Paula M. Marino
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ENCLOSURE CONTAINS INFORMATION NOT FOR PUBLIC DISCLOSURE

SOUTHERN COMPANY

March 14, 2013

Docket Nos.: 50-348
50-364

NL-13-0545

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

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Concerning a Revision to the Technical Specifications Associated with the Low Temperature Overpressure Protection System and the Pressure-Temperature Limits Report

Ladies and Gentlemen:

By letter dated August 15, 2012, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for the Joseph M. Farley Nuclear Plant (FNP). Based on the implementation of new 54 effective power years (EFPY) pressure and temperature limit curves, corresponding changes to the FNP Technical Specifications (TS) associated with the lower temperature overpressure protection system and other limits were proposed. The Nuclear Regulatory Commission (NRC) sent SNC a Request for Additional Information (RAI) by letter dated February 1, 2013. Attachment 1 contains the supporting affidavit signed by Electric Power Research Institute (EPRI), the owner of the proprietary information used in the SNC response. This affidavit sets forth the basis on which the information in Enclosure 1 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to EPRI, be withheld from public disclosure in accordance with 2.390 of the Commission's regulations. Enclosures 1 and 2 contain the proprietary and non-proprietary versions, respectively, of the SNC response to the RAI. The proprietary information contained in Enclosure 1 is contained within brackets with a superscript "1" to the left of the brackets. The proprietary information has been deleted in Enclosure 2, so that only the brackets and superscript "1" remain.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

A001
NRR

Ms. P. M. Marino states she is Vice President - Engineering of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

Paula M. Marino

P. M. Marino
Vice President - Engineering

PMM/RMJ/lac

Sworn to and subscribed before me this 14th day of March, 2013.

Charlotte A. Graham
Notary Public

My commission expires: 11/30/15

Attachment: 1. EPRI letter to NRC, "Request for Withholding of the following Proprietary Information Included in:," dated March 12, 2013

Enclosures: 1. Response to Request for Additional Information (Proprietary)
2. Response to Request for Additional Information (Non-Proprietary)

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. T. A. Lynch, Vice President – FNP
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
Mr. C. R. Pierce, Regulatory Affairs Director
RTYPE: CFA04.054

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Ms. E. A. Brown, NRR Project Manager - FNP
Mr. P. K. Niebaum, Senior Resident - FNP
Mr. J. R. Sowa, Senior Resident - FNP

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Concerning a Revision to the Technical Specifications Associated with the Low Temperature Overpressure Protection System and the Pressure-Temperature Limits Report

Attachment 1

EPRI letter to NRC, "Request for Withholding of the following Proprietary Information Included in:," dated March 12, 2013

March 12, 2013

Kurt Edsinger
Director, PWR & BWR Materials

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the following Proprietary Information Included in:

Enclosure 1 to Southern Nuclear Operating Company letter NL-13-0545, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Concerning a Revision to the Technical Specifications Associated with the Low Temperature Overpressure Protection System and the Pressure-Temperature Limits Report," to the Nuclear Regulatory Commission

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the attached report. Proprietary and non-proprietary versions of the Response and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Proprietary Information in confidence for informational purposes regarding a submittal to the NRC by Southern Nuclear. The Proprietary Information is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Proprietary Information provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (650) 855-2271. Questions on the content of the Proprietary Information should be directed Andy McGehee of EPRI at (704) 502-6440.

Sincerely,

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In:

Enclosure 1 to Southern Nuclear Operating Company letter NL-13-0545, "Joseph M. Farley Nuclear Plant Response to Request for Additional Information Concerning a Revision to the Technical Specifications Associated with the Low Temperature Overpressure Protection System and the Pressure-Temperature Limits Report," to the Nuclear Regulatory Commission

I, Kurt Edsinger, being duly sworn, depose and state as follows:

I am the Director, of the Nuclear PWR & BWR Materials Program at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, California ("EPRI") and I have been specifically delegated responsibility for the above-listed Response that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI requests that the Proprietary Information be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Proprietary Information is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Proprietary Information do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Proprietary Information. The Proprietary Information is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Proprietary Information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information. EPRI made a substantial economic investment to develop the Proprietary Information and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Proprietary Information. If the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

c. EPRI's classification of the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

"'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:

(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

d. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

e. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

The EPRI trade secret information in Enclosure 1 to NL-13-0545 that is sought to be withheld under 10 CFR 2.390 (a)(4) is appropriately marked with a bracket and a superscript "1" to the left of the bracket.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3420 Hillview Avenue, Palo Alto, CA. 94304 being the premises and place of business of Electric Power Research Institute, Inc.

