842. The surveillance weld metal was fabricated with weld wire heat number 90077, Flux Type 124 Lot number 1061. Per Regulatory Guide 1.99, Revision 2, "weight-percent copper " and "weight-percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. The surveillance weld metal was made with the same weld wire heat as all of the vessel beltline weld seams and is therefore representative of all of the beltline weld seams.

(c) These values are used for considering flange requirements for the heatup/cooldown curves. Per the methodology given in WCAP-14040-NP-A (Ref. 4), the minimum boltup temperature is 60°F.

TABLE 5.0-4
Fluence (10^{19} n/cm^2, E > 1.0 MeV) on the Pressure Vessel Clad/Base Metal Interface for Callaway Plant

EFPY	0°	15°	30°	45°
9.85	0.351	0.514	0.601	0.619
17	0.584	0.857	1.01	1.03
20	0.682	(a)	1.182	1.204
32	1.07	1.57	1.87	1.90

NOTES: (a) Fluence values for 15° are not needed for this evaluation.

TABLE 5.0-5
Summary of Adjusted Reference Temperature (ART) Values at the ¼-T and ¾-T Locations for 20 EFPY

Material	20 EFPY ART[a]	
	¼-T ART (°F)	¾-T ART (°F)
Intermediate Shell Plate R2707-1	96.4	79.0
Intermediate Shell Plate R2707-2	77.4	56.6
Intermediate Shell Plate R2707-3	57.4	36.6
Lower Shell Plate R2708-1	124.0	105.4
Using Surveillance Capsule Data	90.9	83.2
Lower Shell Plate R2708-2	77.4	56.6
Lower Shell Plate R2708-3	100.4[b]	84.2[b]
Intermediate & Lower Shell Longitudinal Weld Seams 101-124A & 101-142A (90° Azimuth)	-15.4	-30.8
Using Surveillance Capsule Data	0.8	-17.2
Intermediate & Lower Shell Longitudinal weld Seams 101-124B&C and 101-142B&C (210° & 330° Azimuth)	-6.6	-23.2
Using Surveillance Capsule Data	7.3	-5.8
Intermediate to Lower Shell Circumferential Weld Seam 101-171	-6.0	-22.6
Using Surveillance Capsule Data	7.8	-5.0

(a) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(b) These ART values are used to generate the heatup and cooldown curves.

Note: When two or more credible surveillance data sets become available, the data sets may be used to determine ART values as described in Regulatory Guide 1.99, Revision 2, Position 2.1. If the ART values based on surveillance capsule data are larger than those calculated per Regulatory Guide 1.99, Revision 2, Position 1.1, the surveillance data must be used. If the surveillance capsule data gives lower values, either may be used.

TABLE 5.0-6

Calculation of Adjusted Reference Temperature Values at 20 EFPY for the Limiting Callaway Plant Reactor Vessel Material (Lower Shell Plate R2708-3)

Parameter	ART Value	
Location	**¼-T**	**¾-T**
Chemistry Factor, CF (°F)	51.0	51.0
Fluence, f (10^{19} n/cm^2)[(a)]	0.7174	0.2547
Fluence Factor, FF[(b)]	0.91	0.63
ΔRT_{NDT} = CF x FF, (°F)	46.4	32.1
Initial RT_{NDT}, I (°F)	20	20
Margin, M (°F)[(c)]	34	32.1
ART = I + (CF x FF) + M (°F) per Regulatory Guide 1.99, Rev. 2	100.4	84.2

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 1.204 at 20 EFPY. The Callaway Plant reactor vessel wall thickness is 8.63 inches at the beltline region.

(b) Fluence Factor (FF) per Regulatory Guide 1.99, Revision 2, is defined as $FF = f^{(0.28 - 0.10 \log f)}$.

(c) Margin is calculated as $M = 2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$. The standard deviation for the initial RT_{NDT} margin term σ_I, is 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT} term σ_Δ, is 17°F for the plate, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

TABLE 5.0-7 RT_{PTS} Calculations for Callaway Plant Beltline Region Materials at 35 EFPY[(d)]							
Material	Fluence (10^{19} n/cm^2, E >1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[(c)] (°F)	Margin (°F)	$RT_{NDT(U)}$[(a)] (°F)	RT_{PTS}[(b)] (°F)
Intermediate Shell Plate R2707-1	2.074	1.20	31.0	37.2	34.0	40	111
Intermediate Shell Plate R2707-2	2.074	1.20	37.0	44.4	34.0	10	88
Intermediate Shell Plate R2707-3	2.074	1.20	37.0	44.4	34.0	-10	68
Lower Shell Plate R2708-1	2.074	1.20	44.0	52.8	34.0	50	137
Using S/C Data	2.074	1.20	26.3	31.6	17.0	50	99
Lower Shell Plate R2708-2	2.074	1.20	37.0	44.4	34.0	10	88
Lower Shell Plate R2708-3	2.074	1.20	51.0	61.2	34.0	20	115
Inter. and Lower Shell Long. Weld Seams101-124A & 101-142A (90° Azimuth)	1.167	1.04	29.7	30.9	30.9	-60	2
Using S/C Data	1.167	1.04	43.7	45.4	28.0	-60	13
Inter. and Lower Shell Long. Weld Seams101-124B&C & 101-142B&C (210° & 330° Azimuth)	2.042	1.19	29.7	35.3	35.3	-60	11
Using S/C Data	2.042	1.19	43.7	52.0	28.0	-60	20
Intermediate to Lower Shell Circumferential Weld Seam 101-171	2.074	1.20	29.7	35.6	35.6	-60	11
Using S/C Data	2.074	1.20	43.7	52.4	28.0	-60	20

(a) Initial RT_{NDT} values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \text{Margin} + \Delta RT_{PTS}$

(c) $\Delta RT_{PTS} = CF * FF$

(d) Projected no. of EPFY at the EOL.