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SOUTHWEST RESEARCH INSTITUTE
Post Office Drawer 28510, 6220 Culebra Road
San Antonio, Texas 78284

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM FOR DONALD C. COOK UNIT NO. 2: ANALYSIS OF CAPSULE X

By
P. K. Nair
M. L. Williams (Consultant)

FINAL REPORT
SwRI Project 06-8888

For
Indiana & Michigan Electric Company
Donald C. Cook Nuclear Plant
Bridgeman, Michigan 49106

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Approved:

G. R. Leverant

Gerald R. Leverant, Director
Department of Materials Sciences

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ABSTRACT

Capsule X, the third vessel material surveillance capsule removed from the Donald C. Cook Unit No. 2 nuclear power plant has been tested, and the results have been evaluated. The analysis of the data indicates that the pressure material will retain adequate shelf toughness throughout the 32 EFPY design lifetime. Heatup and cooldown limit curves for normal operation have been developed for up to 12 effective full power years of operation.

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1.0 SUMMARY OF RESULTS AND CONCLUSIONS

The analysis of the third material surveillance capsule removed from the Donald C. Cook Unit No. 2 reactor pressure vessel led to the following conclusions:

(1) Based on a calculated neutron spectral distribution, Capsule X received a fast fluence of 1.002×10^{19} neutrons/cm^2 (E > 1 MeV) at its radial center line.

(2) The surveillance specimens of the core beltline materials experienced shifts in RT_{NDT} of 70°F to 103°F as a result of exposure up to the 1986 refuelling outage.

(3) The core beltline plate materials exhibited the largest shifts in RT_{NDT}. Since the intermediate shell plate material has the highest initial (unirradiated) RT_{NDT}, it will control the heatup and cooldown limitations throughout the design lifetime of the pressure vessel.

(4) The estimated maximum neutron fluence of 3.406×10^{18} neutrons/cm^2 (E > 1 MeV) received by the vessel wall accrued in 5.273 effective full power years (EFPY). The projected maximum neutron fluence after 32 EFPY is 2.067×10^{19} neutrons/cm^2 (E > 1 MeV). This estimate is based on the average fluence rate after 5.273 EFPY of operations.

(5) Based on the analyses of Capsules T, Y and X, the projected values of RT_{NDT} for the Donald C. Cook Unit 2 vessel core beltline region, at the 1/4T and 3/4T positions after 12 EFPY of operation, are 146°F and 102°F, respectively. These values were used as the bases for computing revised heat-up and cooldown limit curves for up to 12 EFPY of operation.

(6) Based on the analyses of Capsules T, Y and X, the values of RT_{NDT} for the Donald C. Cook Unit 2 vessel core beltline region, at the 1/4T and 3/4T positions after 32 EFPY of operation, are projected to be 163°F and

130°F, respectively.

(7) The Donald C. Cook Unit No. 2 vessel plates, weld metal, and HAZ material located in the core beltline region are projected to retain sufficient toughness to meet the current requirements of 10CFR50 Appendix G throughout the design life of the unit.

2.0 BACKGROUND

The allowable loadings on nuclear pressure vessels are determined by applying the rules in Appendix G, "Fracture Toughness Requirements," of 10CFR50 [1]. In the case of pressure-retaining components made of ferritic materials, the allowable loadings depend on the reference stress intensity factor (K_{IR}) curve indexed to the reference nil ductility temperature (RT_{NDT}) presented in Appendix G, "Protection Against Non-Ductile Failure," of Section III of the ASME Code [2]. Further, the materials in the beltline region of the reactor vessel must be monitored for radiation-induced changes in RT_{NDT} per the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10CFR50.

The RT_{NDT} is defined in paragraph NB-2331 of Section III of the ASME Code as the highest of the following temperatures:

(1) Drop-weight Nil Ductility Temperature (DW-NDT) per ASTM E 208 [3];

(2) 60 deg F below the 50 ft-lb Charpy V-notch (C_v) temperature;

(3) 60 deg F below the 35 mil C_v temperature.

The RT_{NDT} must be established for all materials, including weld metal and heat-affected zone (HAZ) material as well as base plates and forgings, which comprise the reactor coolant pressure boundary.

It is well established that ferritic materials undergo an increase in strength and hardness and a decrease in ductility and toughness when exposed to neutron fluences in excess of 10^{17} neutrons per cm^2 (E > 1 MeV) [4]. Also, it has been established that tramp elements, particularly copper and phosphorus, affect the radiation embrittlement response of ferritic materials [5-7]. The relationship between increase in RT_{NDT} and copper content is opening loading (WOL) fracture mechanics specimens. Current technology limitations result in the testing of these specimens at temperatures well below the minimum service temperature in order to obtain valid fracture mechanics data per ASTM E 399 [10], "Standard Method of Test for Plane-Strain Fracture Toughness of Metallic Materials." Currently, these specimens are being stored pending an acceptable testing procedure like the J_{Ic} fracture testing [11] has been defined.

This report describes the results obtained from testing the contents of Capsule X. These data and those obtained previously from Capsules T and Y are analyzed to estimate the radiation-induced changes in the mechanical properties of the pressure vessel at the time of the refuelling outage as well as predicting the changes expected to occur at selected times in the future operation of the Donald C. Cook Unit No. 2 power plant.

3.0 DESCRIPTION OF MATERIAL SURVEILLANCE PROGRAM

The Donald C. Cook Unit No. 2 material surveillance program is described in detail in WCAP 8512 [12], dated November 1975. Eight materials surveillance capsules were placed in the reactor vessel between the thermal shield and the vessel wall prior to startup, see Figure 1. The vertical center of each capsule is opposite the vertical center of the core.

The capsules each contain Charpy V-notches, tensile, and WOL Specimens machined from the SA533 Gr B, CL 2 plate, weld metal, and heat-affected zone (HAZ) materials located at the core beltline. The chemistries and heat treatments of the vessel surveillance materials are summarized in Table 3.1. All test specimens were machined from the test materials at the quarter-thickness (1/4 T) location after performing a simulated postweld stress-relieving treatment. Weld and HAZ specimens were machined from a stress-relieved weldment which joined sections of the intermediate and lower shell plates. HAZ specimens were obtained from the plate C5521-2 side of the weldment. The longitudinal base metal C_V specimens were oriented with their long axis parallel to the primary rolling direction and with V-notches perpendicular to the major plate surfaces. The transverse base metal C_V specimens were oriented with their long axis perpendicular to the primary rolling direction and with V-notches perpendicular to the major plate surfaces. Tensile specimens were machined with the longitudinal axis perpendicular to the plate primary rolling direction. The WOL specimens were machined with the simulated crack parallel to the primary rolling direction and perpendicular to the major plate surfaces. All mechanical test specimens, see Figure 2, were taken at least one plate thickness from the quenched edges of the plate material.

Capsule X contained 44 Charpy V-notched specimens (8 longitudinal and

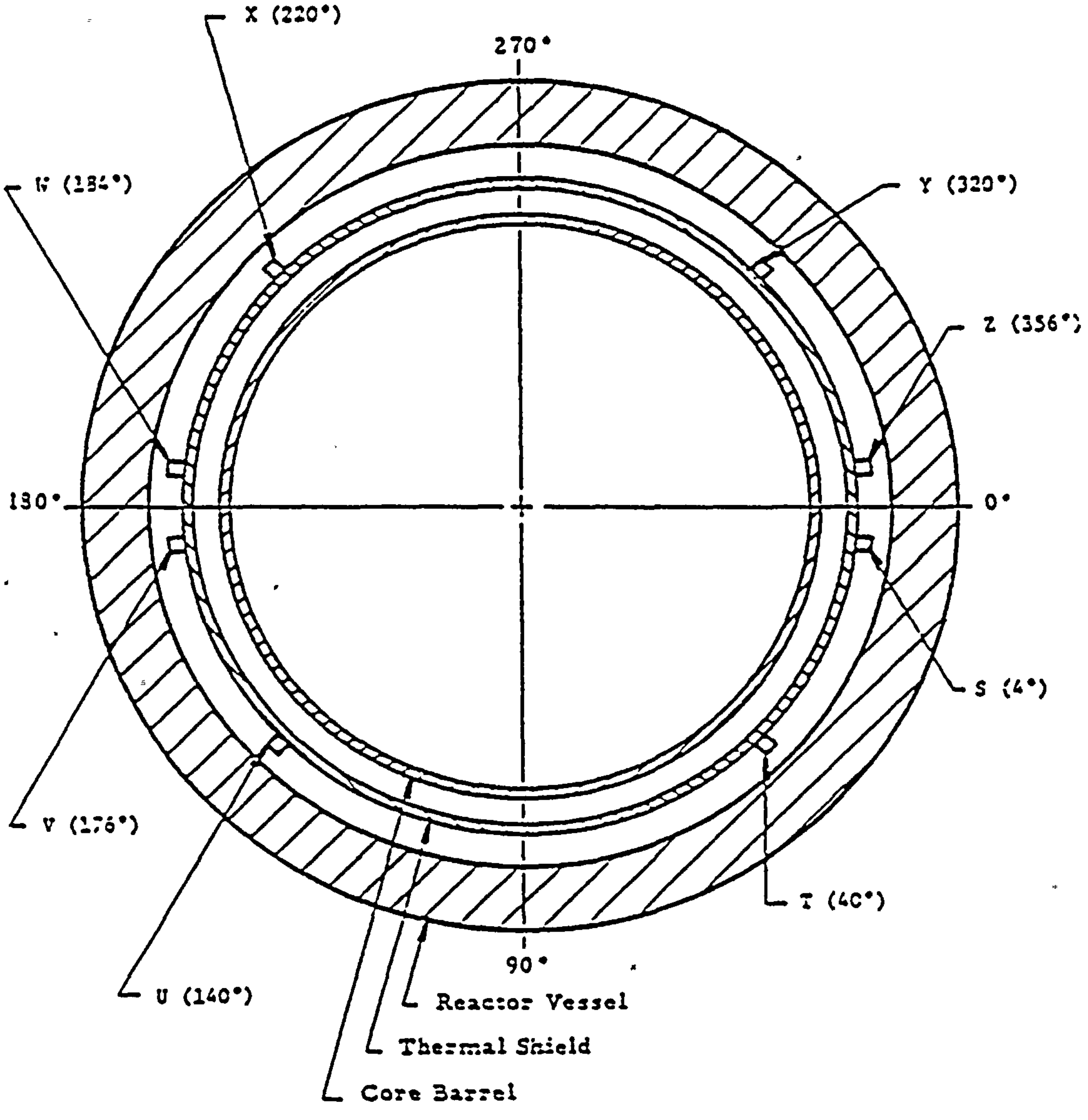


FIGURE 1.. ARRANGEMENT OF SURVEILLANCE CAPSULES IN THE PRESSURE VESSEL

TABLE 3.1

DONALD C. COOK UNIT NO. 2 REACTOR VESSEL SURVEILLANCE MATERIALS [12]

Heat Treatment History

Shell Plate Material:

Heated to 1700 F for 4-1/2 hours. water quenched.
Heated to 1600 F for 5 hours, water quenched.
Tempered at 1250 F for 4-1/2 hours, air cooled.
Stress relieved at 1150 F for 51-1/2 hours, furnace cooled.

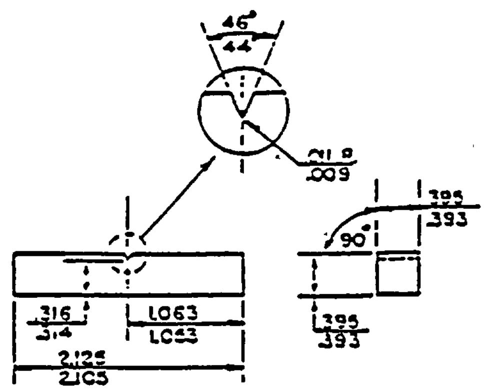
Weldment:

Stress relieved at 1140 F for 9 hours, furnace cooled.

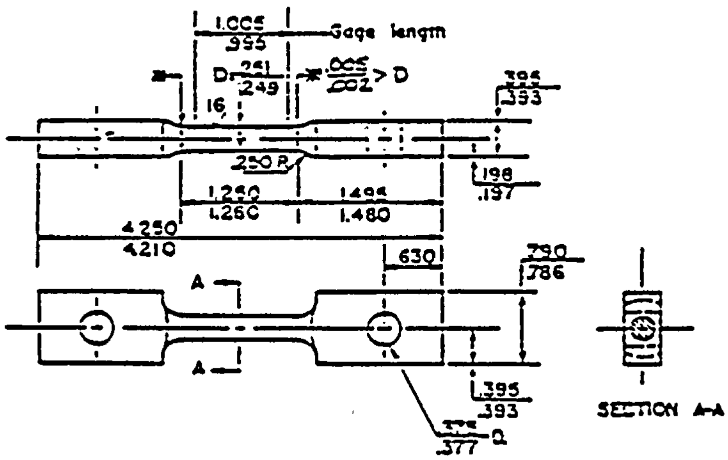
Chemical Composition (Percent)

Material	C	Mn	P	S	Si	Ni	Mo	Cu	Cr
Plate C-5521-2[a]	0.21	1.29	0.013	0.015	0.16	0.58	0.50	0.14	---
Plate C-5521-2[b]	0.22	1.28	0.017	0.014	0.27	0.58	0.55	0.11	0.072
Weld Metal[b]	0.11	1.33	0.022	0.012	0.44	0.97	0.54	0.055	0.068
Weld Metal[c]	0.08	1.42	0.019	0.016	0.36	0.96	--	0.05	0.07

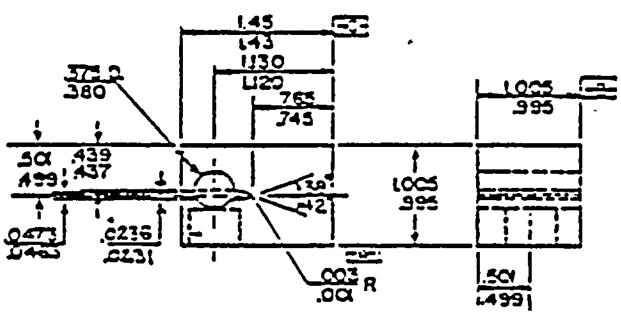
(a) Lukens Steel analysis.
(b) Westinghouse analysis.
(c) Chicago Bridge and Iron analysis.



(a) Charpy v-notch impact specimen



(b) Tensile specimen



(c) Wedge opening loading specimen

FIGURE 2. VESSEL MATERIAL SURVEILLANCE SPECIMENS

12 transverse from the plate material, plus 12 each from weld metal and HAZ material); 4 tensile specimens (2 plate and 2 weld metal); and 4 transverse plate WOL specimens. The specimen numbering system and location within Capsule X is shown in Figure 3.

Capsule X also was reported to contain the following dosimeters for determining the neutron flux density:

Target Element	Form	Quantity
Iron	Bare wire	5
Copper	Bare wire	3
Nickel	Bare wire	3
Cobalt (in aluminum)	Bare wire	2
Cobalt (in aluminum)	Cd shielded wire	2
Uranium-238	Cd shielded oxide	1
Neptunium-237	Cd shielded oxide	1

Two eutectic alloy thermal monitors had been inserted in holes in the steel spacers in Capsule X. One (located at the bottom) was 2.5% Ag and 97.5% Pb with a melting point of 579°F. The other (located at the top of the capsule) was 1.75% Ag, 0.75% Sn, and 97.5% Pb having a melting point of 590°F.