Date: 3/12/2013

Kurt Edsinger

(State of California)
(County of Santa Clara)

(See attached)

Subscribed and sworn to (or affirmed) before me on this ____day of ____________, 20__, by ______________________________, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature ______________________________ (Seal)

My Commission Expires ____day of ____________, 20__,

CALIFORNIA JURAT WITH AFFIANT STATEMENT

☒ See Attached Document (Notary to cross out lines 1–6 below)
☐ See Statement Below (Lines 1–5 to be completed only by document signer[s], *not* Notary)

1 ____
2 ____
3 ____
4 ____
5 ____
6 ______________________ ______________________

Signature of Document Signer No. 1 | Signature of Document Signer No. 2 (if any)

State of California
County of Santa Clara

Subscribed and sworn to (or affirmed) before me on this 12th (Date) day of March (Month), 20 13 (Year), by
(1) Kurt Edsinger (Name of Signer),
proved to me on the basis of satisfactory evidence to be the person who appeared before me (.) (,)
(and
(2) ______________________ (Name of Signer),
proved to me on the basis of satisfactory evidence to be the person who appeared before me.)

KATHY SYLER
Commission # 1937043
Notary Public - California
Santa Clara County
My Comm. Expires May 19, 2015

Place Notary Seal Above

Signature Kathy Syler
Signature of Notary Public

OPTIONAL

Though the information below is not required by law, it may prove valuable to persons relying on the document and could prevent fraudulent removal and reattachment of this form to another document.

Further Description of Any Attached Document

Title or Type of Document: AFF. REQ WITH-ENC #1 SNOC LTR

Document Date: 3/12/13 Number of Pages: 2

Signer(s) Other Than Named Above: ______________________

RIGHT THUMBPRINT OF SIGNER #1
Top of thumb here

RIGHT THUMBPRINT OF SIGNER #2
Top of thumb here

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Concerning a Revision to the Technical Specifications Associated with the Low Temperature Overpressure Protection System and the Pressure-Temperature Limits Report

Enclosure 2

Response to Request for Additional Information (Non-Proprietary)

NRC RAI

Discuss whether the proposed 54 effective full power years pressure-temperature limit curves, and the methodology used to develop these curves, considered all reactor vessel materials (beltline and non-beltline) and replacement ferritic reactor coolant pressure boundary materials (e.g. those installed subsequent to original construction for example, replacement steam generators), consistent with the requirements of Title 10 to the *Code of Federal Regulations,* Part 50, Appendix G.

SNC Response to NRC RAI

Background

The methodology and results for the development of the new 54 EFPY heatup and cooldown pressure and temperature limit curves for Joseph M. Farley (FNP) Units 1 and 2 is contained in WCAP-17122-NP, Revision 0 [Reference 1] and WCAP-17123-NP, Revision 1 [Reference 2], respectively. The FNP Units 1 and 2 54 EFPY Pressure-Temperature (P-T) limit curves in References 1 and 2 were developed using the methodology described in Topical Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Reference 3].

For Westinghouse nuclear steam supply systems, Topical Report WCAP-14040-A, Revision 4 describes the methodology that is used to comply with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements" [Reference 4]. Since only the reactor vessel (RV) undergoes neutron embrittlement, the RV beltline region is considered to be the most limiting reactor coolant system (RCS) component. Therefore, the methodology in WCAP-14040-A, Revision 4 only addresses the RV beltline region of the RCS as the most limiting for the P-T limits. The original NRC Safety Evaluation (SE) for this topical report states, "We find the report to be acceptable for referencing in the administrative controls section of technical specifications for license amendment applications to the extent specified and under the limitations delineated in the report and the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report." The SE further states, "The staff finds the WCAP-14040 methodology consistent with Appendix G to Section III of the ASME Code and SRP Section 5.3.2." and "T is the metal temperature and RTNDT is the ART value of the limiting vessel material" confirming that the reactor vessel is the limiting component evaluated in the development of the P-T limits. Table 1 of the NRC SE provides requirements regarding the fluence methodology, surveillance capsule program requirements, low temperature overpressure protection (LTOP) system requirements, adjusted reference temperature (ART) calculation, and 10 CFR 50 Appendix G temperature requirements, which have all been addressed in WCAP-17122-NP and WCAP-17123-NP, consistent with the NRC SE.

The discussion in this letter report addresses the NRC RAI for the non-beltline reactor vessel components and any replaced ferritic RCS components for FNP Units 1 and 2.

Response to Reactor Vessel Non-Beltline Components

WCAP-14040-A, Revision 4 does not consider the embrittlement of ferritic materials in the area adjacent to the beltline, specifically the stressed inlet and outlet nozzles. The inside corner regions of these nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel; therefore, these components are analyzed in this section. The ART values for the nozzle corner regions are developed as described below.

The initial material properties for the FNP Units 1 and 2 inlet and outlet reactor vessel nozzle materials were originally documented in the FNP Units 1 and 2 PTLRs [Reference 5]. At that time, conservative values for the copper weight-percent (wt%) and initial reference nil-ductility transition temperature (RT_{NDT}) were selected for the FNP Units 1 and 2 nozzle materials to complete the pressurized thermal shock evaluations documented in those reports. However, after the issuance of this NRC RAI, Westinghouse has decided to reevaluate the material properties for the inlet and outlet nozzles as described below.

