ATTACHMENT 3

WCAP-13517

EVALUATION OF PRESSURIZED THERMAL SHOCK
FOR D. C. COOK UNIT 2

AMERICAN ELECTRIC POWER
SERVICE CORPORATION

- [x] APPROVED IN GENERAL
- [ ] APPROVED EXCEPT AS NOTED
- [ ] NOT APPROVED
- [ ] FOR REFERENCE ONLY

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1. INTRODUCTION

A limiting condition on reactor vessel integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a Loss-Of-Coolant-Accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization;

- significant degradation of vessel material toughness caused by radiation embrittlement; and

- the presence of a critical-size defect in the vessel wall.

In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criteria on pressurized water reactor (PWR) vessel embrittlement as measured by the nil-ductility reference temperature, termed RT_{PTS}[1]. RT_{PTS} screening values were set for beltline axial welds, forgings or plates and for beltline circumferential weld seams for end-of-license plant operation. The screening criteria were determined using conservative fracture mechanics analysis techniques. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with the criteria through end-of-license. The NRC has amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, May 15, 1991 with an effective date of June 14, 1991[2]. This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2[3].

The purpose of this report is to determine the RT_{PTS} values for the D. C. Cook Unit 2 reactor vessel to address the revised PTS Rule. Section 2 discusses the Rule and its requirements. Section 3 provides the methodology for calculating RT_{PTS}. Section 4 provides the reactor vessel beltline region material properties for the D. C. Cook Unit 2 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5. The results of the RT_{PTS} calculations are presented in Section 6. The conclusions and references for the PTS evaluation follow in Sections 7 and 8, respectively.

2. PRESSURIZED THERMAL SHOCK

The PTS Rule requires that the PTS submittal be updated whenever there are changes in core loadings, surveillance measurements or other information that indicates a significant change in projected RT_{PTS} values.

The Rule outlines regulations to address the potential for PTS events on pressurized water reactor vessels in nuclear power plants that are operated with a license from the United States Nuclear Regulatory Commission (USNRC). PTS events have been shown from operating experience to be transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may result in the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The Rule establishes the following requirements for all domestic, operating PWRs:

* All plants must submit projected values of RT_{PTS} for reactor vessel beltline materials by giving values for time of submittal, the expiration date of the operating license, and the projected expiration date if a change in the operating license or renewal has been requested. This assessment must be submitted within six months after the effective date of this Rule if the value of RT_{PTS} for any material is projected to exceed the screening criteria.

Otherwise, it must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel surveillance capsule report, or within 5 years from the effective date of this Rule change, whichever comes first. These values must be calculated based on the methodology specified in this rule. The submittal must include the following:

1) the bases for the projection (including any assumptions regarding core loading patterns), and
2) copper and nickel content and fluence values used in the calculations for each beltline material. (If these values differ from those previously submitted to the NRC, justification must be provided.)

* The RT_{PTS} (measure of fracture resistance) screening criteria for the reactor vessel beltline region is

  270°F for plates, forgings, axial welds; and,
  300°F for circumferential weld materials.

* The following equations must be used to calculate the RT_{PTS} values for each weld, plate or forging in the reactor vessel beltline:

  Equation 1: $RT_{PTS} = I + M + \Delta RT_{PTS}$

  Equation 2: $\Delta RT_{PTS} = (CF)f^{(0.28-0.10 \log f)}$

* All values of RT_{PTS} must be verified to be bounding values for the specific reactor vessel. In doing this each plant should consider plant-specific information that could affect the level of embrittlement.

* Plant-specific PTS safety analyses are required before a plant is within 3 years of reaching the screening criteria, including analyses of alternatives to minimize the PTS concern.

* NRC approval for operation beyond the screening criteria is required.

3. METHOD FOR CALCULATION OF RT_{PTS}

In the PTS Rule, the NRC Staff has selected a conservative and uniform method for determining plant-specific values of RT_{PTS} at a given time.