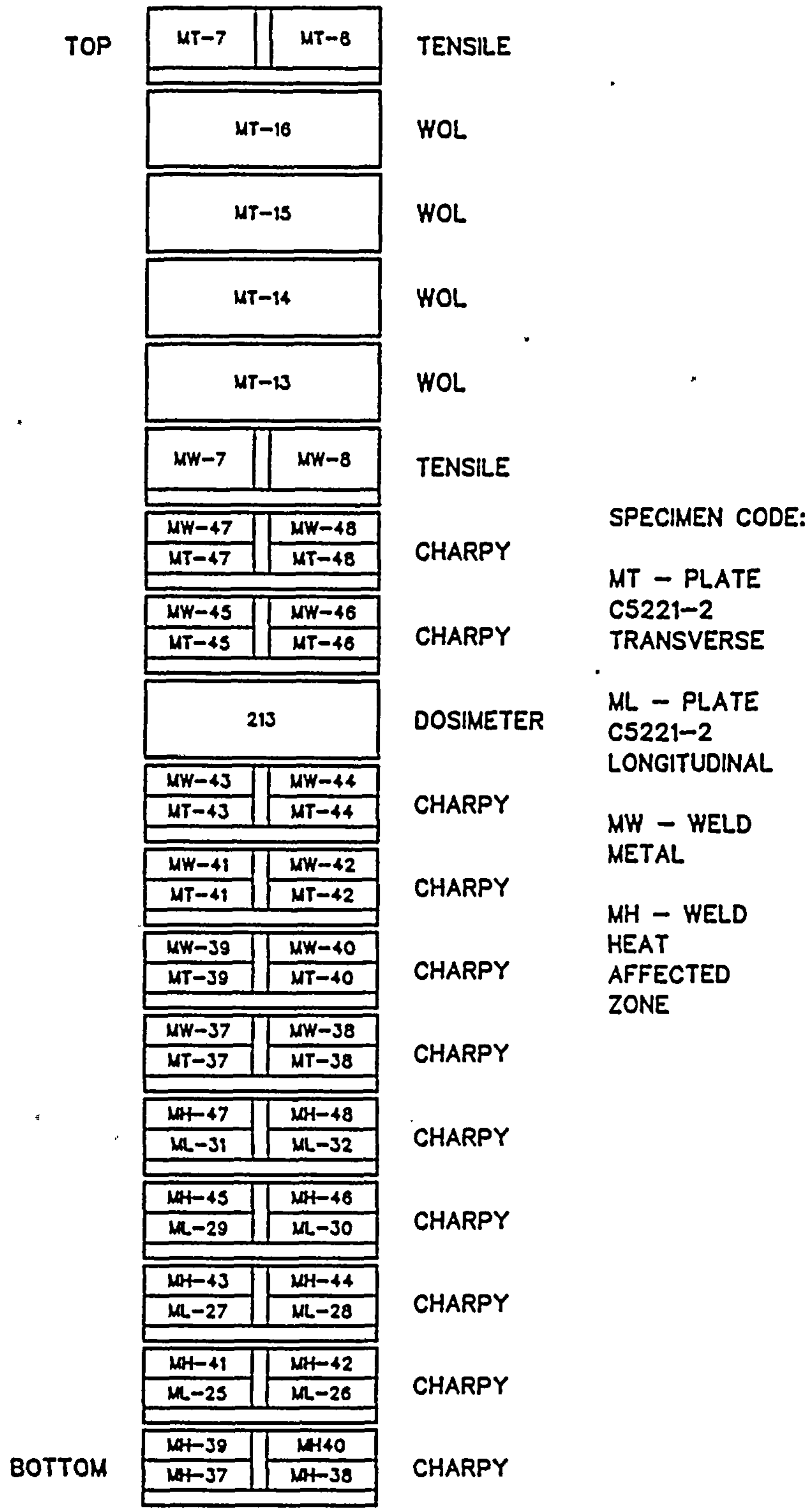


FIGURE 3. ARRANGEMENT OF SPECIMENS IN CAPSULE X

4.0 TESTING OF SPECIMENS FROM CAPSULE X

The capsule shipment, capsule opening, specimen testing, and reporting of results were carried out in accordance with the Project Plan for Donald C. Cook Unit No. 2 Reactor Vessel Irradiation Surveillance Program. The SwRI Nuclear Projects Operating Procedures called out in this plan include:

(1) XI-MS-101-1, "Determination of Specific Activity and Analysis of Radiation Detector Specimens"

(2) XI-MS-103-1, "Conducting Tension Tests on Metallic Specimens"

(3) XI-MS-104-1, "Charpy Impact Tests on Metallic Specimens"

(4) XIII-MS-103-1, "Opening Radiation Surveillance Capsules and Handling and Storing Specimens"

(5) XIII-MS-104-2, "Shipment of Westinghouse PWR Vessel Material Surveillance Capsule Using SwRI Cask and Equipment"

Copies of the above documents are on file at SwRI.

4.1 Shipment, Opening, and Inspection of Capsule

Southwest Research Institute prepared Procedure XIII-MS-104-2 for the shipment of Capsule X to the SwRI laboratories. SwRI personnel severed the capsule from its extension tube, sectioned the extension tube into several lengths, and supervised the loading of the capsule and extension tube materials into the shipping cask for transport to San Antonio, Texas.

The capsule was opened and the contents identified and stored in accordance with Procedure XIII-MS-103-1. After sawing off the capsule ends, the long seam welds were milled off using a Bridgeport vertical milling machine. The top half of the capsule shell was removed and the specimens and spacer blocks were carefully removed and placed in indexed receptacles identifying each capsule location. After the disassembly had been completed, each specimen was carefully checked to insure agreement with the identification and location as listed in WCAP 8512.[12] No discrepancies were found.

The thermal monitors and neutron dosimeter wires were removed from the holes in the spacers. The thermal monitors, contained in quartz vials, were examined and no melting was observed, thus indicating that the maximum temperature during exposure of Capsule X did not exceed 579°F.

4.2 Neutron Transport and Dosimetry Analysis

As part of the surveillance testing and evaluation program, the neutron transport and dosimetry analysis serves two purposes: (1) to determine the neutron fluence (E > 1.0 MeV) in the surveillance capsule where the metallurgical test specimens are located and (2) to determine the neutron fluence (E > 1.0 MeV) incident on and within the reactor pressure vessel (RPV).

The current methodology for RPV fluence determination is based on combining results of transport calculations with measured dosimeter activities. The transport calculations provide three important sets of data in the overall analysis: (1) spectrum-weighted, effective dosimeter cross sections, (2) lead factors for various locations in the RPV, and (3) fluence rates at locations of interest.

The calculated effective cross sections for different dosimeters are divided into the measured reaction rates in order to obtain the fluence rate (E > 1.0 MeV) at the capsule location. The corresponding fluence rates at various depths into the RPV are obtained by dividing the capsule fluence rate by the appropriate lead factors. Both the effective cross sections and the lead factors depend only on ratios of computed results so that absolute

calculations are not required. The measured dosimeter activities provide the fluence rate normalization. However, absolute fluence rates are calculated to compare with measurements to provide a measure of the uncertainty involved in the RPV fluence determination procedure.

4.2.1 Neutron Transport Analysis

A discrete ordinates calculation using the DOT [13] code was performed to obtain the radial (R) and azimuthal (θ) fluence-rate distribution for the geometry shown in Figure 4. The inclusion of the surveillance capsules in the R-θ model is mandatory to account for the significant perturbation effects from the physical presence of the capsule.

The 47-group energy structure for the SAILOR[14] cross-section library is given in Table 4.1. An S_8 angular structure and a P_3 Legendre cross-section expansion were used in the computations. The fine-group dosimeter cross sections for the $^{63}Cu(n,\alpha)^{60}Co$ reaction were obtained from ENDF/B-V file and were collapsed to 47 groups using a fission plus 1/E weighting spectrum. The other reaction cross sections were taken from the SAILOR cross-section library. The reaction cross sections are given in Table 4.2.

The results of the transport calculations required for the RPV fluence analysis are presented in Tables 4.3 through 4.9. Table 4.3 contains the calculated absolute fluence-rate spectra for the centerline of the surveillance capsules and in Table 4.4 are the calculated saturated activities obtained by folding the results of Tables 4.3 and 4.2 The spectrum-average cross sections, Table 4.5, are obtained from the results of Tables 4.3 and 4.4. Table 4.6 shows that the peak fluence rates at the inner radius, 1/4-T, and 3/4-T locations are at the θ = 45° azimuthal, and Table 4.7 are the group fluxes at the peak location. Table 4.8 shows the radial gradients of the fluence rates (E > 1.0 MeV) through the reactor pressure vessel. The peak

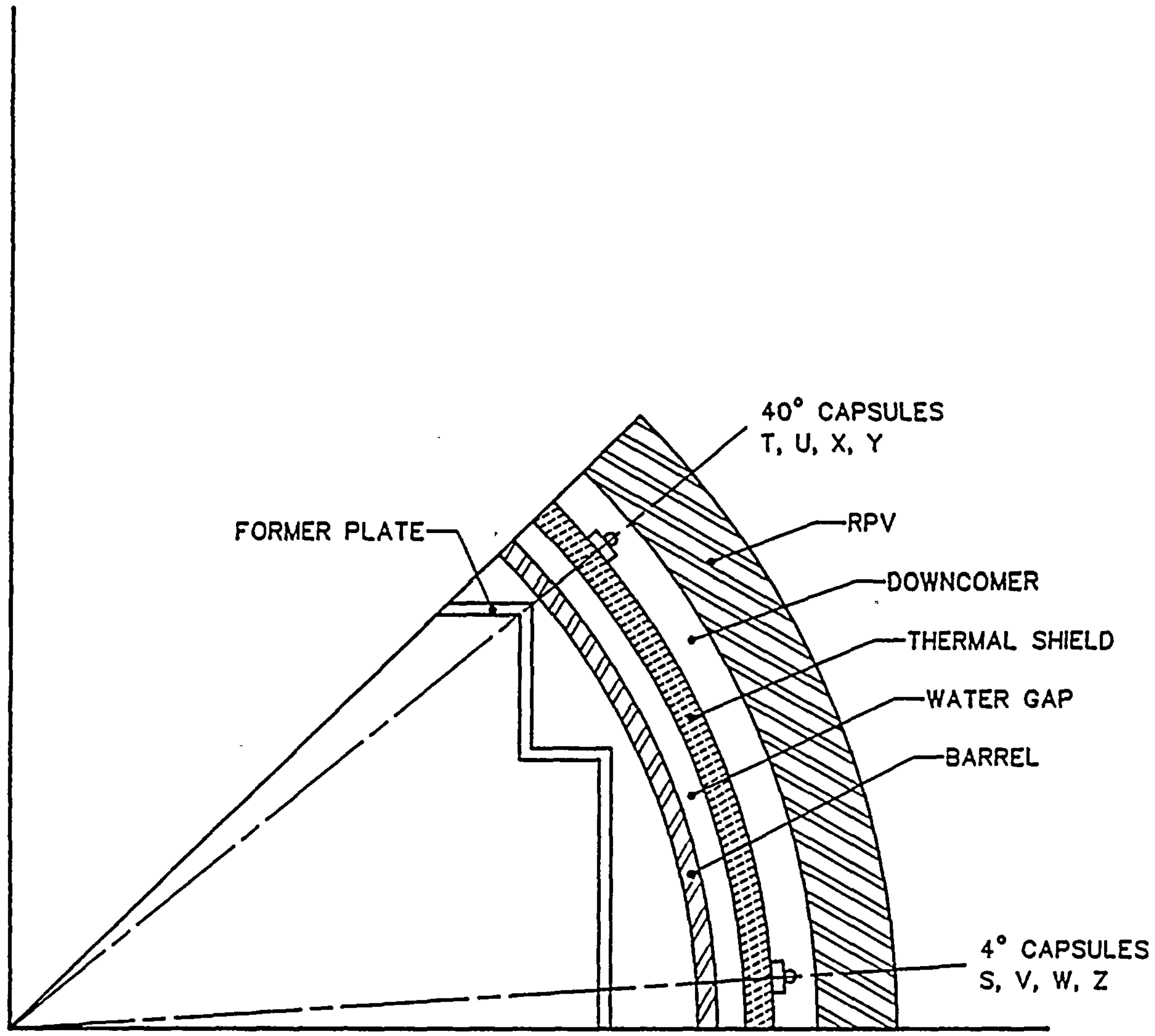


FIGURE 4. R-Θ Geometry for Donald C. Cook Unit 2.

TABLE 4.1

47-GROUP ENERGY STRUCTURE

Group	Lower energy (MeV)	Group	Lower energy (MeV)
1	14.19*	25	0.183
2	12.21	26	0.111
3	10.00	27	0.0674
4	8.61	28	0.0409
5	7.41	29	0.0318
6	6.07	30	0.0261
7	4.97	31	0.0242
8	3.68	32	0.0219
9	3.01	33	0.0150
10	2.73	34	7.10×10^{-3}
11	2.47	35	3.36×10^{-3}
12	2.37	36	1.59×10^{-3}
13	2.35	37	4.54×10^{-4}
14	2.23	38	2.14×10^{-4}
15	1.92	39	1.01×10^{-4}
16	1.65	40	3.73×10^{-5}
17	1.35	41	1.07×10^{-5}
18	1.00	42	5.04×10^{-6}
19	0.821	43	1.86×10^{-6}
20	0.743	44	8.76×10^{-7}
21	0.608	45	4.14×10^{-7}
22	0.498	46	1.00×10^{-7}
23	0.369	47	1.00×10^{-11}
24	0.298		

*The upper energy of Group 1 is 17.33 MeV.