Nozzle Chemistry Data

Best-estimate nickel (Ni) weight-percent (wt%) values were obtained from the respective PTLRs [Reference 5] for FNP Units 1 and 2. These Ni values were taken directly from the material-specific "check" analyses documented in each nozzle's respective Certified Material Test Report (CMTR). The CMTRs did not contain copper (Cu) wt% values because at the time that the FNP Units 1 and 2 nozzles were manufactured, it was not required for SA-508, Class 2 low-alloy steel. Generic Cu wt% values were previously taken from an Oak Ridge National Laboratory (ORNL) report [Reference 6] for FNP Units 1 and 2. However, based on a recent RAI received by the Callaway Plant on their License Renewal Application [Reference 7], a more conservative Cu wt% value should be selected. Therefore, Westinghouse is now using the best-estimate Cu wt% value of [1][] from Section 4 of the NRC-approved BWRVIP report, BWRVIP-173-A [Reference 8], for the FNP Units 1 and 2 inlet and outlet nozzles.

The Cu wt% value from the ORNL report, 0.16, was the maximum value out of nine data points. This value is greater than the mean of those nine data points plus one standard deviation, which is an acceptable estimation method per Regulatory Guide 1.99, Revision 2 [Reference 9]. However, based on the Callaway RAI, the NRC recommended that a mean plus two standard deviations method be used for additional conservatism when applying generic data to a material. This mean plus two standard deviations methodology was applied to the data in BWRVIP-173-A to determine the more conservative Cu wt% value of [1][]. The data in the BWRVIP report were tabulated from an industry-wide database of SA-508, Class 2 forging materials. Therefore, as stated above, the conservative best-estimate Cu wt% from the BWRVIP report of [1][] was assigned to the FNP Units 1 and 2 inlet and outlet nozzles.

The chemistry factor (CF) values used in this NRC RAI response were recalculated using the 10 CFR 50.61 [Reference 10] methodology. The CF values were calculated using the new wt% copper value of [1][] from BWRVIP-173-A and the previously documented wt% nickel values in parallel with Table 2 of 10 CFR 50.61. The CF values documented in Tables 1 and 2 of this NRC RAI response for FNP Units 1 and 2, respectively, differ from those documented in the respective PTLRs because the Cu wt% values were revised to the conservative values in BWRVIP-173-A for this analysis. The CF values using the Cu wt% documented in the FNP Units 1 and 2 PTLR reports, 0.16, is approximately 15 to 20 degrees Fahrenheit lower, on average, for each of the inlet and outlet nozzles, as compared to the CF values calculated using the Cu wt% from the BWRVIP report, [1][].

Nozzle Initial RT_{NDT} Values

For this NRC RAI response, Westinghouse has updated the conservative initial RT_{NDT} ($RT_{NDT(U)}$) values presented in the respective PTLRs for the FNP Units 1 and 2 inlet and outlet reactor vessel nozzle materials using the BWRVIP-173-A, Alternative Approach 2 methodology for the initial RT_{NDT} determination, contained in Appendix B of that report. Charpy impact test data were plotted using the hyperbolic tangent curve-fit software CVGraph, Version 5.3 to generate the transition temperatures at 35 and 50 ft-lb as specified in the Alternative Approach 2 methodology. These values were then evaluated, per the Alternative Approach 2 methodology presented in BWRVIP-173-A, to determine the new initial RT_{NDT} values for the inlet and outlet nozzle materials for FNP Units 1 and 2. These revised values provide a more accurate representation of the initial RT_{NDT} for the FNP Units 1 and 2 nozzle materials and are summarized in Tables 1 and 2, respectively.

Nozzle Neutron Fluence Values

The FNP Units 1 and 2 calculated neutron fluence projections at the reactor vessel clad/base metal interface at 54 EFPY for the nozzle materials were originally documented in the respective PTLRs. These fluence values were conservatively assigned to the nozzle materials even though they were calculated at the lowest extent of the nozzles, i.e., the nozzle to upper shell weld locations. However, based on the nozzle initial RT_{NDT} value assessment for Inlet Nozzle B6917-2 of Unit 1, the material-specific Charpy data resulted in a significantly higher initial RT_{NDT} value, 29°F, than the other 11 nozzles, which were all less than 0°F. Therefore, for this particular nozzle, fluence at a higher elevation, equivalent to the height of the postulated flaw in this nozzle material, is considered. This elevation was determined using vessel design and in-service inspection drawings. Nevertheless, conservatisms were still included in this elevation determination. The height of the postulated flaw was calculated starting from the centerline of the nozzle to upper shell weld locations, whereas the fluence height calculations started from the lowest extent of this weld. Also, the fluence chosen was at an elevation lower than the actual calculated elevation of the postulated flaw for additional conservatism. Therefore, the fluence used for the FNP Unit 1 inlet nozzle was not taken at the crack tip, but was conservatively below the crack tip.