For the purpose of comparison with the screening criteria, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate or forging in the beltline region as follows.

$$RT_{PTS} = I + M + \Delta RT_{PTS}, \text{ where } \Delta RT_{PTS} = (CF) f^{(0.28-0.10 \log f)}$$

I = Initial reference temperature (RT_{NDT}) in °F of the unirradiated material

M = Margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. M = 66°F for welds and 48°F for base metal if generic values of I are used.
M = 56°F for welds and 34°F for base metal if measured values of I are used.

f = Neutron fluence, n/cm^2 (E > 1MeV at the clad/base metal interface), divided by 10^{19}

CF = Chemistry factor from tables[2] for welds and for base metal (plates and forgings). If plant-specific surveillance data has been deemed credible per Reg. Guide 1.99, Rev. 2[3], it may be considered in the calculation of the chemistry factor.

4. VERIFICATION OF PLANT-SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties was performed.

The beltline region is defined by the PTS Rule[2] to be "the region of 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 irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Figure 1 identifies and indicates the location of all beltline region materials for the D. C. Cook Unit 2 reactor vessel.

Material property values were obtained from material test certifications from the original fabrication as well as the additional material chemistry tests performed as part of the surveillance capsule program[5]. The average copper and nickel values were calculated for each of the beltline region materials using all the available material chemistry information.

A summary of the pertinent chemical and mechanical properties of the beltline region plate and weld materials of the D. C. Cook Unit 2 reactor vessel are given in Table 1. All of the initial RT_{NDT} values (I-RTNDT) are also presented in Table 1.