TABLE 4.2

REACTION CROSS SECTIONS (BARNS) USED IN CALCULATIONS
FOR DONALD C. COOK UNIT 2

Group	Energy (MeV)	U-238 (n,f)	Np-237 (n,f)	Fe-54 (n,p)	Ni-58 (n,p)	Cu-63 (n,α)
1	1.733E+01	1.275E+00	2.535E+00	2.686E+01	2.962E-01	3.682E-02
2	1.419E+01	1.086E+00	2.320E+00	4.137E-01	4.416E-01	4.540E-02
3	1.221E+01	9.844E-01	2.334E+00	5.276E-01	6.103E-01	5.357E-02
4	1.000E+01	9.864E-01	2.329E+00	5.781E-01	6.588E-01	3.811E-02
5	8.607E+00	9.891E-01	2.248E+00	5.888E-01	6.553E-01	1.906E-02
6	7.408E+00	8.574E-01	1.965E+00	5.590E-01	6.285E-01	9.277E-03
7	6.065E+00	5.849E-01	1.520E+00	4.697E-01	5.365E-01	2.915E-03
8	4.966E+00	5.615E-01	1.538E+00	3.199E-01	3.917E-01	4.437E-04
9	3.679E+00	5.475E-01	1.638E+00	1.762E-01	2.287E-01	3.568E-05
10	3.012E+00	5.463E-01	1.680E+00	1.155E-01	1.658E-01	5.831E-06
11	2.725E+00	5.527E-01	1.697E+00	7.755E-02	1.131E-01	1.707E-06
12	2.466E+00	5.521E-01	1.695E+00	5.111E-02	9.308E-02	6.834E-07
13	2.365E+00	5.512E-01	1.694E+00	4.756E-02	9.232E-02	4.637E-07
14	2.346E+00	5.504E-01	1.693E+00	4.484E-02	8.614E-02	3.430E-07
15	2.231E+00	5.390E-01	1.677E+00	2.008E-02	4.661E-02	1.150E-07
16	1.920E+00	4.685E-01	1.645E+00	4.771E-03	2.660E-03	1.536E-08
17	1.653E+00	2.706E-01	1.604E+00	6.335E-04	1.337E-02	0
18	1.353E+00	4.502E-02	1.543E+00	1.311E-05	4.438E-03	0
19	1.003E+00	1.102E-02	1.389E+00	0	5.023E-04	0
20	8.208E-01	2.881E-03	1.205E+00	0	1.729E-04	0
21	7.427E-01	1.397E-03	9.845E-01	0	4.914E-05	0
22	6.081E-01	5.378E-04	6.437E-01	0	7.673E-06	0
23	4.979E-01	1.502E-04	2.642E-01	0	8.903E-07	0
24	3.688E-01	8.333E-05	8.800E-02	0	4.070E-08	0
25	2.972E-01	6.168E-05	3.552E-02	0	1.832E-15	0
26	1.832E-01	4.668E-05	2.043E-02	0	0	0
27	1.111E-01	4.015E-05	1.542E-02	0	0	0
28	6.738E-02	4.000E-05	1.228E-02	0	0	0
29	4.087E-02	6.176E-05	1.088E-02	0	0	0
30	3.183E-02	8.610E-05	1.023E-02	0	0	0
31	2.606E-02	8.700E-05	1.002E-02	0	0	0
32	2.418E-02	8.700E-05	9.906E-03	0	0	0
33	2.188E-02	8.700E-05	9.723E-03	0	0	0
34	1.503E-02	5.650E-05	1.004E-02	0	0	0
35	7.102E-03	4.860E-11	6.506E-03	0	0	0
36	3.355E-03	7.439E-10	8.716E-03	0	0	0
37	1.585E-03	4.199E-04	2.303E-02	0	0	0
38	4.540E-04	1.464E-08	3.701E-02	0	0	0
39	2.144E-04	1.044E-08	6.129E-02	0	0	0
40	1.013E-04	1.243E-08	9.027E-02	0	0	0
41	3.727E-05	1.955E-08	2.296E-02	0	0	0
42	1.068E-05	3.086E-08	1.014E-02	0	0	0
43	5.043E-06	4.770E-08	4.011E-03	0	0	0
44	1.855E-06	7.171E-08	9.350E-03	0	0	0
45	8.764E-07	5.067E-08	1.407E-02	0	0	0
46	4.140E-07	1.881E-08	4.328E-03	0	0	0
47	1.000E-07	1.182E-09	8.332E-02	0	0	0

TABLE 4.3

ABSOLUTE CALCULATED NEUTRON FLUENCE RATE SPECTRA [$\phi(E)$] AT THE CENTER OF SURVEILLANCE CAPSULES (SC) FOR DONALD C. COOK UNIT 2

Group	Upper Energy (MeV)	$\phi(E)$ * $n \cdot cm^{-2} \cdot s^{-1}$	
		SC at 40°	SC at 4°
1	1.733E+01	6.93656E+06	5.76403E+06
2	1.419E+01	3.09479E+07	2.51896E+07
3	1.221E+01	1.27275E+08	9.75622E+07
4	1.000E+01	2.59658E+08	1.92220E+08
5	8.607E+00	4.64990E+08	3.27455E+08
6	7.408E+00	1.10830E+09	7.51266E+08
7	6.065E+00	1.59842E+09	1.00403E+09
8	4.966E+00	3.24363E+09	1.79877E+09
9	3.679E+00	2.93332E+09	1.45231E+09
10	3.012E+00	2.36696E+09	1.12970E+09
11	2.725E+00	2.89003E+09	1.33287E+09
12	2.466E+00	1.42825E+09	6.52104E+08
13	2.365E+00	4.42338E+08	1.98677E+08
14	2.346E+00	2.12501E+09	9.45496E+08
15	2.231E+00	5.48432E+09	2.41337E+09
16	1.920E+00	7.12292E+09	2.98454E+09
17	1.653E+00	1.03149E+10	4.21588E+09
18	1.353E+00	2.05020E+10	7.93826E+09
19	1.003E+00	1.54321E+10	5.72833E+09
20	8.208E-01	6.80836E+09	2.54752E+09
21	7.427E-01	2.08115E+10	7.26207E+09
22	6.081E-01	1.90620E+10	6.55344E+09
23	4.979E-01	1.87027E+10	6.48139E+09
24	3.688E-01	1.87067E+10	6.28913E+09
25	2.972E-01	2.59350E+10	8.87760E+09
26	1.832E-01	2.32048E+10	7.80143E+09
27	1.111E-01	1.63390E+10	5.48592E+09
28	6.738E-02	1.52521E+10	5.10511E+09
29	4.087E-02	5.03766E+09	1.69700E+09
30	3.183E-02	1.71555E+09	6.14043E+08
31	2.606E-02	5.79265E+09	1.78767E+09
32	2.418E-02	3.69441E+09	1.19550E+09
33	2.188E-02	8.14806E+09	2.67201E+09

TABLE 4.4

CALCULATED SATURATED ACTIVITIES AT THE CENTER OF SURVEILLANCE CAPSULES FOR DONALD C. COOK UNIT 2

Reaction	Surveillance Capsule at 4° (Bq/g)	Surveillance Capsule at 40° (Bq/g)
$^{54}Fe(n,p)^{54}Mn$	1.535E+6	2.648E+6
$^{58}Ni(n,p)^{58}Co$	2.260E+7	4.054E+7
$^{63}Cu(n,\alpha)^{60}Co$	2.026E+5	2.867E+5
$^{237}Np(n,f)^{137}Cs$	1.119E+7	2.749E+7
$^{238}U(n,f)^{137}Cs$	1.561E+6	3.260E+6

TABLE 4.5

DONALD C. COOK UNIT 2 SPECTRUM-AVERAGED CROSS SECTIONS AT CENTER OF SURVEILLANCE CAPSULES (SC)

Reaction	$\bar{\sigma}$(barns)$^{(1)}$	
	SC at 40°	SC at 4°
$^{54}Fe(n,p)$	0.0678	0.0894
$^{58}Ni(n,p)$	0.0927	0.1174
$^{63}Cu(n,\alpha)$	0.000700	0.00113
$^{237}Np(n,f)$	2.763	2.558
$^{238}U(n,f)$	0.344	0.374
$^{46}Ti(n,p)$		0.0152

(1) $$\bar{\sigma} = \frac{\int_0^{\infty} \sigma(E)\phi(E)dE}{\int_1^{\infty} \phi(E)dE}$$

TABLE 4.6

AZIMUTHAL VARIATION OF $\phi(>1)$ IN RPV OF DONALD C. COOK UNIT 2

		ϕ(E > 1.0 MeV) $n/cm^{-2} \cdot s^{-1}$		
J	$\bar{\theta}$°	0-T R = 219.78	1/4-T R = 225.19	3/4-T R = 236.142
1	1.56	9.480E+09	5.221E+09	1.028E+09
2	3.28	9.169E+09	5.176E+09	1.041E+09
3	4.00	9.025E+09	5.175E+09	1.052E+09
4	4.72	9.486E+09	5.037E+09	1.073E+09
5	5.94	1.015E+10	5.597E+09	1.106E+09
6	8.00	1.085E+10	6.001E+09	1.175E+09
7	10.00	1.150E+10	6.375E+09	1.247E+09
8	12.00	1.217E+10	6.749E+09	1.320E+09
9	14.00	1.286E+10	7.122E+09	1.389E+09
0	16.00	1.350E+10	7.466E+09	1.450E+09
11	18.00	1.402E+10	7.738E+09	1.497E+09
12	20.00	1.432E+10	7.883E+09	1.523E+09
13	21.50	1.427E+10	7.876E+09	1.527E+09
14	22.50	1.418E+10	7.839E+09	1.527E+09
15	23.50	1.408E+10	7.799E+09	1.526E+09
16	24.39	1.401E+10	7.779E+09	1.527E+09
17	25.02	1.399E+10	7.781E+09	1.530E+09
18	25.48	1.399E+10	7.784E+09	1.532E+09
19	26.31	1.399E+10	7.787E+09	1.537E+09
20	27.49	1.408E+10	7.847E+09	1.551E+09
21	28.30	1.424E+10	7.937E+09	1.568E+09
22	28.74	1.434E+10	7.990E+09	1.578E+09
23	29.48	1.449E+10	8.078E+09	1.597E+09
24	30.50	1.482E+10	8.251E+09	1.628E+09
25	31.50	1.522E+10	8.469E+09	1.666E+09
26	32.47	1.568E+10	8.712E+09	1.708E+09
27	33.47	1.620E+10	8.983E+09	1.754E+09
28	34.50	1.678E+10	9.277E+09	1.803E+09
29	35.25	1.722E+10	9.498E+09	1.837E+09
30	35.75	1.751E+10	9.630E+09	1.858E+09
31	36.25	1.778E+10	9.741E+09	1.877E+09
32	36.75	1.800E+10	9.828E+09	1.893E+09
33	37.25	1.815E+10	9.887E+09	1.907E+09
34	37.75	1.822E+10	9.908E+09	1.920E+09
35	38.25	1.817E+10	9.900E+09	1.935E+09
36	38.81	1.804E+10	9.902E+09	1.954E+09
37	39.28	1.776E+10	9.924E+09	1.975E+09
38	39.66	1.766E+10	9.975E+09	1.994E+09
39	40.00	1.779E+10	1.006E+10	2.012E+09
40	40.34	1.802E+10	1.016E+10	2.028E+09
41	40.72	1.852E+10	1.032E+10	2.047E+09
42	41.05	1.899E+10	1.046E+10	2.064E+09
43	41.45	1.955E+10	1.066E+10	2.085E+09
44	41.92	2.008E+10	1.090E+10	2.112E+09
45	42.39	2.047E+10	1.112E+10	2.139E+09
46	42.87	2.075E+10	1.130E+10	2.165E+09
47	43.34	2.097E+10	1.144E+10	2.186E+09
48	43.82	2.112E+10	1.154E+10	2.203E+09
49	44.29	2.121E+10	1.161E+10	2.215E+09
50	44.76	2.125E+10	1.164E+10	2.221E+09

TABLE 4.7

CALCULATED NEUTRON FLUENCE RATE [$\phi(E)$] SPECTRA IN REACTOR PRESSURE VESSEL AT PEAK AXIAL AND AXIMUTHAL LOCATION ($\theta = 45°$) FOR DONALD C. COOK UNIT 2

Group	Upper Energy (MeV)	$\phi(E > 1.0$ MeV) n/cm$^{-2}\cdot$s^{-1}		
		0-T R = 219.78	1/4-T R = 225.19	3/4-T R = 236.142
1	1.733E+01	0.53166E+07	0.22286E+07	0.36063E+06
2	1.419E+01	0.23088E+08	0.97553E+07	0.15732E+07
3	1.221E+01	0.90374E+08	0.36426E+08	0.53124E+07
4	1.000E+01	0.17693E+09	0.70333E+08	0.96453E+07
5	8.607E+00	0.30438E+09	0.11754E+09	0.14818E+08
6	7.408E+00	0.71052E+09	0.26569E+09	0.30518E+08
7	6.065E+00	0.97912E+09	0.35272E+09	0.37525E+08
8	4.966E+00	0.17730E+10	0.64140E+09	0.67721E+08
9	3.679E+00	0.13497E+10	0.53264E+09	0.63806E+08
0	3.012E+00	0.10299E+10	0.43784E+09	0.55198E+08
11	2.725E+00	0.11992E+10	0.53614E+09	0.70522E+08
12	2.466E+00	0.60323E+09	0.27104E+09	0.36044E+08
13	2.365E+00	0.17406E+09	0.84240E+08	0.12500E+08
14	2.346E+00	0.80461E+09	0.40595E+09	0.62522E+08
15	2.231E+00	0.19961E+10	0.10353E+10	0.15980E+09
16	1.920E+00	0.22153E+10	0.13200E+10	0.25036E+09
17	1.653E+00	0.30608E+10	0.19119E+10	0.38146E+09
18	1.353E+00	0.47574E+10	0.36067E+10	0.96084E+09
19	1.003E+00	0.31781E+10	0.27155E+10	0.92694E+09
20	8.208E-01	0.16647E+10	0.11772E+10	0.35203E+09
21	7.427E-01	0.43628E+10	0.46686E+10	0.19763E+10
22	6.081E-01	0.38778E+10	0.40155E+10	0.18109E+10
23	4.979E-01	0.42456E+10	0.45651E+10	0.20894E+10
24	3.688E-01	0.41077E+10	0.53608E+10	0.29320E+10
25	2.972E-01	0.60974E+10	0.61226E+10	0.29813E+10
26	1.832E-01	0.55796E+10	0.62975E+10	0.33266E+10
27	1.111E-01	0.42564E+10	0.41358E+10	0.20823E+10
28	6.738E-02	0.37388E+10	0.33406E+10	0.15865E+10
29	4.087E-02	0.15103E+10	0.89469E+09	0.40075E+09
30	3.183E-02	0.99039E+09	0.28232E+09	0.12523E+09
31	2.606E-02	0.13253E+10	0.18702E+10	0.10917E+10
32	2.418E-02	0.90043E+09	0.11019E+10	0.71618E+09
33	2.188E-02	0.22970E+10	0.20128E+10	0.11316E+10

TABLE 4.8

RADIAL GRADIENT OF FAST FLUENCE RATE [$\phi(E>1)$] THROUGH RPV, AT PEAK AZIMUTHAL AND AXIAL LOCATIONS IN DONALD C. COOK UNIT 2

$\bar{R}^{(1)}$ (cm)	$\phi(E>1)$ $\frac{n}{cm^2\text{-}s}$
219.978	2.109E+10
221.14	1.922E+10
222.92	1.572E+10
224.70	1.239E+10
226.48	9.649E+9
228.26	7.452E+9
230.04	5.721E+9
231.82	4.369E+9
233.60	3.316E+9
235.39	2.494E+9
237.17	1.849E+9
238.95	1.331E+9
240.73	8.723E+9

(1) RPV liner begins at $\bar{R}$ = 219.71 cm.
RPV begins at 220.25 and ends at 241.62 cm.
1/4-T = 225.19 cm.
3/4-T = 236.14 cm.

TABLE 4.9

CALCULATED FLUENCE RATES AND LEAD FACTORS IN DONALD C. COOK UNIT 2

Location	Radius (cm)	Fluence Rate $[n/(cm^{-2} \cdot s^{-1})]$	Lead Factors	
			4° Capsule	40° Capsule
Capsules ID				
S, V, W, Z (4°)	211.41	2.746E+10	-	-
T, U, X, Y (40°)	211.41	6.245E+10	-	-
Vessel ID	219.71	2.125E+10	1.29	2.94
Vessel 1/4-T	225.19	1.164E+10	2.36	5.37
Vessel 3/4-T	236.14	2.221E+9	12.36	28.12

fluence rates at the inner radius, 1/4-T, and 3/4-T locations in Table 4.9 are obtained from Table 4.8 by interpolation (or extrapolation). The capsule fluence rates and the lead factors are also summarized in Table 4.9.