Nozzle ART Values

The ART values for the nozzle corner regions were calculated and are documented in Tables 1 and 2. The ART values were conservatively calculated at the clad/base metal interface, rather than at the standard vessel 1/4T location. These ART values were then used for the 1/4T flaw evaluation at the nozzle corner region.

Table 1

ART Calculations for the FNP Unit 1 Reactor Vessel Nozzle Materials Using Conservative Fluence Values at the Lowest Extent of the Nozzles at 54 EFPY

Reactor Vessel Material	Wt % Cu[(a)] 1	Wt % Ni[(a)]	CF[(b)] (°F)	Fluence at Lowest Extent of Nozzle[(c)] (n/cm^2, E > 1.0 MeV)	FF	$RT_{NDT(U)}$[(d)] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
Inlet Nozzle B6917-1		0.83	141.45	3.49E+17	0.2397	-18	33.9	0	17.0	33.9	49.4
Inlet Nozzle B6917-2		0.80	141	7.78E+16[(f)]	0.0922	29	13.0	0	6.5	13.0	55.1
Inlet Nozzle B6917-3		0.87	142.05	1.39E+17	0.1365	-48	19.4	0	9.7	19.4	-8.9
Outlet Nozzle B6916-1		0.77	139.95	9.22E+16	0.1037	-17	14.5	0	7.3	14.5	11.9
Outlet Nozzle B6916-2		0.78	140.3	1.26E+17	0.1280	-29	18.0	0	9.0	18.0	7.2
Outlet Nozzle B6916-3		0.78	140.3	2.31E+17	0.1879	-23	26.4	0	13.2	26.4	29.8

Notes for Table 1:

(a) Cu wt% values are the best-estimate values for SA-508, Class 2 low-alloy steel as documented in BWRVIP-173-A. The Ni wt% values are material-specific values as documented in each respective material's CMTR.

(b) CF values were calculated using the Cu and Ni wt% values and Table 2 of 10 CFR 50.61.

(c) Neutron fluence values were taken from the FNP Unit 1 PTLR, unless otherwise noted.

(d) $RT_{NDT(U)}$ values were determined using the Alternative Approach 2 methodology as described in Appendix B of BWRVIP-173-A.

(e) Per 10 CFR 50.61, the base metal nozzle forging materials σ_Δ = 17°F for Position 1.1 without surveillance data. However, σ_Δ need not exceed 0.5 * ΔRT_{NDT}.

(f) Fluence value assigned to this nozzle at the height of the postulated flaw. Conservatisms are still included in this fluence value, as described previously.

Table 2

ART Calculations for the FNP Unit 2 Reactor Vessel Nozzle Materials Using Conservative Fluence Values at the Lowest Extent of the Nozzles at 54 EFPY

Reactor Vessel Material	Wt % Cu[(a)] 1	Wt % Ni[(a)]	CF[(b)] (°F)	Fluence at Lowest Extent of Nozzle[(c)] (n/cm^2, E > 1.0 MeV)	FF	$RT_{NDT(U)}$[(d)] (°F)	ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
Inlet Nozzle B7218-1		0.71	137.85	4.49E+17	0.2760	-55	38.1	0	17.0	34.0	16.6
Inlet Nozzle B7218-2		0.68	136.8	2.54E+17	0.1990	-55	27.2	0	13.6	27.2	-0.9
Inlet Nozzle B7218-3		0.72	138.2	1.86E+17	0.1644	-60	22.7	0	11.4	22.7	-14.3
Outlet Nozzle B7217-1		0.73	138.55	1.26E+17	0.1280	-47	17.7	0	8.9	17.7	-11.4
Outlet Nozzle B7217-2		0.72	138.2	1.72E+17	0.1565	-71	21.6	0	10.8	21.6	-27.7
Outlet Nozzle B7217-3		0.72	138.2	3.04E+17	0.2213	-43	30.6	0	15.3	30.6	18.3

Notes for Table 2:

(a) Cu wt% values are the best-estimate values for SA-508, Class 2 low-alloy steel as documented in BWRVIP-173-A. The Ni wt% values are material-specific values as documented in each respective material's CMTR.

(b) CF values were calculated using the Cu and Ni wt% values and Table 2 of 10 CFR 50.61.

(c) Neutron fluence values were taken from the FNP Unit 2 PTLR.

(d) $RT_{NDT(U)}$ values were determined using the Alternative Approach 2 methodology as described in Appendix B of BWRVIP-173-A.

(e) Per 10 CFR 50.61, the base metal nozzle forging materials σ_Δ = 17°F for Position 1.1 without surveillance data. However, σ_Δ need not exceed 0.5 * ΔRT_{NDT}.