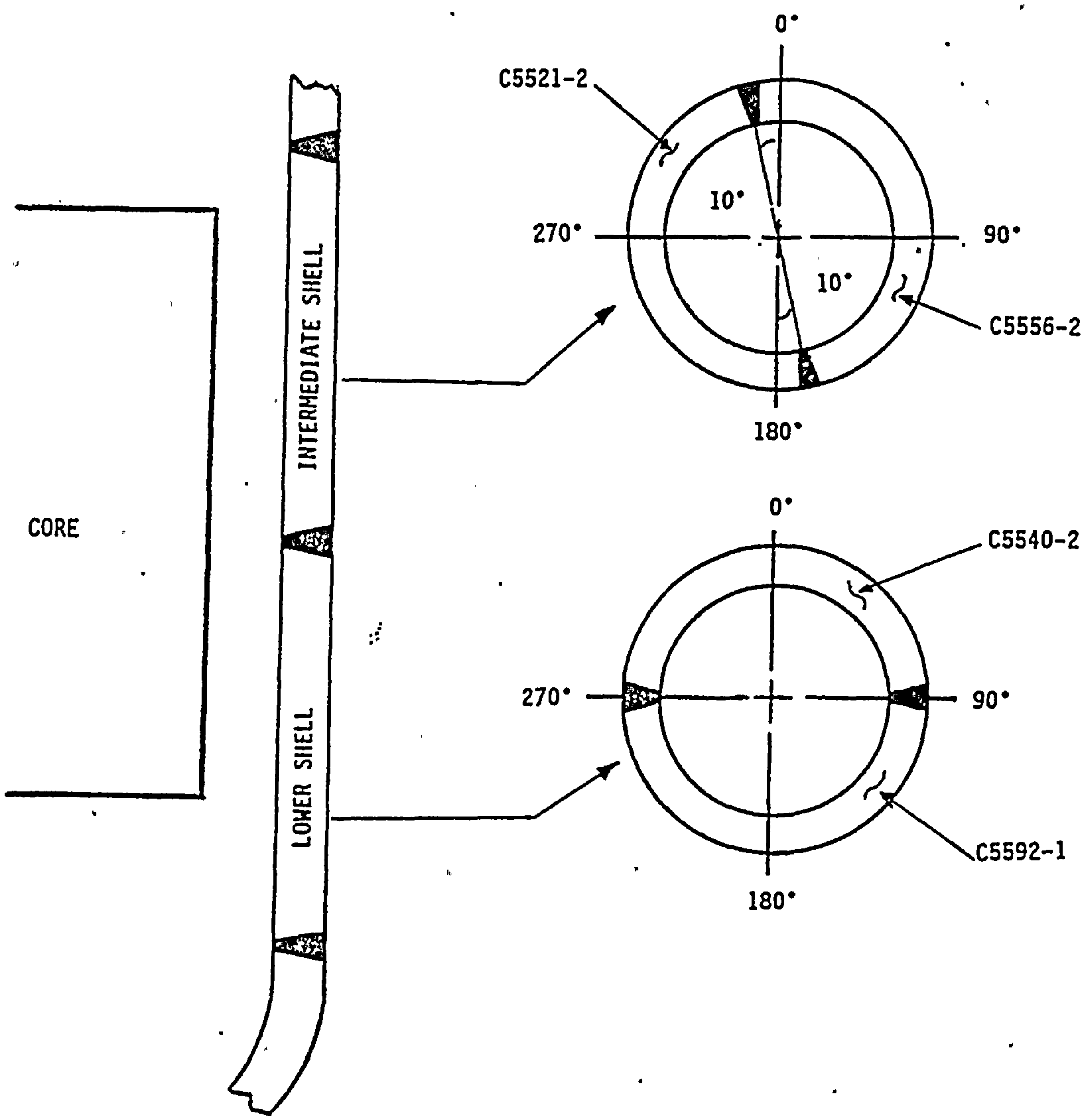

Figure 1. Identification and Location of Beltline Region Materials for the D. C. Cook Unit 2 Reactor Vessel

TABLE 1

D. C. COOK UNIT 2 REACTOR VESSEL

BELTLINE REGION MATERIAL PROPERTIES

Material Description	CU (%)	NI (%)	I-RTNDT (°F)
Intermediate Shell, C5556-2	0.15	0.57	58
Intermediate Shell, C5521-2 *	0.125	0.58	38
Lower Shell, C5540-2	0.11	0.64	-20
Lower Shell, C5592-1	0.14	0.59	-20
Longitudinal Welds *	0.052	0.967	-35
Circumferential Weld *	0.052	0.967	-35

* Mean values of copper and nickel as indicated below

Material	Data Source	Copper (wt. %)	Nickel (wt. %)
Plate, C5521-2	Original Mill Test Report	0.14	0.58
	Surveillance Program [5]	0.11	0.58
	Mean value	0.125	0.58
Weld	Original Mill Test Report	0.05	0.97
	Surveillance Program [5]	0.055	0.97
	Surveillance Program [5]	0.05	0.96
	Mean value	0.052	0.967

5. NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E>1 MeV) at the inner surface of the D. C. Cook Unit 2 reactor vessel is shown in Table 2. These values were projected using the results of the Capsule U radiation surveillance program[4].

TABLE 2

NEUTRON EXPOSURE PROJECTIONS* AT KEY LOCATIONS IN THE D. C. COOK UNIT 2 PRESSURE VESSEL CLAD/BASE METAL INTERFACE FOR 8.65 AND 32 EFPY[4]

EFPY	0°	10°	30°	45°
8.65	0.179	0.244	0.309	0.465
32	0.663	0.902	1.14	1.71

*Fluence x 10^{19} n/cm^2 (E>1.0 MeV)

6. DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the D. C. Cook Unit 2 reactor vessel as a function of present time (8.65 EFPY per Capsule U analysis) and end-of-license 32 EFPY) fluence values. The fluence data were generated based on the most recent surveillance capsule program results[4].

The PTS Rule requires that each plant assess the RT_{PTS} values based on plant specific surveillance capsule data under certain conditions. These conditions are:

- Plant specific surveillance data has been deemed credible as defined in Regulatory Guide 1.99, Revision 2, and

- RT_{PTS} values change significantly. (Changes to RT_{PTS} values are considered significant if the value determined with RT_{PTS} equations (1) and (2), or that using capsule data, or both, exceed the screening criteria prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.)