4.4.2 Neutron Dosimeter Testing and Analysis

The gamma activities of the dosimeters were determined in accordance with Procedure XI-MS-101-0 using an IT-5400 multi-channel analyzer and a Ge(Li) coaxial detector system. The calibration of the equipment was accomplished with ^{54}Mn, ^{60}Co, and ^{137}Cs radioactivity standards obtained from the U.S. Department of Commerce National Bureau of Standards. The dosimeter wires were weighed on a Mettler-Type H6T balance. All activities were corrected to the time-of-removal (TOR) at reactor shutdown.

The references for the procedures used in processing the dosimeters are:

ASTM E181-82, "Detector Calibration and Analysis Radionuclides"

ASTM E261-77, "Determining Neutron flux, Fluence, and Spectra Radioactive Techniques"

ASTM E262-85, "Determining Thermal Neutron Flux by Radioactive Techniques

ASTM-E263-82, "Determining Fast Neutron Flux by Radioactivation of Iron"

ASTM E264-82, "Determining Fast Neutron Flux by Radioactivation of Nickel"

ASTM E523-82, "Measuring Fast Neutron Flux Density of Radioactivation of Copper

ASTM E704-84, "Determining Fast Neutron Flux Density by Radioactivation of Uranium-238"

ASTM E705-84, "Determining Fast Neutron Flux Density by Radioactivation of Neptunium-237"

The results of the neutron dosimetry analysis procedure are summarized in Tables 4.10 to 4.16. The equations and definitions used for neutron dosimetry analysis are summarized in table 4.10. The neutron

TABLE 4.10

EQUATIONS AND DEFINITIONS FOR NEUTRON DOSIMETRY ANALYSIS

Equations

$$A_{TOR} = N_o Y \int_0^\infty \sigma(E)\phi(E)dE \sum_{j=1}^{J} P_j(1-e^{-\lambda T_j})e^{-\lambda(T-t_j)} \quad (4.1)$$

where A_{TOR} = product nuclide activity at end of irradiation, Bq/mg;
$\sigma(E)$ = energy-dependent activation cross section (cm^2) for dosimeter m,
$\phi(E)$ = energy-dependent fluence rate at surveillance location;
Y = product nuclide per reaction (fission yield);
λ = decay constant of the product nuclide (d^{-1});
P_j = fraction of full power during operating period j;
T_j = length of time for irradiation interval j;
T = time from beginning of irradiation to time of removal;
t_j = elapsed time from beginning of irradiation to end of interval j;
N_o = number of target atoms per mg in dosimeter; and
J = number of irradiation intervals.

$$A_{SAT} = \int_0^\infty \sigma(E)\phi(E)dE \quad (4.2)$$

where A_{SAT} = reaction rate per target nucleus.

$$\bar{\sigma}_{E_t} = \frac{\int_0^\infty \sigma(E)\phi(E)dE}{\int_{E_t}^\infty \phi(E)dE} = \frac{A_{SAT}}{\phi(E>E_t)} \quad (4.3)$$

where $\bar{\sigma}_{E_t}$ = effective spectrum-averaged cross section and

$\int_{E_t}^\infty \phi(E)dE$ = fluence rate for neutrons with energies greater than E_tMeV[$\phi(E>E_t)$].

Substituting Eq. (2.2) into Eq. (2.1) and solving for A_{SAT}, one obtains

$$A_{SAT} = \frac{A_{TOR}}{N_o Y \sum_{j=1}^{J} P_j(1-e^{-\lambda T_j})e^{-\lambda(T-t_j)}} \quad . \quad (4.4)$$

Replacing A_{SAT} in Eq. (2.4) by A_{SAT} in Eq. (2.3), one obtains

$$\phi(E>E_t) = \frac{A_{TOR}}{N_o Y \bar{\sigma}_{E_t} \sum_{j=1}^{J} P_j(1-e^{-\lambda T_j})e^{-\lambda(T-t_j)}} \quad . \quad (4.5)$$

The total fluence is then given by

$$\Phi(E>E_t) = \phi(E>E_t) \sum_{j=1}^{J} P_j T_j \quad . \quad (4.6)$$

The thermal neutron fluence rate (ϕ_{th}) is determined from the bare and cadmium-covered cobalt activities using Eq. (2.7) below.

$$\phi_{th} = \frac{A_B - A_{Cd}}{N_o \sigma_o \sum_{j=1}^{J} P_j(1-e^{-\lambda T_j})e^{-\lambda(T-t_j)}} \quad , \quad (4.7)$$

where A_B = bare cobalt activity (dps/mg),
A_{Cd} = cadmium-covered cobalt activity (dps/mg),
N_o = number of cobalt-59 atoms per mg of cobalt, and
σ_o = 37.1 barns.

Definitions

The lead factor (LF)* is defined as follows

$$LF \stackrel{d}{=} \frac{\text{neutron fluence rate } (E>E_t) \text{ at the capsule center}}{\text{maximum neutron fluence rate at the PV inner radius}} \quad .$$

The saturation factor (SF) is given by

$$SF = \frac{1}{\sum_{j=1}^{J} P_j(1-e^{-\lambda T_j})e^{-\lambda(T-t_j)}} \quad .$$

*A more general definition can be stated by replacing the denominator by the maximum neutron fluence rate at any point in the pressure vessel (PV).

TABLE 4.11

CONSTANTS FOR PROCESSING DOSIMETRY DATA

Reaction	N_0 (atoms/mg)	Half-Life	λ (day^{-1})	X-ray Branching Intensity	Fission Yield (%)	Atom Fraction	Atomic Weight
$^{46}Ti(n,p)^{46}Sc$	1.018×10^{18}	83.85 d	8.261×10^{-3}	0.9998 @ 889 keV 0.9999 @ 1120 keV	–	0.081	47.90
$^{54}Fe(n,p)^{54}Mn$	6.254×10^{17}	312.50 d	2.218×10^{-3}	0.9997 @ 835 keV	–	0.058	55.847
$^{58}Ni(n,p)^{58}Co$	7.004×10^{18}	70.85 d	9.783×10^{-3}	0.9944 @ 811 keV	–	0.6827	58.70
$^{59}Co(n,\gamma)^{60}Co$	1.022×10^{19}	5.271 y	3.600×10^{-4}	0.9990 @ 1173 keV 0.9998 @ 1332 keV	–	1.0000	58.9332
$^{63}Cu(n,\alpha)^{60}Co$	6.555×10^{18}	5.271 y	3.600×10^{-4}	0.9990 @ 1173 keV 0.9998 @ 1332 keV	–	0.6917	63.546
$^{237}Np(n,f)^{137}Cs$	2.540×10^{18}	30.17 y	6.290×10^{-5}	0.8530 @ 662 keV	6.267	1.0000	237.0482
$^{238}U(n,f)^{137}Cs$	2.530×10^{18}	30.17 y	6.290×10^{-5}	0.8530 @ 662 keV	6.000	1.0000	238.0508

TABLE 4.12

REACTOR POWER-TIME HISTORY FOR DONALD C. COOK UNIT 2 CAPSULE X

Time Step	Operating Period	Fraction of Full Power* P_j	Irradiation Interval T_j	Decay Time $T-t_j$
1	3/78	0.2437	10	2891
2	4/78	0.1544	30	2861
3	5/78	0.2594	31	2830
4	6/78	0.6382	30	2800
5	7/78	0.4396	31	2769
6	8/78	0.6066	31	2738
7	9/78	0.8531	30	2708
8	10/78	0.8825	31	2677
9	11/78	0.4808	30	2647
10	12/78	0.9257	31	2616
11	1/79	0.9257	31	2585
12	2/79	0.9257	28	2557
13	3/79	0.9257	31	2526
14	4/79	0.9142	30	2496
15	5/79	0.5835	31	2465
16	6/79	0.0000	30	2435
17	7/79	0.9033	31	2404
18	8/79	0.9656	31	2373
19	9/79	0.9656	30	2343
20	10/79	0.5918	31	2312
21	11/79	0.0000	30	2282
22	12/79	0.0000	31	2251
23	1/80	0.4447	31	2220
24	2/80	0.9191	29	2191
25	3/80	0.9191	31	2160
26	4/80	0.9191	30	2130
27	5/80	0.9191	31	2099
28	6/80	0.8272	30	2069
29	7/80	0.5926	31	2038
30	8/80	0.9669	31	2007
31	9/80	0.9669	30	1977
32	10/80	0.5614	31	1946
33	11/80	0.0000	30	1916
34	12/80	0.5979	31	1885
35	1/81	0.9782	31	1854
36	2/81	0.9782	28	1826
37	3/81	0.4418	31	1795
38	4/81	0.0000	30	1765
39	5/81	0.3525	31	1734
40	6/81	0.7806	30	1704
41	7/81	0.7201	31	1673

TABLE 4.12 (Continued)

REACTOR POWER-TIME HISTORY FOR DONALD C. COOK UNIT 2 CAPSULE X

Time Step	Operating Period	Fraction of Full Power* P_j	Irradiation Interval T_j	Decay Time $T-t_j$
42	8/81	0.9516	31	1642
43	9/81	0.9516	30	1612
44	10/81	0.1343	31	1581
45	11/81	0.9612	30	1551
46	12/81	0.9612	31	1520
47	1/82	0.9612	31	1489
48	2/82	0.9612	28	1461
49	3/82	0.4028	31	1430
50	4/82	0.9569	30	1400
51	5/82	0.9569	31	1369
52	6/82	0.9569	30	1339
53	7/82	0.9569	31	1308
54	8/82	0.4115	31	1277
55	9/82	0.9076	30	1247
56	10/82	0.9215	31	1216
57	11/82	0.6669	30	1186
58	12/82	0.0000	31	1155
59	1/83	0.1217	31	1124
60	2/83	0.9748	28	1096
61	3/83	0.9989	31	1065
62	4/83	0.9930	30	1035
63	5/83	0.9692	31	1004
64	6/83	0.7712	30	974
65	7/83	0.6673	31	943
66	8/83	0.9157	31	912
67	9/83	0.9172	30	882
68	10/83	0.4815	31	851
69	11/83	0.1659	30	821
70	12/83	0.9397	31	790
71	1/84	0.9623	31	759
72	2/84	0.9410	29	730
73	3/84	0.3054	31	699
74	4/84	0.0000	30	669
75	5/84	0.0000	31	638
76	6/84	0.0000	30	608
77	7/84	0.5424	31	577
78	8/84	0.9200	31	546
79	9/84	0.9430	30	516
80	10/84	0.9575	31	485
81	11/84	0.8472	30	455
82	12/84	0.4321	31	424

TABLE 4.12 (Continued)

REACTOR POWER-TIME HISTORY FOR DONALD C. COOK UNIT 2 CAPSULE X

Time Step	Operating Period	Fraction of Full Power* P_j	Irradiation Interval T_j	Decay Time $T-t_j$
83	1/85	0.5208	31	393
84	2/85	0.9916	28	365
85	3/85	0.9764	31	334
86	4/85	0.9924	30	304
87	5/85	0.9986	31	273
88	6/85	0.9985	30	243
89	7/85	0.4295	31	212
90	8/85	0.0237	31	181
91	9/85	0.0000	30	151
92	10/85	0.0641	31	120
93	11/85	0.5437	30	90
94	12/85	0.7942	31	59
95	1/86	0.8000	31	28
96	2/86	0.5997	28	0

*Full power level for Cook Unit 2 is 3391 MWt. Time of removal is referenced to 2/28/86, 2400 hr.

TABLE 4.13

CORRECTION FACTORS TO OBTAIN MEASURED SATURATED ACTIVITIES
AT CAPSULE X CENTERLINE

Reaction	Saturation Factor	Gradient Factor	Impurity Factor*
$^{54}Fe(n,p)^{54}Mn$	1.631	1.051	1.0
$^{58}Ni(n,p)^{58}Co$	1.720	1.164	1.0
$^{63}Cu(n,\alpha)^{60}Co$	2.340	0.9538	1.0
$^{237}Np(n,f)^{137}Cs$	9.037	1.0	1.0
$^{238}U(n,f)^{137}Cs$	9.037	1.0	1.0
$^{39}Co(n,\gamma)^{60}Co$	2.340	1.164	1.0

*Impurities were assumed negligible.

TABLE 4.14

CALCULATED SATURATED MIDPLANE ACTIVITIES IN DONALD C. COOK UNIT 2 SURVEILLANCE CAPSULES

Dosimeter or Flux	Saturated Activities for 40° Surveillance Capsule, Bq/g			Saturated Activities for 4° Surveillance Capsule, Bq/g		
	R=210.41 cm	R=211.41 cm	R=212.41 cm	R=210.41 cm	R=211.41 cm	R=212.41 cm
$^{54}Fe(n,p)^{54}Mn$	3.240E+06	2.648E+06	2.170E+06	1.856E+06	1.535E+06	1.275E+06
$^{58}Ni(n,p)^{58}Co$	4.953E+07	4.054E+07	3.313E+07	2.732E+07	2.260E+07	1.847E+07
$^{63}Cu(n,\alpha)^{60}Co$	3.471E+05	2.867E+05	2.390E+05	2.428E+05	2.026E+05	1.704E+05
$^{237}Np(n,f)^{137}Cs$	3.279E+07	2.749E+07	2.234E+07	1.332E+07	1.119E+07	9.241E+06
$^{238}U(n,f)^{137}Cs$	3.963E+06	3.260E+06	2.640E+06	1.880E+06	1.561E+06	1.286E+06
$^{46}Ti(n,p)^{46}Sc$	7.872E+05	6:454E+05	5.337E+05	5.114E+05	4.240E+05	3.545E+05
$\phi(E > 1.0\ MeV)$	7.544E+10	6.245E+10	5.048E+10	3.297E+10	2.746E+10	2.258E+10
$\phi(E > 0.1\ MeV)$	2.506E+11	2.111E+11	1.717E+11	9.354E+10	7.901E+10	6.521E+10

TABLE 4.15

COMPARISON OF MEASURED AND CALCULATED SATURATED ACTIVITIES FOR FAST THRESHOLD DETECTORS

Reaction ID	Radial Location (cm)	Time of Removal Activity, A_{TOR} (Bq/mg)	Measured Saturated Activity, A^E_{SAT} (Bq/mg)	Calculated Saturated Activity, A^C_{SAT} (Bq/mg)	Calculated (C) Divided by Measures (E) Activity (Bq/mg)
$^{54}Fe(n,p)^{54}Mn$					
Top	211.68	1.375E+3			
Top-middle	211.68	1.407E+3			
Middle	211.68	1.399E+3			
Bottom-middle	211.68	1.423E+3			
Bottom	211.68	1.367E+3			
Average		1.394 ± 0.023E+3	2.390E+3	2.648E+3	1.108
$^{63}Cu(n,\alpha)^{60}Co$					
Top-middle	211.18	1.197E+2			
Middle	211.18	1.202E+2			
Bottom-middle	211.18	1.216E+2			
Average		1.205 ± 0.010E+2	2.689E+2	2.867E+2	1.066
$^{58}Ni(n,p)^{58}Co$					
Top-middle	212.18	1.837E+4			
Middle	212.18	1.808E+4			
Bottom-middle	212.18	1.840E+4			
Average		1.828 ± 0.018E+4	3.660E+4	4.054E+4	1.108
$^{237}Np(n,f)^{137}Cs$					
Middle	211.41	3.142E+3	2.839E+4	2.749E+4	0.9683
$^{238}U(n,f)^{137}Cs$					
Middle	211.41	3.763E+2	3.400E+3	3.260E+3	0.9588

TABLE 4.16

THERMAL NEUTRON FLUENCE RATE IN CAPSULE X

Axial Location	Saturated Activity (Bq/mg)		Thermal Fluence Rate [n/(cm^{-2}·s^{-1})]
	Bare	Cadmium-Covered	
Top Co	3.448E+07	1.445E+07	5.283E+10
Bottom Co	3.402E+07	1.445E+07*	5.161E+10
Average			5.222E+10

*Assumed to be same as top value.

dosimeters and the constants used in processing the dosimeters are given in Table 4.11. The reactor power-time history data given in Table 4.12 are used to calculate the saturation factors (see definition, Table 4.10) shown in Table 4.13. In Table 4.13, the gradient correction factors are obtained from the transport calculations given in Table 4.14 and the impurity correction factors are assumed to be negligible. Each of the measured activities A_{TOR}, Table 4.15 are multiplied by the three appropriate correction factors in Table 4.13 to obtain the measured saturated activities A_{SAT}, for comparison with the calculated values. The results (Table 4.15) indicate that the calculated values are +11% to -4% from the measured values. The thermal neutron fluence rates are given in Table 4.16 and are obtained using Eq. (4.7) from Table 4.10. These values were too low to cause any significant burnin or burnout corrections.