A summary of the limiting inlet and outlet nozzle ART values at FNP Units 1 and 2 is presented in Table 3.

Table 3
Summary of the Limiting ART Values for the Inlet and Outlet Nozzle Materials at FNP Units 1 and 2 at 54 EFPY

Unit	Nozzle Material and ID Number	Limiting ART Value (°F)
FNP 1	Inlet Nozzle B6917-2	55.1
	Outlet Nozzle B6916-3	29.8
FNP 2	Inlet Nozzle B7218-1	16.6
	Outlet Nozzle B7217-3	18.3

A calculation of the FNP Units 1 and 2 nozzle cooldown P-T limits was completed using the inlet and outlet nozzle ART values at 54 EFPY to account for nozzle embrittlement. The stress intensity factor correlations used for the nozzle corners are consistent with the ASME PVP2011-57015 [Reference 11] and ORNL study, ORNL/TM-2010/246 [Reference 12]. The methodology used included postulating an inside surface 1/4T nozzle corner flaw, along with calculating through-wall nozzle corner stresses for a cooldown rate of 100°F/hour.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1x + A_2x^2 + A_3x^3$$

Where:

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

$$K_1 = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

Where:

K_1 = stress intensity factor for a circular nozzle corner crack on a nozzle with a rounded inner radius corner

a = crack depth at the nozzle corner, for use with 1/4T (25% of the wall thickness)

The FNP Units 1 and 2 inlet and outlet nozzle P-T limit curves are shown in Figures 1 through 4, based on the stress intensity factor expression discussed above; also shown in these figures are the traditional beltline P-T limits from WCAP-17122-NP and WCAP-17123-NP. The nozzle P-T limits are provided for a cooldown rate of -100°F/hr, along with a steady-state curve.

It should be noted that an outside surface nozzle flaw was not considered because the pressure stress is significantly lower at the outside surface than the inside surface. A heatup nozzle P-T limit curve is not provided, since it would be less limiting than the nozzle P-T limit curve in Figures 1 through 4 for an inside surface flaw.

Based on the results shown in Figures 1 through 4, it is concluded that the nozzle P-T limits are bounded by the traditional beltline curves. Therefore, the P-T limits provided in WCAP-17122-NP and WCAP-17123-NP for 54 EFPY are still applicable for the beltline and non-beltline reactor vessel components.

Response to Replaced Ferritic Components in the Reactor Coolant System

Both units at FNP have replaced their steam generators and reactor vessel closure heads since original construction. The FNP Units 1 and 2 replacement steam generators are Westinghouse Model 54F design. These were major projects involving detailed design, fabrication, and material testing. These components were evaluated and designed for protection against non-ductile failure. The replacement steam generators were designed to the fracture mechanics requirements of ASME 1989 Code Edition, Section III, Appendix G, "Protection Against Nonductile Failure." The replacement reactor vessel closure heads for FNP Units 1 and 2 were designed to the requirements of 1998 ASME Code Edition, with 2000 Addenda, Section III, Appendix G. ASME Boiler and Pressure Vessel Code, Section III, Appendix G presents a method for obtaining allowable loadings for protection against non-ductile failure for ferritic pressure-retaining materials in Class 1 components.

These components do not have to be addressed in the 54 EFPY pressure-temperature limits, since they have been designed to the requirements of ASME Section III and have not undergone neutron embrittlement that would affect their P-T limits.