For D. C. Cook Unit 2, the use of plant specific surveillance capsule data arises for the intermediate shell plate, C5521-2 and the welds because of the following reasons:

1) There have been three capsules removed from the reactor vessel, and the data is deemed credible per Regulatory Guide 1.99, Revision 2.

2) The surveillance capsule materials are representative of the actual vessel materials.

The chemistry factors for the intermediate shell plate, C5521-2 and welds were calculated using the surveillance capsule data as shown in Table 3. All other chemistry factor values for the remaining beltline materials were calculated using the Tables 1 and 2 from Regulatory Guide 1.99, Revision 2.

TABLE 3

CALCULATION OF CHEMISTRY FACTORS USING

D. C. COOK UNIT 2 SURVEILLANCE CAPSULE DATA[4]

Component	Capsule	Fluence	FF	DRTNDT	FF*DRTNDT	(FF)^2
Intermediate Shell	T	0.264	0.638	55	35.072	0.407
Plate C5521-2 (Long.)	Y	0.683	0.893	90	80.378	0.798
	X	1.06	1.016	95	96.548	1.033
	U	1.58	1.126	95	107.004	1.269
Intermediate Shell	T	0.264	0.638	80	51.013	0.407
Plate C5521-2 (Trans.)	Y	0.683	0.893	100	89.309	0.798
	X	1.06	1.016	103	104.679	1.033
	U	1.58	1.126	138	155.438	1.269
					719.442	7.012

Chemistry Factor = 719.442 / 7.012 = 102.61

Component	Capsule	Fluence	FF	DRTNDT	FF*DRTNDT	(FF)^2
Weld Metal	T	0.264	0.638	40	25.507	0.407
	Y	0.683	0.893	50	44.655	0.798
	X	1.06	1.016	70	71.141	1.033
	U	1.58	1.126	75	84.477	1.269
					225.779	3.506

Chemistry Factor = 225.779 / 3.506 = 64.40

Tables 4 and 5 provide a summary of the RT_{PTS} values for all beltline region materials for 8.65 EFPY and end-of-life (32 EFPY), respectively, using the PTS Rule.

TABLE 4

RT_{PTS} VALUES FOR D. C. COOK UNIT 2 FOR 8.65 EFPY

Material	ΔRT_{NDT}(°F) (CF	x FF*)	+ Initial RT_{NDT} (°F)	+ Margin (°F)	= RT_{PTS} (°F)
Intermediate Shell Plate, C5556-2	108.35	0.787	58	34	177
Intermediate Shell Plate, C5521-2	86.50 (102.61)	0.787 0.787	38 38	34 34	140 (153)
Lower Shell Plate, C5540-2	74.60	0.787	-20	34	73
Lower Shell Plate, C5592-1	99.55	0.787	-20	34	92
Intermediate Shell Longitudinal Welds	70.80 (64.40)	0.618 0.618	-35 -35	56 56	65 61
Lower Shell Longitudinal Welds	70.80 (64.40)	0.543 0.543	-35 -35	56 56	60 56
Circumferential Weld Seam	70.80 (64.40)	0.787 0.787	-35 -35	56 56	77 72

() Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 4.65 x 10^{18} n/cm^2 [4], except for the longitudinal welds. For the intermediate shell longitudinal welds, the fluence factor is based on a neutron fluence of 2.44 x 10^{18} n/cm^2 [4] at the inner surface of the weld at the 10° location. For the lower shell longitudinal welds, the fluence factor is based on a neutron fluence of 1.79 x 10^{18} n/cm^2 [4] at the inner surface of the weld at the 0° location.