4.2.3 Results of Neutron Transport and Dosimetry Analysis

The comparison of the calculated and the derived fluence rates in Table 4.17 indicates very good agreement: 6.019×10^{10} from the measurements and 6.245×10^{10} from the calculations. The derived fluence rate from the measurements is used to determine the fluences shown in Table 4.18.

The assembly-wise source distribution for Donald C. Cook Unit 2 Capsule X analysis is provided in Appendix A. The three-dimensional (3-D) flux synthesis method used in this report is given in Appendix B.

4.3 Mechanical Property Tests

The irradiated Charpy V-notch specimens were tested on a calibrated* SATEC Model SI-1K 240 ft-lb, 16 ft/sec impact machine in accordance with Procedure XI-MS-104-1. The test temperatures, selected to develop the ductile-brittle transition and upper shelf regions, were obtained using a liquid conditioning

TABLE 4.17

COMPARISON OF FAST NEUTRON FLUENCE RATES FROM TRANSPORT CALCULATIONS AND DOSIMETRY MEASUREMENTS FOR CAPSULE X

Reaction	Measured Saturated Activity (Bq/mg)	Fluence Rate Derived from Measurements $[n/(cm^{-2} \cdot s^{-1})]$	Calculated Fluence Rate $[n/(cm^{-2} \cdot s^{-1})]$	Calculated Divided by Derived Fluence Rate
$^{54}Fe(n,p)^{54}Mn$	2.390E+03	5.637E+10	6.245E+10	1.108
$^{63}Cu(n,\alpha)^{60}Co$	2.689E+02	5.860E+10	6.245E+10	1.066
$^{58}Ni(n,p)^{58}Co$	3.660E+04	5.637E+10	6.245E+10	1.108
$^{237}Np(n,f)^{137}Cs$	2.839E+04	6.452E+10	6.245E+10	0.9679
$^{238}U(n,f)^{137}Cs$	3.400E+03	6.511E+10	6.245E+10	0.9591
Average		6.019 ± 0.432E+10	6.245E+10	1.042 ± 0.074

TABLE 4.18

CALCULATED PEAK FLUENCES IN PRESSURE VESSEL BASED ON CAPSULE X DOSIMETRY

Location	5.273 EFPY Fluence $(n \cdot cm^{-2})$	10 EFPY Fluence $(n \cdot cm^{-2})$	15 EFPY Fluence $(n \cdot cm^{-2})$	32 EFPY Fluence $(n \cdot cm^{-2})$
Surveillance Capsule*	1.002E+19	1.899E+19	2.849E+19	6.078E+19
Pressure Vessel IR	3.406E+18	6.460E+18	9.690E+18	2.067E+19
Pressure Vessel 1/4-T	1.865E+18	3.538E+18	5.306E+18	1.132E+19
Pressure Vessel 3/4-T	3.562E+17	6.753E+17	1.013E+18	2.161E+18

*Based on averaged fluence rate derived from dosimetry measurements.

both monitored with a Fluke Model 2168A digital thermometer. The Charpy V-notch impact data obtained by SwRI on the specimens contained in Capsule X are presented in Tables 4.19 through 4.22. The shifts in the Charpy V-notch transition temperatures determined for the vessel plate, the weld metal and the HAZ materials are shown in Figures 5 through 8. The Capsule T and Y results are included for comparison.

A summary of the shifts in RT_{NDT} determined at the 30 ft-lb level as specified in Appendix G to 10 CFR 50 [1], and the reduction in C_V upper shelf energies for each material, is presented in Table 4.23.

Tensile tests were carried out in accordance with Procedure XI-MS-103-1 using a 22-kip capacity MTS Model 810 Material Test System equipped with an Instron Catalogue No. G-51-13A 2-in. strain gage extensometer and Hewlett Packard Model 7004B X-Y autographic recording equipment. Tensile tests on the plate material and the weld metal were run at 250°F and 550°F at a strain rate of 0.005 in/in/min. through the 0.2% offset yield strength using servocontrol and ramp generator. The results, along with tensile data reported by Westinghouse on the unirradiated materials [12], are presented in Table 4.24. The load-strain records are included in Appendix C.

Testing of the WOL specimens was deferred at the request of Indiana & Michigan Electric Company. The specimens are in storage at the SwRI radiation laboratory.

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* Inspected and calibrated using specimens and procedures obtained from the Army Materials and Mechanics Research Center.

TABLE 4.19

CHARPY IMPACT PROPERTIES OF LONGITUDINAL PLATE
DONALD C. COOK UNIT 2
CAPSULE X

Southwest Research Institute
Department of Materials Sciences

CHARPY TEST DATA SHEET

MATERIAL - LONGITUDINAL

Project No. 06-8888-001
Date 4/28/87

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH ___X
ML-25	RT-71	17.0	.017	5	
ML-26	+100	28.5	.026	5	
ML-32	+125	30.5	.026	15	
ML-27	+150	40.0	.037	30	
ML-31	+175	70.0	.061	45	
ML-28	+200	83.5	.072	90	
ML-29	+250	99.0	.085	100	
ML-30	+300	107.0	.085	100	

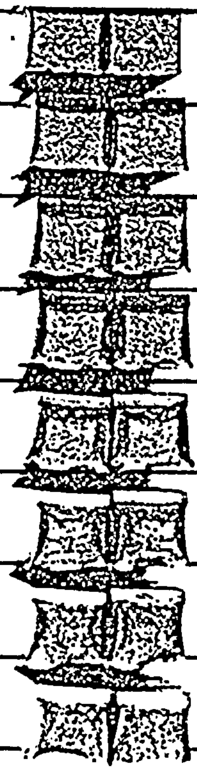


TABLE 4.20

CHARPY IMPACT PROPERTIES OF TRANSVERSE PLATE
DONALD C. COOK UNIT NO. 2
CAPSULE X

Southwest Research Institute
Department of Materials Sciences

CHARPY TEST DATA SHEET

MATERIAL - TRANSVERSE

Project No. 06-8888-001
Date 4/28/87

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH X
MT-48	+ 50	8.0	.007	0	
MT-37	RT-71	14.5	.013	0	
MT-38	+100	23.0	.022	15	
MT-46	+100	20.5	.019	10	
MT-47	+125	24.5	.024	10	
MT-39	+150	30.0	.029	20	
MT-40	+200	50.0	.048	30	
MT-45	+200	53.0	.050	50	
MT-44	+225	60.0	.055	30	
MT-41	+250	68.[illegible]	.061	90	
MT-42	+250	[illegible]	[illegible]	100	
MT-43	+300	[illegible]	[illegible]	[illegible]	

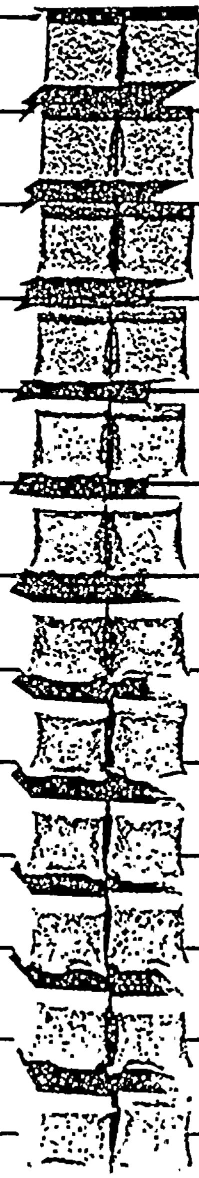


TABLE 4.21

CHARPY IMPACT PROPERTIES OF HAZ MATERIAL
DONALD C. COOK UNIT 2
CAPSULE X

Southwest Research Institute
Department of Materials Sciences

CHARPY TEST DATA SHEET

MATERIAL - HAZ

Project No. 06-8888-001
Date 4/28/87

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH X
MH-43	- 25	25.0	.018	10	
MH-47	+ 50	48.5	.039	45	
MH-37	RT+71	41.5	.036	40	
MH-45	+100	64.5	.054	60	
MH-38	+100	95.0	.068	70	
MH-48	+125	117.0	.082	100	
MH-42	+150	97.0	.067	80	
MH-41	+200	100.0	.081	100	
MH-40	+200	71.0	.061	100	
MH-46	+225	110.0	.076	100	
MH-44	+250	119.0	.083	100	
MH-39	+300	103.0	.080	100	

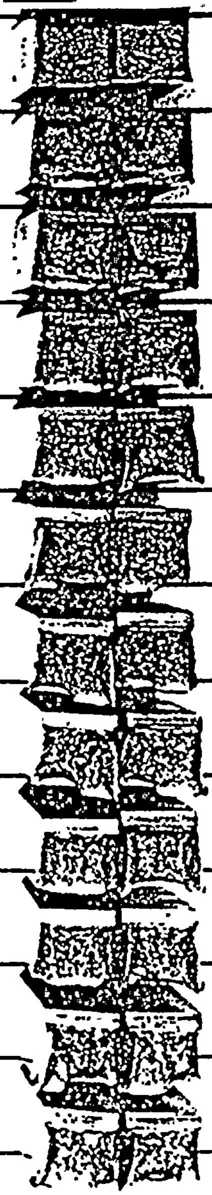


TABLE 4.22

CHARPY IMPACT PROPERTIES OF WELD METAL
DONALD C. COOK UNIT 2
CAPSULE X

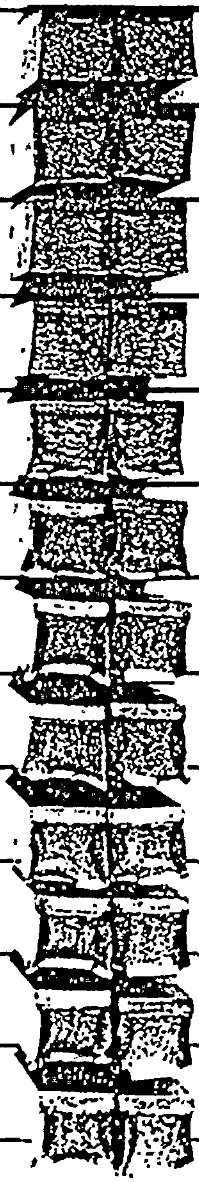
Southwest Research Institute
Department of Materials Sciences

CHARPY TEST DATA SHEET

MATERIAL - WELD

Project No. 06-8888-001
Date 4/28/87

SPECIMEN NO.	TEMP °F	ENERGY FT-LBS	LATERAL EXPANSION	FRACTURE APPEARANCE	PHOTOGRAPH X
MW-47	- 25	24.5	.022	10	
MW-48	0	16.0	.018	5	
MW-45	+ 50	19.5	.017	10	
MW-37	RT+71	24.0	.020	15	
MW-38	+100	27.0	.030	25	
MW-46	+125	61.5	.057	45	
MW-40	+150	70.5	.064	100	
MW-39	+200	75.5	.069	100	
MW-43	+200	61.0	.058	85	
MW-42	+250	64.0	.061	100	
MW-41	+250	66.0	.057	100	
MW-44	+300	68.5	.069	100	



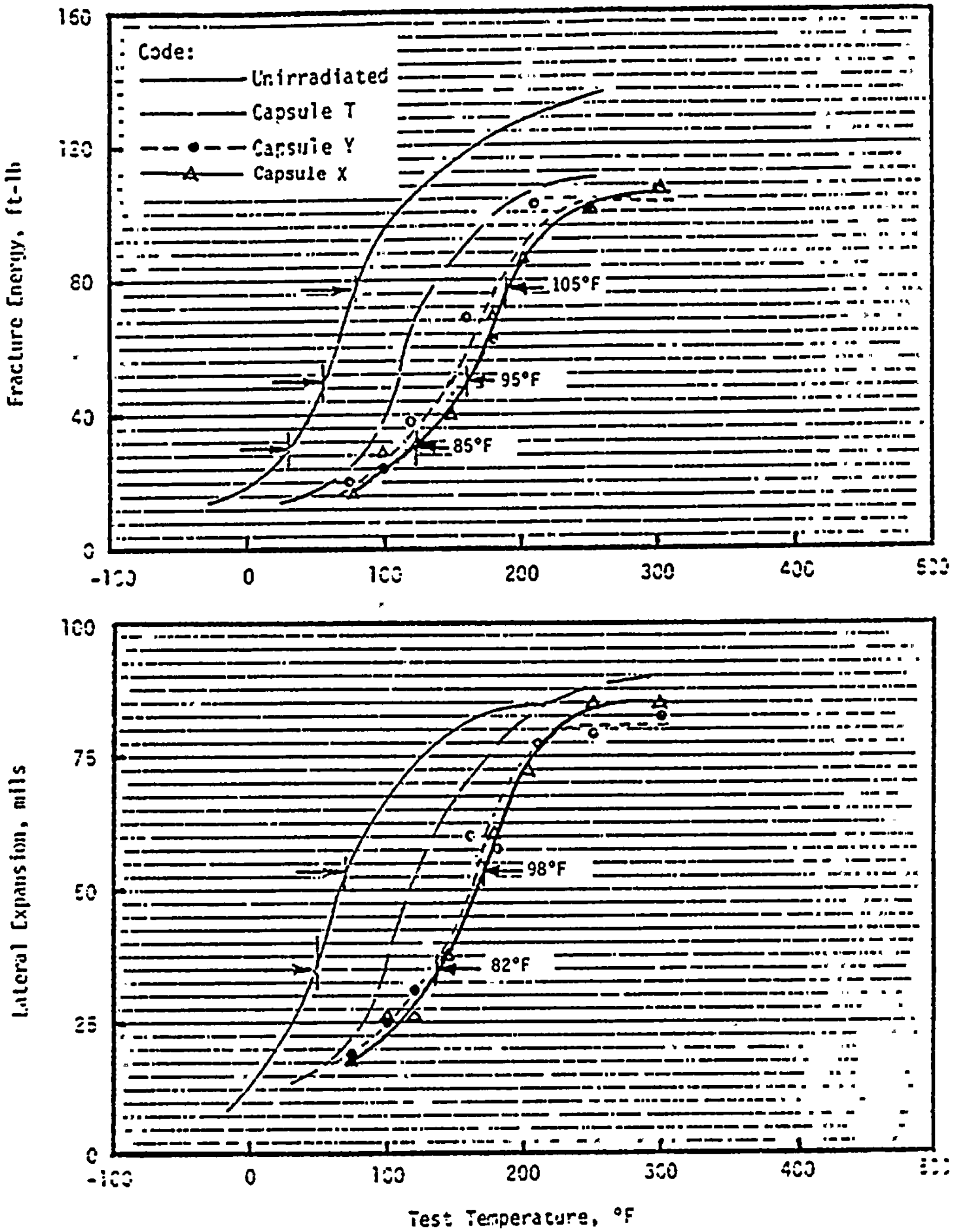


FIGURE 5. RADIATION RESPONSE OF DONALD C. COOK UNIT NO. 2 VESSEL SHELL PLATE C5521-2 (LONGITUDINAL ORIENTATION)