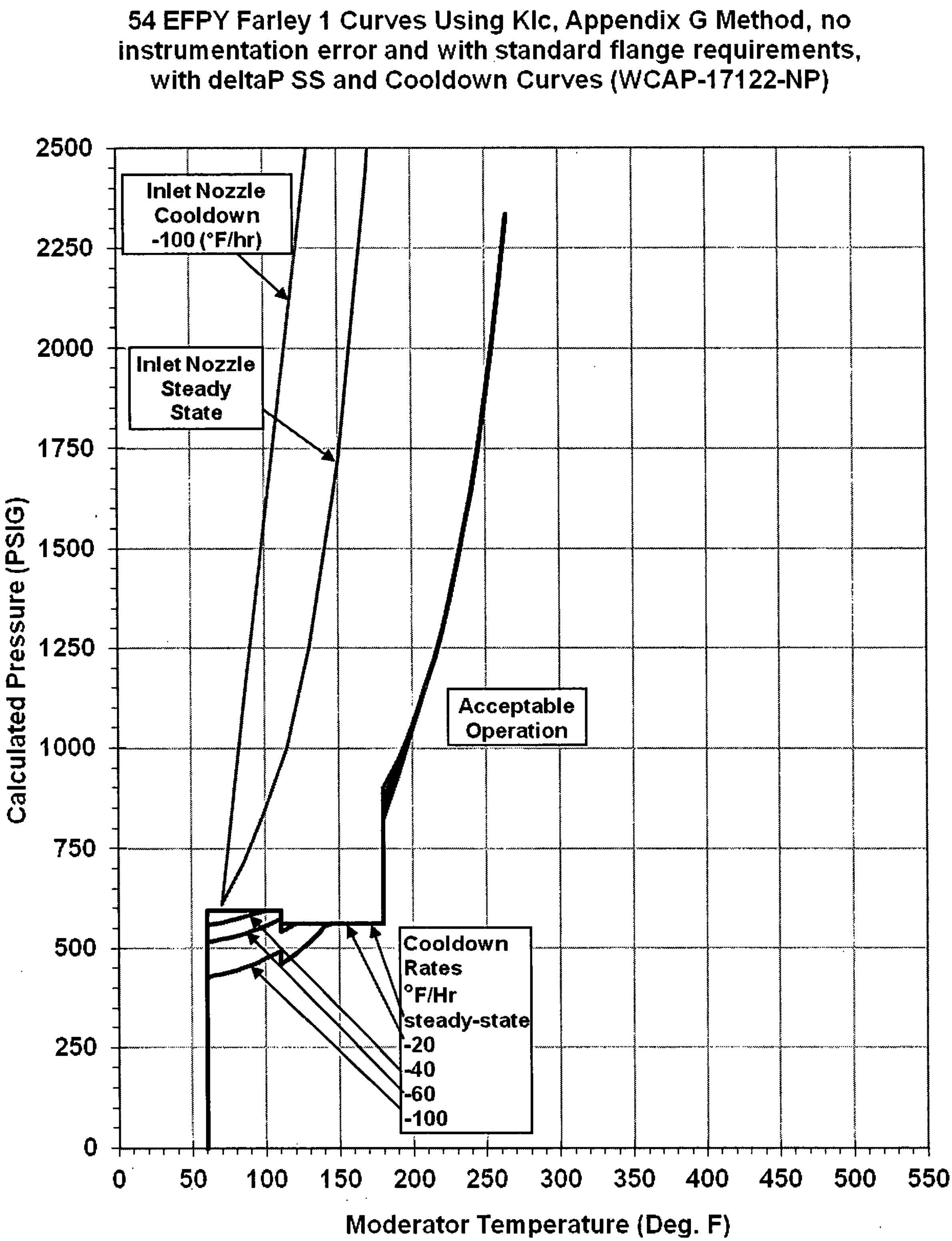

Figure 1: Comparison of FNP Unit 1 WCAP-17122-NP P-T Limits to Inlet Nozzle Limits