TABLE 5

RT_{PTS} VALUES FOR D. C. COOK UNIT 2 FOR 32 EFPY

Material	ΔRT_{NDT}(°F) + (CF x FF*)		Initial RT_{NDT} + (°F)	Margin = (°F)	RT_{PTS} (°F)
Intermediate Shell Plate, C5556-2	108.35	1.148	58	34	216
Intermediate Shell Plate, C5521-2	86.50 (102.61)	1.148 1.148	38 38	34 34	171 (190)
Lower Shell Plate, C5540-2	74.60	1.148	-20	34	100
Lower Shell Plate, C5592-1	99.55	1.148	-20	34	128
Intermediate Shell Longitudinal Welds	70.80 (64.40)	0.971 0.971	-35 -35	56 56	90 85
Lower Shell Longitudinal Welds	70.80 (64.40)	0.885 0.885	-35 -35	56 56	84 78
Circumferential Weld Seam	70.80 (64.40)	1.148 1.148	-35 -35	56 56	102 95

() Indicates numbers were calculated using surveillance capsule data.

* Fluence factor based upon peak inner surface neutron fluence of 1.71 x 10^{19} n/cm^2[4], except for the longitudinal welds. For the intermediate shell longitudinal welds, the fluence factor is based on a neutron fluence of 9.02 x 10^{18} n/cm^2 [4] at the inner surface of the weld at the 10° location. For the lower shell longitudinal welds, the fluence factor is based on a neutron fluence of 6.63 x 10^{18} n/cm^2 [4] at the inner surface of the weld at the 0° location.

7. CONCLUSIONS

As shown in Tables 4 and 5, all the RT_{PTS} values remain below the NRC screening values for PTS using the fluence values for the present time (8.65 EFPY) and the projected fluence values for the end-of-life (32 EFPY). A plot of the RT_{PTS} values versus the fluence is shown in Figure 2 for the most limiting material, the intermediate shell plate, C5556-2 in the D. C. Cook Unit 2 reactor vessel beltline region.