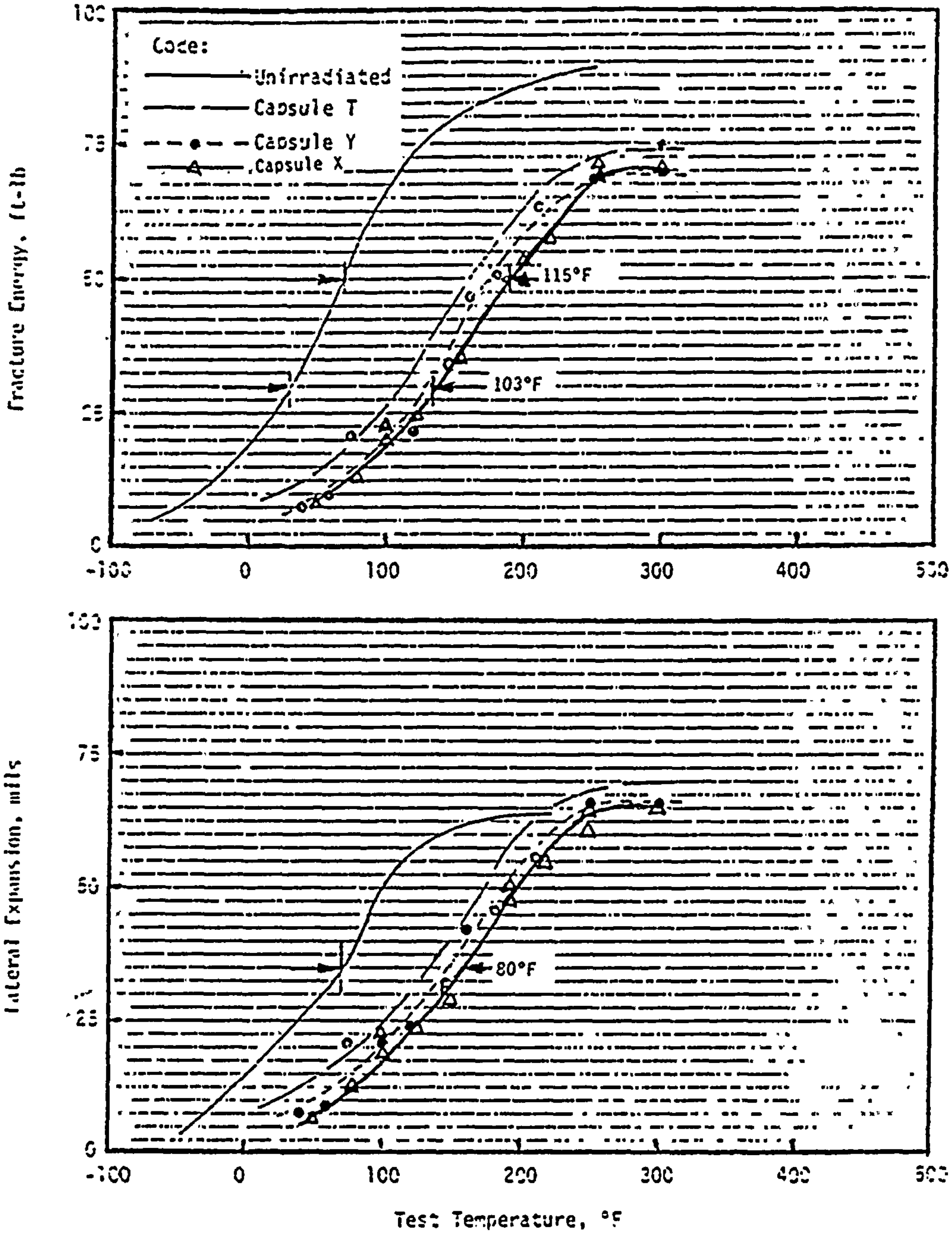


FIGURE 6. RADIATION RESPONSE OF DONALD C. COOK UNIT NO. 2 VESSEL SHELL PLATE C5521-2 (TRANSVERSE ORIENTATION)

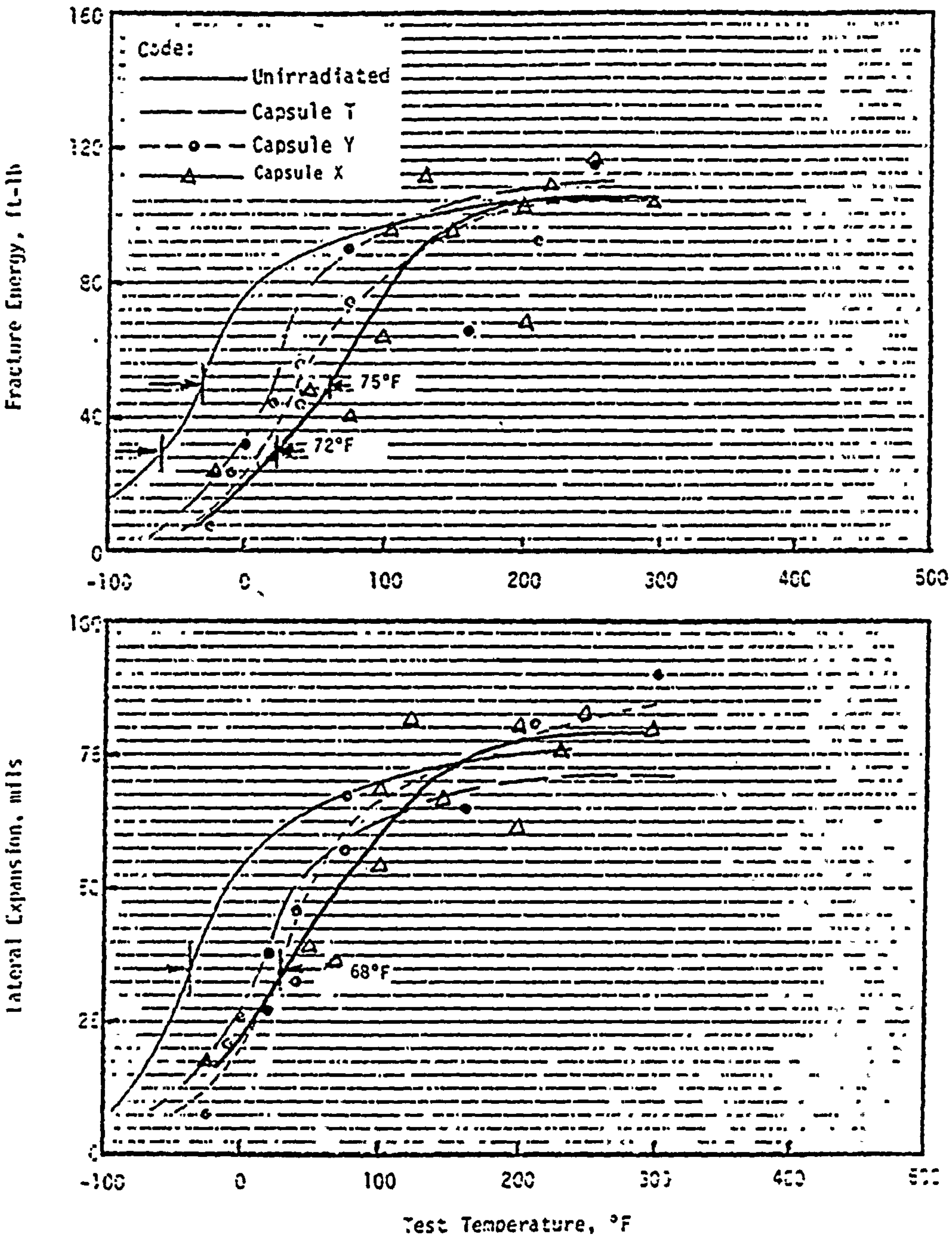


FIGURE 7. RADIATION RESPONSE OF DONALD C. COOK UNIT NO. 2 REACTOR VESSEL HEAT-AFFECTED ZONE MATERIAL

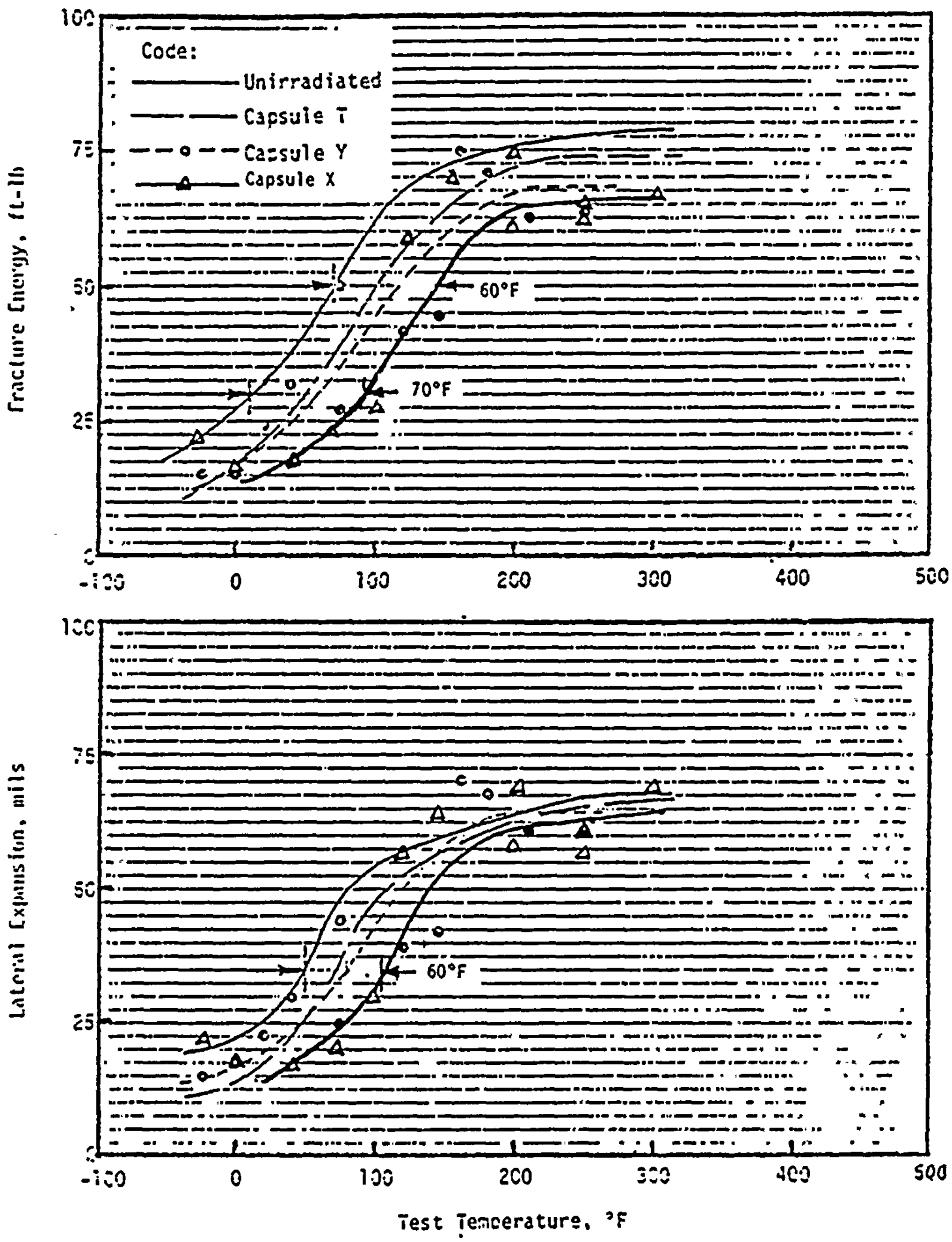


FIGURE 8. RADIATION RESPONSE OF DONALD C. COOK UNIT NO. 2 REACTOR VESSEL WELD MATERIAL

TABLE 4.23

EFFECT OF IRRADIATION ON CAPSULE X SURVEILLANCE MATERIALS
DONALD C. COOK UNIT NO. 2

Criterion[1]	Weld Metal[2]	HAZ Material[2]	Trans. Plate C5521-2[3]	Long Plate C5521-2[3,5]
Transition Temperature Shift				
@ 50 ft-lb	60°F	75°F	115°F	105°F
@ 30 ft-lb	70°F	72°F	103°F	95°F
@ 35 mil	60°F	68°F	80°F	98°F
RT_{NDT}[4]	70°F	72°F	103°F	95°F
Cv Upper Shelf Drop	11 ft-lb (15%)	46 ft-lb (38%)	23 ft-lb (27%)	42 ft-lb (33%)

(1) Refer to Figures 4-7.
(2) Fluence = 8.53×10^{18} n/cm^2, E > 1 MeV.
(3) Fluence = 1.05×10^{19} n/cm^2, E > 1 MeV.
(4) Transition temperature shift at 30 ft-lb (46 ft-lb for longitudinal plate).
(5) Transition temperatures at 77 ft-lb, and 54 mils [17].

TABLE 4.24

TENSILE PROPERTIES OF SURVEILLANCE MATERIALS
DONALD C. COOK UNIT NO. 2

Condition	Test Material	Spec. No.	Temp. (°F)	0.2% YS (ksi)	UTS (ksi)	Fracture Load (lb)	Fracture Stress (ksi)	Uniform Elongation (%)	Total Elongation (%)	R.A. (%)
Capsule X[a]	Plate C5521-2 (Transverse)	MT-8	250	76.0	93.9	3588	156.0	15.0	18.7	52.8
		MT-7	550	72.1	92.3	3672	163.9	14.8	17.3	54.0
	Weld Metal	MW-8	210	79.9	94.5	3112	183.1	13.9	21.4	65.3
		MW-7	550	73.7	92.5	3148	166.6	11.4	18.8	61.4
(b)	Plate C5521-2 (Transverse)	-	Room	67.4	87.3	3200	161.2	13.4	23.4	59.6
		-	Room	65.4	85.9	2950	156.4	15.0	27.1	61.7
		-	300	58.8	78.6	2650	146.1	13.0	22.6	63.1
		-	300	60.5	79.5	2675	157.6	10.6	19.8	65.4
		-	550	57.5	83.0	3225	142.1	11.5	19.0	53.8
		-	553	58.9	83.1	3150	145.6	12.7	20.5	56.0
	Weld Metal	-	Room	75.7	93.2	2850	173.4	13.9	25.7	66.8
		-	Room	76.9	91.3	2950	178.8	12.2	22.6	66.6
		-	300	70.7	88.0	2900	171.0	10.7	20.7	66.0
		-	300	71.0	85.3	2875	179.0	10.3	21.2	67.5
		-	550	70.0	87.2	3160	157.2	10.1	19.2	59.6
		-	550	68.2	87.8	3050	166.0	9.3	20.2	62.8

(a) Fluence = 1.002×10^{19} m/cm^2, E > 1 MeV.
(b) Unirradiated [12].