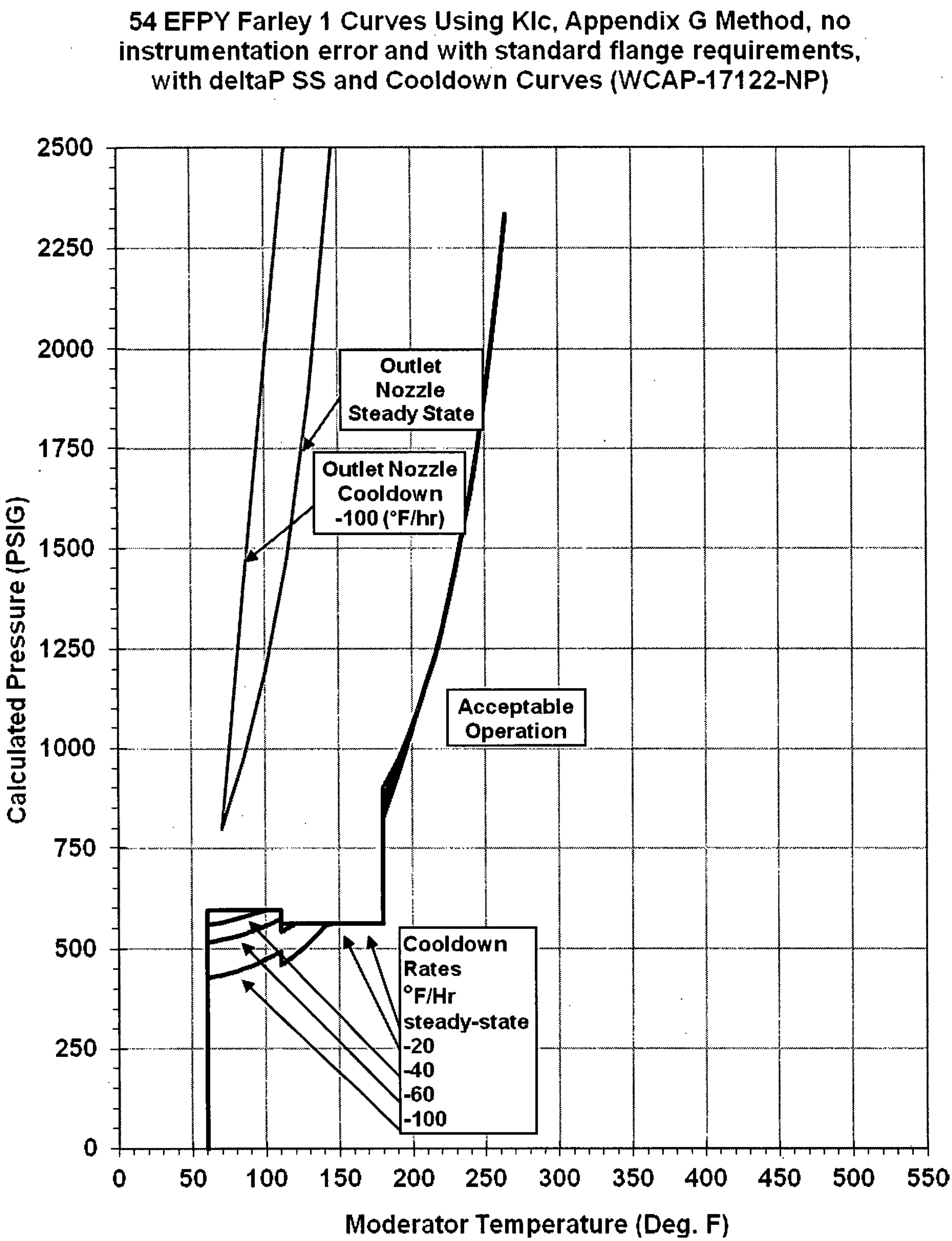

Figure 2: Comparison of FNP Unit 1 WCAP-17122-NP P-T Limits to Outlet Nozzle Limits