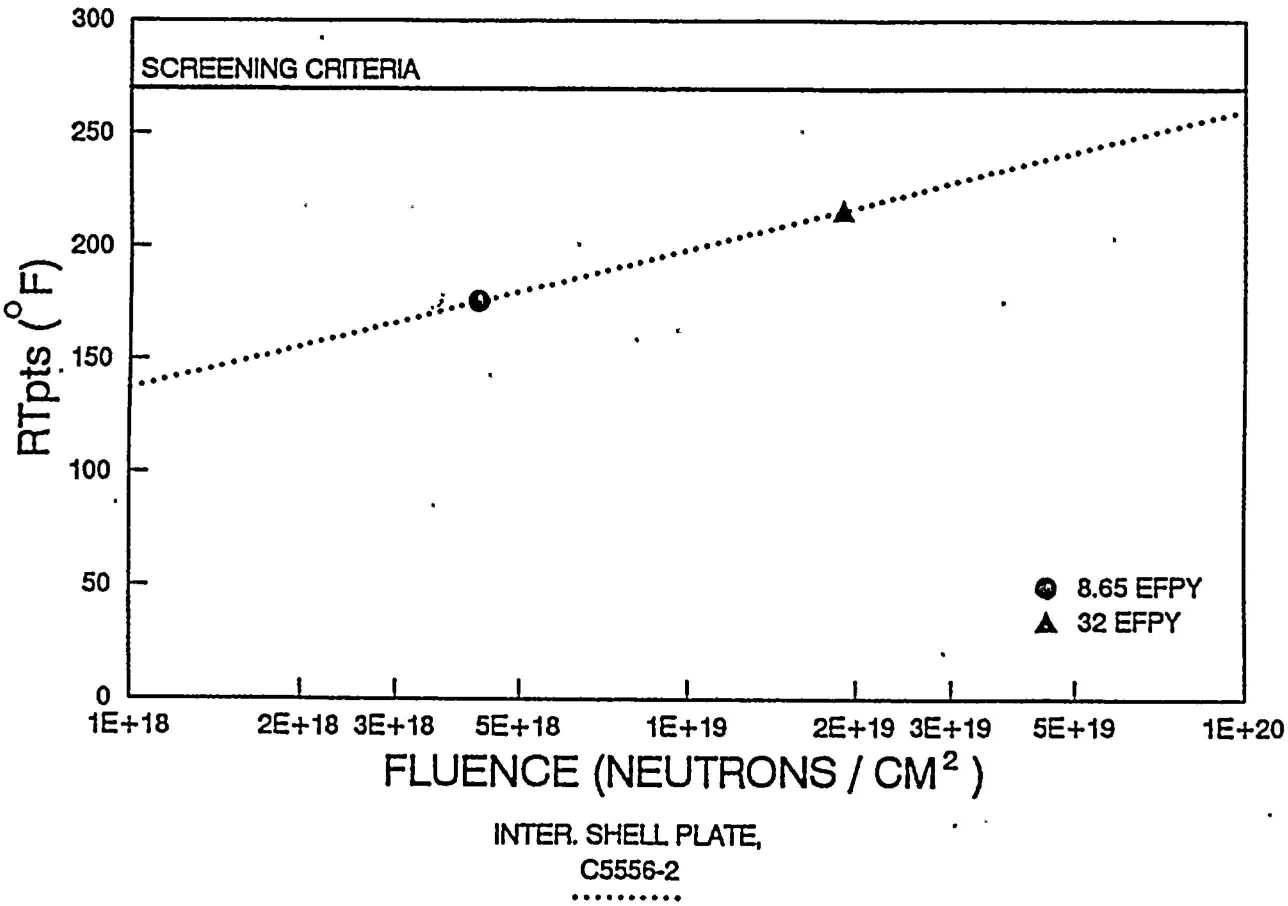

Figure 2. RT_{PTS} versus Fluence Curves for D. C. Cook Unit 2 Limiting Material - Intermediate Shell Plate, C5556-2

8. REFERENCES

[1] 10 CFR Part 50, "Analysis of Potential Pressurized Thermal Shock Events," July 23, 1985.

[2] 10 CFR Part 50, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)

[3] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

[4] WCAP-13515, "Analysis of Capsule U from the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program," page 6-28, J. M. Chicots, et al., October 1992. (Westinghouse Proprietary Class 3)

[5] WCAP-8512, "American Electric Power Company Donald C. Cook Unit No. 2 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, et al., November 1975.

[6] MT/SMART-090(89), "D. C. Cook Unit 2 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation", N. K. Ray, April 1989, Table 1.

ENCLOSURE 2 TO AEP:NRC:1173C

MARKUP OF PRESSURIZED THERMAL SHOCK AND
UPPER SHELF ENERGY SUMMARY TABLES

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{NDT}	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
D. C. Cook 1 EOL: 10/25/ 2014	Nozzle Shell B4405-1	C3594	~~1.41E19~~ (1) 1.10 E 18	2°F	MTEB 5-2	93.7	Table	0.14	0.46
	Nozzle Shell B4405-2	C3594	~~[illegible]~~ (1) 1.10E18	34°F	MTEB 5-2	93.25	Table	0.14	0.45
	Nozzle Shell B4405-3	C3872	~~1.41E19~~ (1) 1.10E19	40°F	MTEB 5-2	94.6	Table	0.14	0.43
	Int. Shell B4406-1	C1260	1.41E19	5°F	Plant Specific	81.4	Table	0.12	0.52
	Int. Shell B4406-2	C3506	1.41E19	33°F	Plant Specific	104.5	Table	0.15	0.50
	Int. Shell B4406-3	C3506	1.41E19	40°F	Plant Specific	102.94	Calculated	0.15	0.49
	Lower Shell B4407-1	C3929	1.41E19	23°F	Plant Specific	97.75	Table	0.14	0.55
	Lower Shell B4407-2	C3932	1.41E19	-12°F	Plant Specific	82.8	Table	0.12	0.59
	Lower Shell B4407-3	C3929	1.41E19	38°F	Plant Specific	95.5	Table	0.14	0.50
	Nozzle Shell Axial Welds 1-442 A/C	13253 and 12008 (T)	~~1.41E19~~ (1) 1.10E18	-56°F	Generic	206.4	Table	~~0.27~~ (3) 0.28	0.74
	Nozzle/ Int. Shell Circ. Weld 8-442	20291	~~1.41E19~~ (1) 1.10 E18	-56°F	Generic	232	Table	0.35	0.74
	Int. to Lower Shell Circ. Weld 9-442	1P3571	1.41E19	-56°F	Generic	219	Table	~~0.37~~ (3) 0.28	0.74
	Int. Shell Axial Welds 2-442A/C	13253 and 12008 (T)	~~1.41E19~~ (2) 0.95 E19	-56°F	Generic	206.4	Table	~~0.27~~ (3) 0.28	0.74
	Lower Shell Axial Welds 3-442A/C	13253 and 12008 (T)	~~1.41E19~~ (2) 0.95E19	-56°F	Generic	206.4	Table	~~0.27~~ (3) 0.28	0.74