5.0 ANALYSIS OF RESULTS

The analysis of data obtained from surveillance program specimens has the following goals:

(1) Estimate the period of time over which the properties of the vessel beltline materials will meet the fracture toughness requirements of Appendix G of 10CFR50. This requires a projection of the measured reduction in C_v upper shelf energy to the vessel wall using knowledge of the energy and spatial distribution of the neutron flux and the dependence of C_v upper shelf energy on the neutron fluence.

(2) Develop heatup and cooldown curves to describe the operational limitations for selected periods of time. This requires a projection of the measured shift in RT_{NDT} to the vessel wall using knowledge of the dependence of the shift in RT_{NDT} on the neutron fluence and the energy and spatial distribution of the neutron flux.

The energy and spatial distribution of the neutron flux for Donald C. Cook Unit No. 2 was calculated for Capsule X with a discrete ordinates transport Code. This analysis, predicted that the lead factor (ratio of fast flux at the capsule location to the maximum pressure vessel flux) was 2.94 at the capsule centerline, 3.09 for the core-side Charpy layer, and 2.50 for the vessel-side Charpy layer (see Table 4.9). This analysis also predicted that the fast flux at the 1/4T and 3/4T positions in the 8.5-in. pressure vessel wall would be 55% and 11% respectively of that at the vessel I.D.

A method for estimating the increase in RT_{NDT} as a function of neutron fluence and chemistry is given in Regulatory Guide 1.99, Revision 1 [8]. However, the Guide also permits interpolation between credible surveillance data and extrapolation by extending the response curves parallel to the Guide trend curves. The data from Capsules T, Y and X are deemed to be credible because (1) the surveillance materials are judged to be controlling with regard to radiation damage, (2) the scatter in the transverse plate and weld metal Charpy data is small, and (3) the changes in yield strength are consistent with the Charpy curve shifts. Except for the longitudinal plate material, the slopes of the response curves constructed in Figure 9 are less than the square root of fluence utilized in Regulatory Guide 1.99. Although recent work [7] indicates that the square root of fluence dependence may be too high, the projected responses of the Donald C. Cook Unit No. 2 vessel beltline materials are based on the trend curves of Figure 9 which were constructed in accordance with Regulatory Guide 1.99 procedures.

The Donald C. Cook Unit No. 2 vessel plate surveillance material is more sensitive than the weld metal and HAZ surveillance materials to irradiation embrittlement. Since the unirradiated values of RT_{NDT} for the intermediate shell plate C5521-2 is higher than those of the weld and HAZ materials [16], the beltline region plate material is projected to control the adjusted value of RT_{NDT} through the 32 EFPY design life of Donald C. Cook Unit No. 2. A summary of the projected values of RT_{NDT} for 12 and 32 EFPY of operation of Donald C. Cook Unit No. 2, is presented in Table 5.1.

A method for estimating the reduction in C_v upper shelf energy as a function of neutron fluence is also given in Regulatory Guide 1.99, Revision 1 [8]. The results from Capsules T [16], Y [17], and X are compared to a portion of Figure 2 of the Regulatory Guide 1.99, Revision 1, in Figure 10. Although the shelf energy response of the weld surveillance material from Capsules X fall below them, the predictive trend curves of Regulatory Guide 1.99, Revision 1, will be used in this analysis for conservatism. Response curves have been drawn through the HAZ Transverse Plate and Longitudinal plate

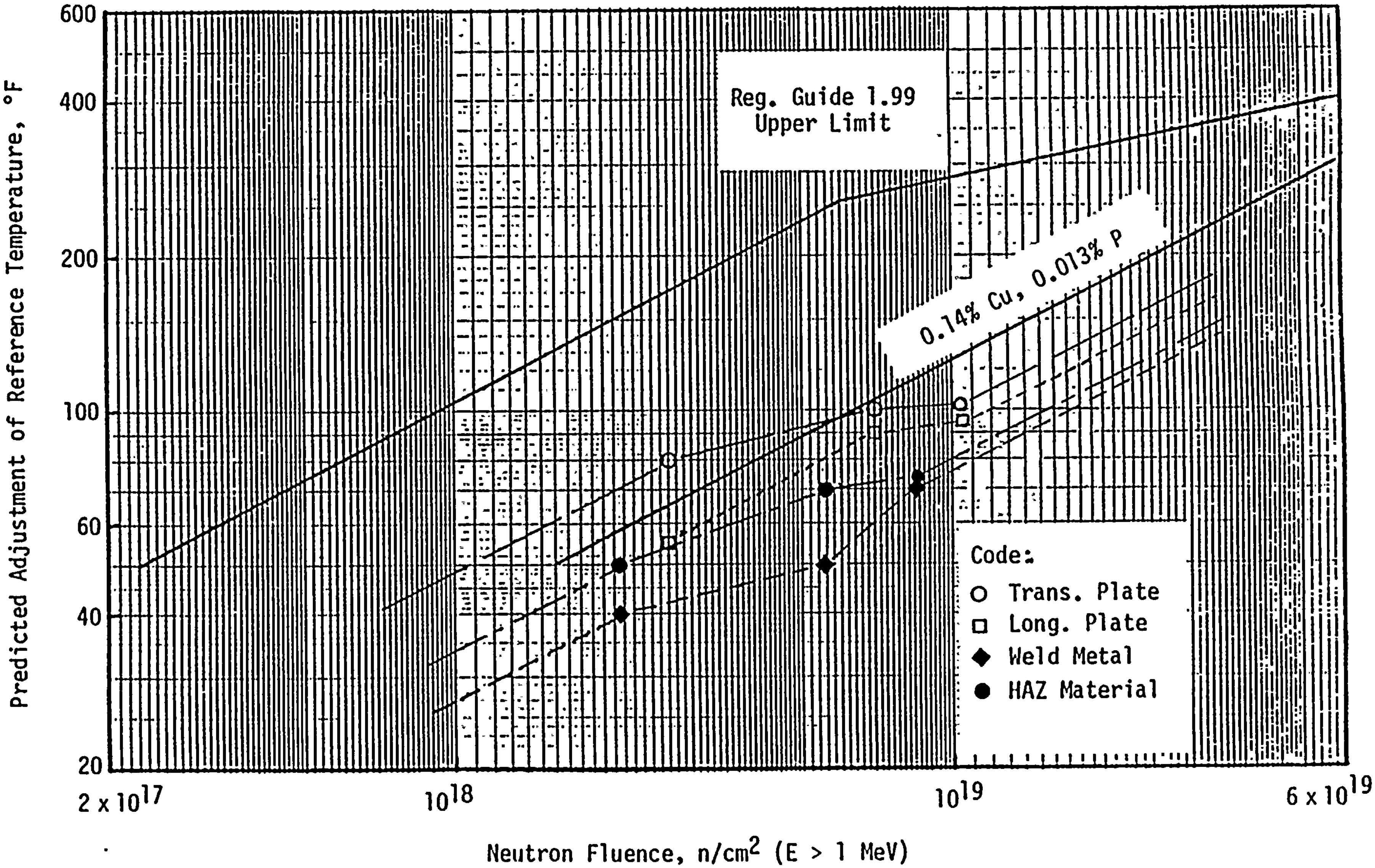


FIGURE 9. EFFECT OF NEUTRON FLUENCE ON RT_{NDT} SHIFT, DONALD C. COOK UNIT NO. 2

TABLE 5.1

PROJECTED VALUES OF RT_{NDT} FOR DONALD C. COOK UNIT NO. 2

EFPY	P.V. Material	Location	ΔRT_{NDT}	Fluence[(a)]	ΔRT_{NDT}	Adj. RT_{NDT}
12	Plate C5521-2	I.D.	58°F[(b)]	7.8×10^{18}	101	159
		1/4T	58°F	4.3×10^{18}	88	146
		3/4T	58°F	8.1×10^{17}	44	102
	HAZ Material	I.D.	20°F	7.8×10^{18}	74	94
		1.4T	20°F	4.3×10^{18}	63	83
		3/4T	20°F	8.1×10^{17}	31	51
	Weld Metal	I.D.	0°F[(c)]	7.8×10^{18}	66	66
		1/4T	0°F	4.3×10^{18}	47	47
		3/4T	0°F	8.1×10^{17}	23	23
32	Plate C5521-2	I.D.	58°F[(b)]	2.1×10^{19}	140	198
		1/4T	58°F	1.1×10^{19}	105	163
		3/4T	58°F	2.2×10^{18}	72	130
	HAZ Material	I.D.	20°F[(b)]	2.1×10^{19}	113	133
		1/4T	20°F	1.1×10^{19}	84	104
		3/4T	20°F	2.2×10^{18}	50	70
	Weld Metal	I.D.	0°F[(c)]	2.1×10^{19}	108	108
		1/4T	0°F	1.1×10^{19}	80	80
		3/4T	0°F	2.2×10^{18}	40	40

(a) Neutrons/cm^2, E > 1 MeV.
(b) Reference 16.
(c) Estimated per Reference 18

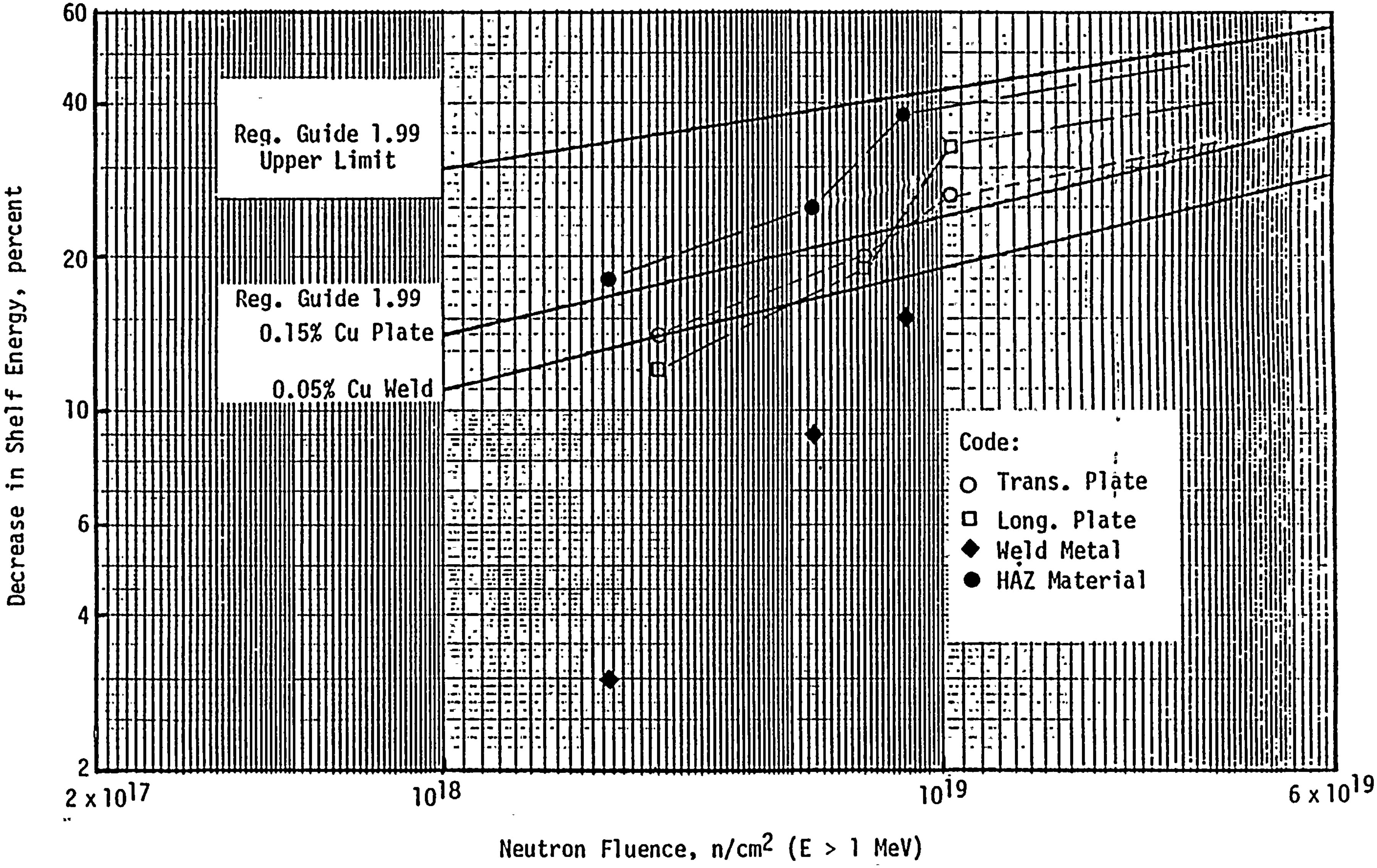


FIGURE 10. DEPENDENCE OF C_V UPPER SHELF ENERGY ON NEUTRON FLUENCE, DONALD C. COOK UNIT NO. 2

data since these results fall above the plate trend curve.

Referring to the conservative trend curves for 0.05% Cu weld metal and the HAZ and plate response curves, the projected C_V shelf energies of the vessel materials are as follows:

- <u>Plate C5521-2 (Unirradiated C_V Shelf = 86 ft-lb)</u>

  32 EFPY at I.D. -- 60 ft-lb (30% reduction)

  32 EFPY at 1/4T -- 63 ft-lb (27% reduction)

  32 EFPY at 3/4T -- 71 ft-lb (17% reduction)

Note: For shelf energies below the 0.15% Cu plate curve the conservative plate curve is used.

- <u>Weld Metal (Unirradiated C_V Shelf = 75 ft-lb)</u>

  32 EFPY at I.D. -- 58 ft-lb (23% reduction)

  32 EFPY at 1/4T -- 60 ft-lb (20% reduction)

  32 EFPY at 3/4T -- 65 ft-lb (13% reduction)

- <u>HAZ Material (Unirradiated C_V Shelf = 122 ft-lb)</u>

  32 EFPY at I.D. -- 68 ft-lb (44% reduction)

  32 EFPY at 1/4T -- 73 ft-lb (40% reduction)

  32 EFPY at 3/4T -- 100 ft-lb (18% reduction)

These projections indicate that the core beltline materials in the Donald C. Cook Unit No. 2 pressure vessel material will retain adequate shelf toughness throughout the 32 EFPY design lifetime.

The current Donald C. Cook Unit No. 2 reactor vessel surveillance program removal schedule, revised to conform to ASTM 185-79 [9], is summarized in Table 5.2. There are five capsules remaining in the vessel, of which three are standbys.

TABLE 5.2

REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE [16]
DONALD C. COOK UNIT NO. 2

Capsule	WOL Material	Removal Time	Equivalent Vessel Fluence
T	Weld Metal	1.08 EFPY[(a)]	3.4 EFPY at I.D.
Y	Weld Metal	3.24 EFPY[(b)]	11 EFPY at I.D.
X	Trans. Plate	5.27 EFPY[(c)]	E.O.L. at 1/4T
U	Weld Metal	9 EFPY	E.O.L. at I.D.
S	Trans. Plate	32 EFPY	E.O.L. at I.D.
V	Trans. Plate	Standby	-
W	Trans. Plate	Standby	-
Z	Weld Metal	Standby	-

(a) Removed after core cycle 1.
(b) Removed after core cycle 3.
(c) Removed after core cycle 5.