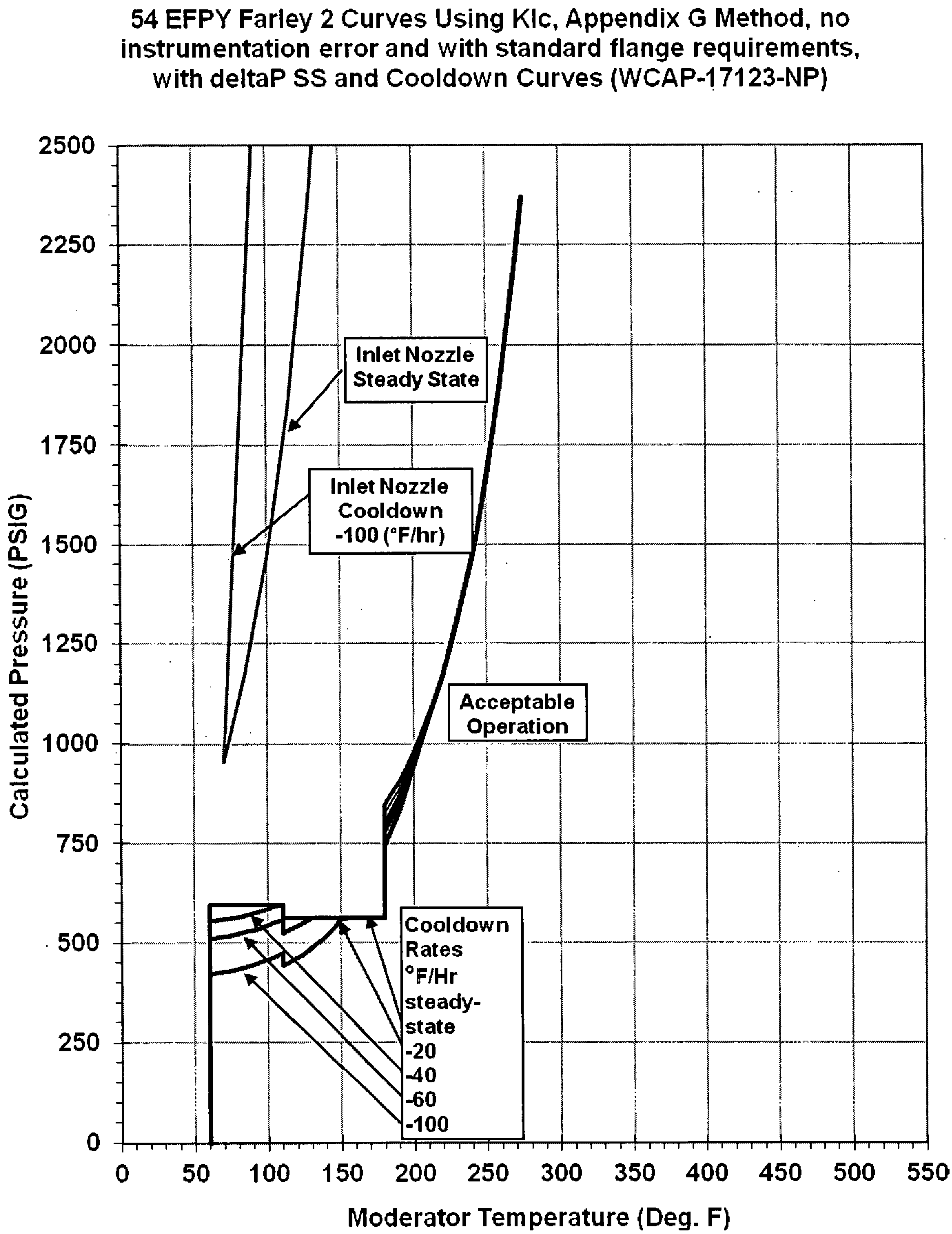

Figure 3: Comparison of FNP Unit 2 WCAP-17123-NP P-T Limits to Inlet Nozzle Limits

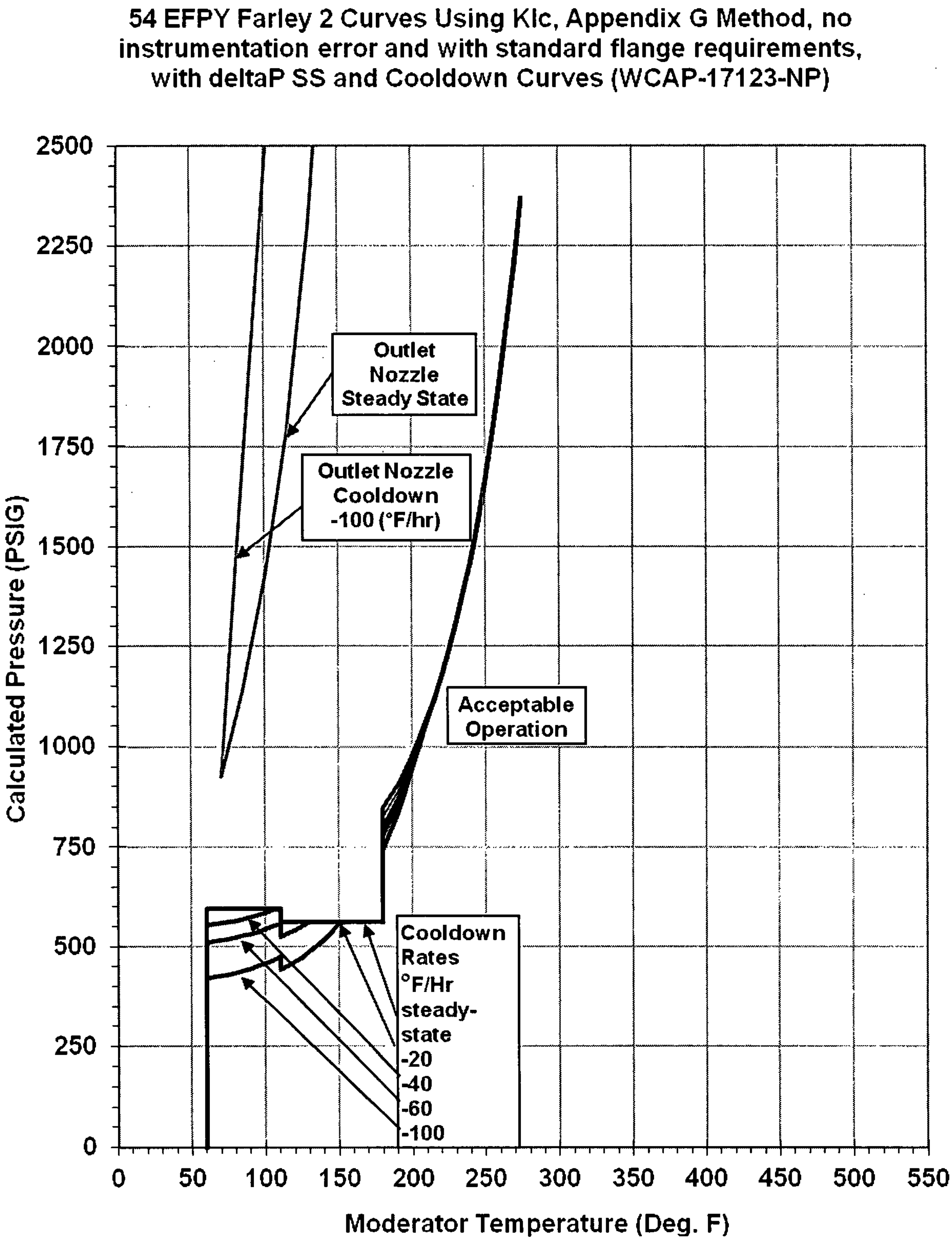

Figure 4: Comparison of FNP Unit 2 WCAP-17123-NP P-T Limits to Outlet Nozzle Limits

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