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{NDT}	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni

References D.C. Cook 1

All data except as noted below were from the July 13, 1992 letter from E.E. Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plant Units 1 and 2... Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity."

Information regarding chemical composition, initial RT_{NDT} and methodology of determination for the UUSE for the welds are from the November 29, 1993 letter from E.E. Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plant Units 1 and 2... Response to Request for Additional Information for Generic Letter 92-01, Revision 1." No value for % Cu of weld 8-442 was provided, therefore the default value of 0.35 was used. ~~The UUSE values for welds 1-442, 2-442 and 3-442 represent an NRC staff calculated average of D.C. Cook and Sister plant data (McGuire, Unit 1 and Diablo Canyon, Unit 2) for weld wire heats containing heat nos. 13253 and 12008.~~

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Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT$_{NDT}$	Method of Determin. IRT$_{NDT}$	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
D. C. Cook 2 EOL: 12/23/ 2017	Int. Shell 10-1	C5556-2	1.71E19	58°F	Plant Specific	108.9	Table	0.15	~~0.58~~ 6 0.57
	Int. Shell 10-2	C5521-2	1.71E19	38°F	Plant Specific	102.61	Calculated	~~0.14~~ 6 0.125	0.58
	Lower Shell 9-1	C5540-2	~~1.71E18~~ 5 1.71E19	-20°F	Plant Specific	74.6	Table	0.11	0.64
	Lower Shell 9-2	C5592-1	~~1.71E18~~ 5 1.71E19	-20°F	Plant Specific	100	Table	0.14	~~0.60~~ 6 0.59
	Int. Shell Axial Welds	S3986	9.02E18	-35°F	Plant Specific	64.4	Calculated	0.05	0.97
	Lower Shell Axial Welds	S3986	6.63E18	-35°F	Plant Specific	64.4	Calculated	0.05	0.97
	Circum. Weld	S3986	1.71E19	-35°F	Plant Specific	64.4	Calculated	0.05	0.97

References

Initial RT$_{NDT}$ and chemical composition for the plates and are from the July 13, 1992 letter from E.E...Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plant Units 1 and 2... Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity."

Updated fluences, chemical composition, initial RT$_{NDT}$ and calculated chemistry factors are from the April 12, 1993 letter from E.E. Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plant Unit 2...Updated Reference Temperature and Pressurized Thermal Shock Analyses."

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
D. C. Cook 1	Nozzle Shell B4405-1	C3594	A 533B-1	68	~~8.47E18~~ 9 5.77E17	87	65%
EOL: 10/25/ 2014	Nozzle Shell B4405-2	C3594	A 533B-1	72	~~8.47E18~~ 9 5.77E17	92	65%
	Nozzle Shell B4405-3	C3872	A 533B-1	63	~~8.47E18~~ 9 5.77E17	80	65%
	Int. Shell B4406-1	C1260	A 533B-1	66	8.47E18	83	Direct
	Int. Shell B4406-2	C3506	A 533B-1	74	8.47E18	96	Direct
	Int. Shell B4406-3	C3506	A 533B-1	76	8.47E18	98	Direct
	Lower Shell B4407-1	C3929	A 533B-1	80	8.47E18	103	Direct
	Lower Shell B4407-2	C3932	A 533B-1	101	8.47E18	126	Direct
	Lower Shell B4407-3	C3929	A 533B-1	84	8.47E18	108	Direct
	Nozzle Shell Axial Welds 1-442 A/C	13253 and 12008 (T)	Linde 1092, SAW	66 7	~~8.47E18~~ 9 5.77E17	108	NRC Generic
	Nozzle/Int. Shell Circ. Weld. 8-442	20291	Linde 1092, SAW	64	~~8.47E18~~ 9 5.77E17	110	Sister Plant
	Int. to Lower Shell Circ. Weld, 9-442	1P3571	Linde 1092, SAW	61 7	8.47E18	105	Sister Plant
	Int. Shell Axial Welds 2-442A/C	13253 and 12008 (T)	Linde 1092, SAW	66 7	8.47E18	108	NRC Generic
	Lower Shell Axial Welds 3-442A/C	13253 and 12008 (T)	Linde 1092, SAW	66 7	8.47E18	103	NRC Generic

Summary File for Upper Shelf Energy

<table>
<tr><th>Plant Name</th><th>Beltline Ident.</th><th>Heat No.</th><th>Material Type</th><th>1/4T USE at EOL</th><th>1/4T Neutron Fluence at EOL</th><th>Unirrad. USE</th><th>Method of Determin. Unirrad. USE</th></tr>
<tr><td colspan="8">
<u>References</u> D.C. Cook 1

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Information regarding UUSE and methodology of determination for the UUSE for the welds are from the November 29, 1993 letter from E.E. Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plant Units 1 and 2... Response to Request for Additional Information for Generic Letter 92-01, Revision 1. ~~No value for % Cu of weld 8-442 was provided, therefore the default value of 0.35 was used.~~

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The UUSE values for welds 1-442, 2-442 and 3-442 represent the NRC staff calculated average of D.C. Cook and Sister Plant data (McGuire, Unit 1 and Diablo Canyon, Unit 2) for weld wire heats containing heat nos. 13253 and 12008.

The UUSE for welds 9-442 is from WCAP-12819, "Analysis of the Maine Yankee Reactor Vessel Second Wall Capsule located at 253°," March 1991.

The UUSE for welds 8-442 is from WCAP-10786, the Report for Surveillance Capsule U from McGuire Unit 1.
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Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
D. C. Cook 2 EOL: 12/23/ 2017	Int. Shell 10-1	C5556-2	A 533B-1	67.5	1.02E19	90	Direct
	Int. Shell 10-2	C5521-2	A 533B-1	58.4	1.02E19	86	Direct
	Lower Shell 9-1	C5540-2	A 533B-1	88.1	1.02E19	110	Direct
	Lower Shell 9-2	C5592-1	A 533B-1	78.4	1.02E19	103	Direct
	Int. Shell Axial Welds	S3986	Linde 124 SAW	64	5.38E18	77	Direct
	Lower Shell Axial Welds	S3986	Linde 124 SAW	65	3.95E18	77	Direct
	Circ. Weld	S3986	Linde 124, SAW	62	1.02E19	77	Direct

References

UUSE data for the plates and welds are from the July 15, 1992 letter from E.E. Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plant Units 1 and 2... Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity."y

Information regarding the methodology of determination for the UUSE of plates C5556-2, C5540-2, and C5592-1 is from the November 29, 1993 letter from E.E. Fitzpatrick to T.E. Murley, "Donald C. Cook Nuclear Plants Units 1 and 2... Response to Request for Additional Information for Generic Letter 92-01, Revision 1.

Updated fluences are from the April 12, 1993 letter from E.E. Fitzpatrick to T.E. Murley, "Donald Cook Nuclear Plant Unit 2...Updated Reference Temperature and Pressurized Thermal Shock Analyses."