6.0 HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION OF DONALD C. COOK UNIT NO. 2

Donald C. Cook Unit No. 1 is a 3391 Mw_t pressurized water reactor operated by Indiana and Michigan Electric Company. The unit has bee provided with a reactor vessel material surveillance program as required by 10CFR50, Appendix H.

The third surveillance capsule (Capsule X) was removed during the 1986 refuelling outage. This capsule was tested as described in earlier sections of this report. In summary, these test results indicate that:

(1) The RT_{NDT} of the surveillance plate material in Capsule X increased 103°F as a result of exposure to a neutron fluence of 1.002×10^{19} neutrons/cm^2 (E > 1 MeV).

(2) Based on an analysis of the dosimeters in Capsule X, the vessel wall fluence at the I.D. was 3.406×10^{18} neutrons/cm^2 (E > 1 MeV) at the time of its removal.

(3) The maximum RT_{NDT} after 12 effective full power years (EFPY) of operation was predicted to be 146°F at the 1/4T and 102°F at the 3/4T vessel wall locations, as controlled by the core beltline shell plate. These projections are comparable to those resulting from the evaluation of the data from capsule Y.

(4) The maximum RT_{NDT} after 32 EFPY of operation was predicted to be 163°F at the 1/4T and 130°F at the 3/4T vessel wall locations, as controlled by the core beltline shell plate. These predictions are lower than that predicted from Capsule Y analysis.

The Unit No. 2 heatup and cooldown limit curves for 12 EFPY and 32 EFPY have been computed on the bases of (3) and (4) above. The following pressure vessel contents were employed as input data in this analysis:

Vessel Inner Radius, r_i	= 86.50 in., including cladding
Vessel Outer Radius, r_o	= 95.2 in.
Operating Pressure, P_o	= 2235 psig
Initial Temperature, T_o	= 70°F
Final Temperature, T_f	= 550°F
Effective Coolant Flow Rate, Q	= 134.6×10^6 lb/hr
Effective Flow Area, A	= 26.72 ft^2
Effective Hydraulic Diameter, D	= 15.05 in.

The SwRI computer program calculates the allowable pressure over the temperature range 70°F - 550°F such that the reference stress intensity factor, K_{IR}, is always greater than the sum of twice K_{Ip} (pressure induced) and K_{It} (thermal gradient induced) as dictated by Appendix G of the Code [2]. The current version of the SwRI program incorporates the physical property data specified by Appendix I of the Code through the 1982 Summer Adenda. The changes in thermal conductivity code allowables made in the early 1980's reduced the calculated allowable pressure at coolant temperatures below about 200°F from that obtained when using the previously specified values.

Heatup curves were computed for a heatup rate of 100°F/hr. Since lower rates tend to raise the curve in the central region, these curves apply to all heating rates up to 100°F/hr. Cooldown curves were computed for cooldown rates of 0°F/hr (steady state), 20°F, 40°F/hr, 60°F/hr, and 100°F/hr. The 20°F/hr curve would apply to cooldown rates up to 20°F/hr; the 40°F/hr curve would apply to rates up to 40°F/hr; the 60°F/hr curve would apply to rates up to 60°F/hr; the 100°F/hr curve would apply to rates up to

100°F/hr.

The unit No. 2 heatup and cooldown curves developed for up to 12 EFPY after Capsule Y is identical to the Capsule X data. It is recommended that the current technical specification for 12 EFPY not be changed. These curves are reproduced in Figures 11 and 12. The limit curves developed in the Capsule Y report for 32 EFPY is conservative compared to the data generated here for Capsule X. These curves are reproduced in Figures 13 and 14.

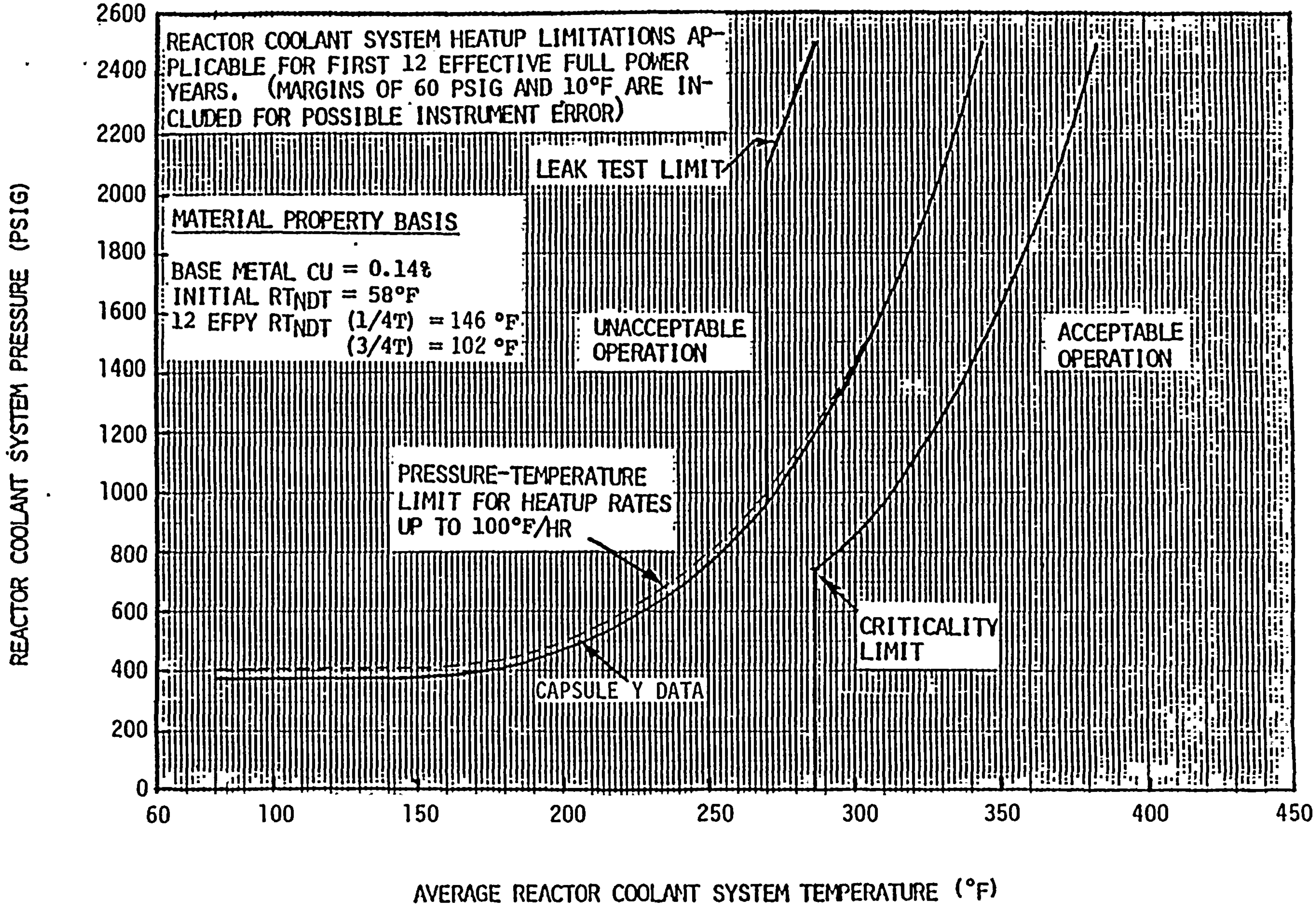


FIGURE 11. REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS 100°F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT, 12 EFPY

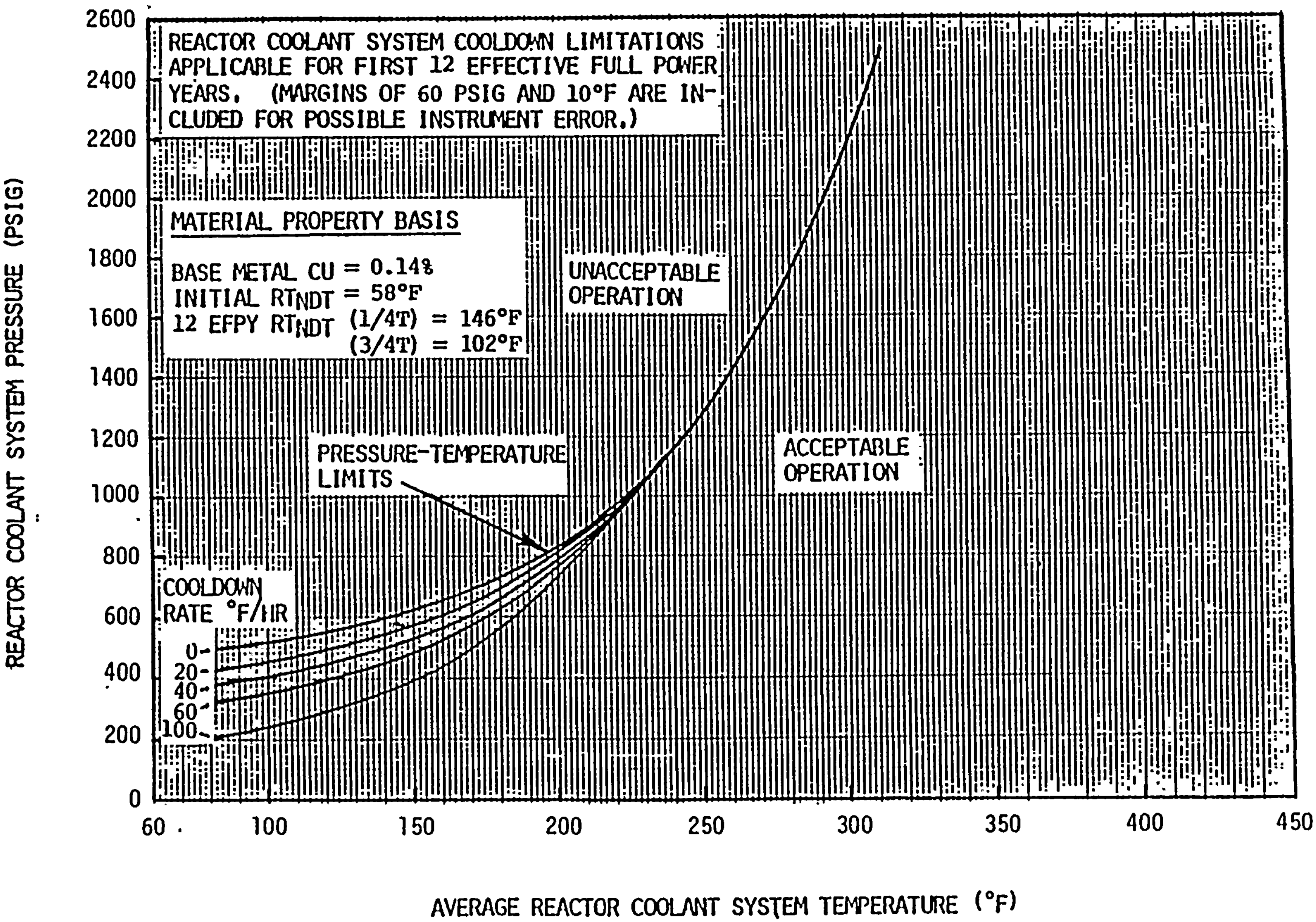


FIGURE 12. REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS COOLDOWN RATES, 12 EFPY

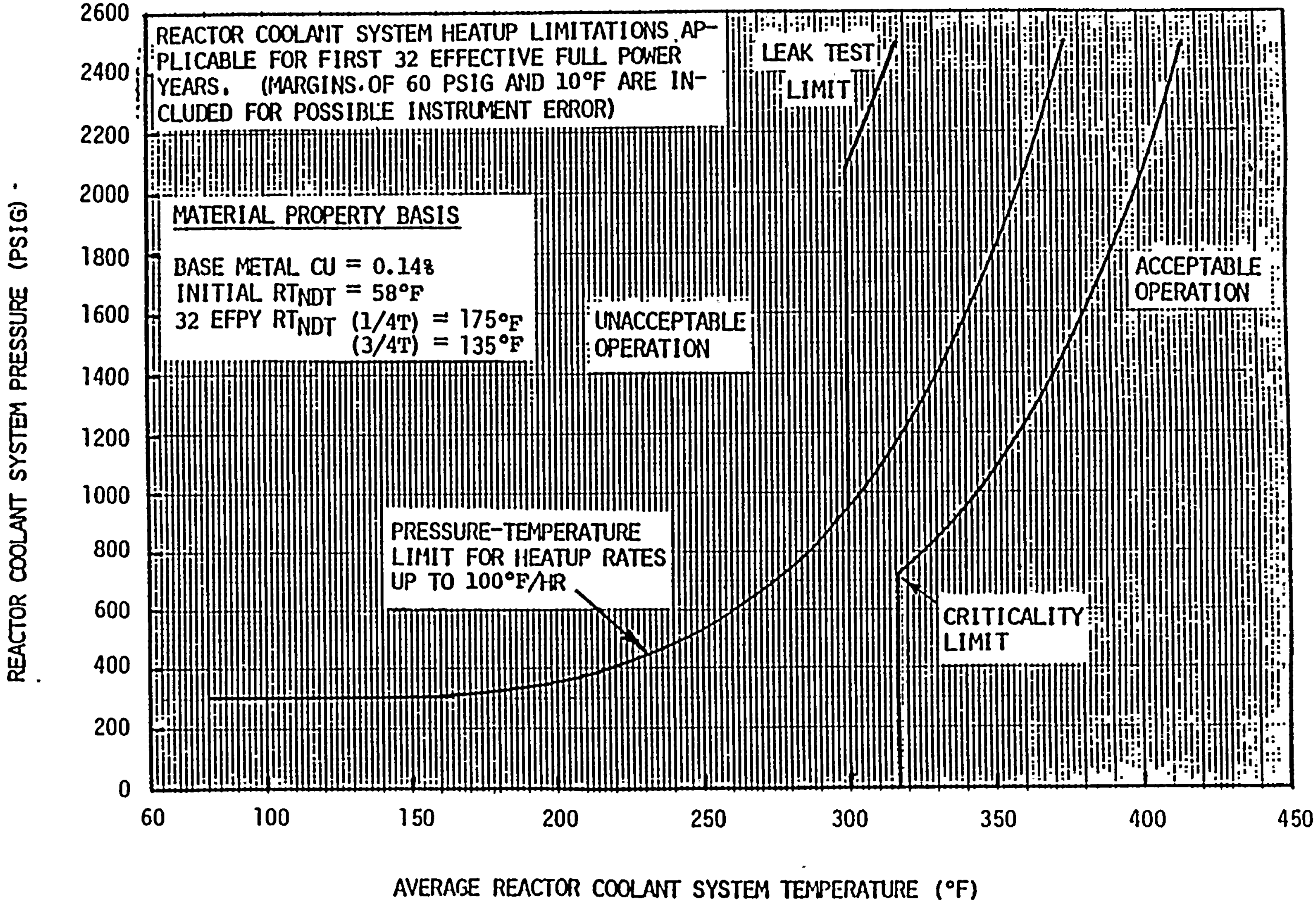


FIGURE 13. REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS 100°F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT, 32 EFPY (Ref. 17)

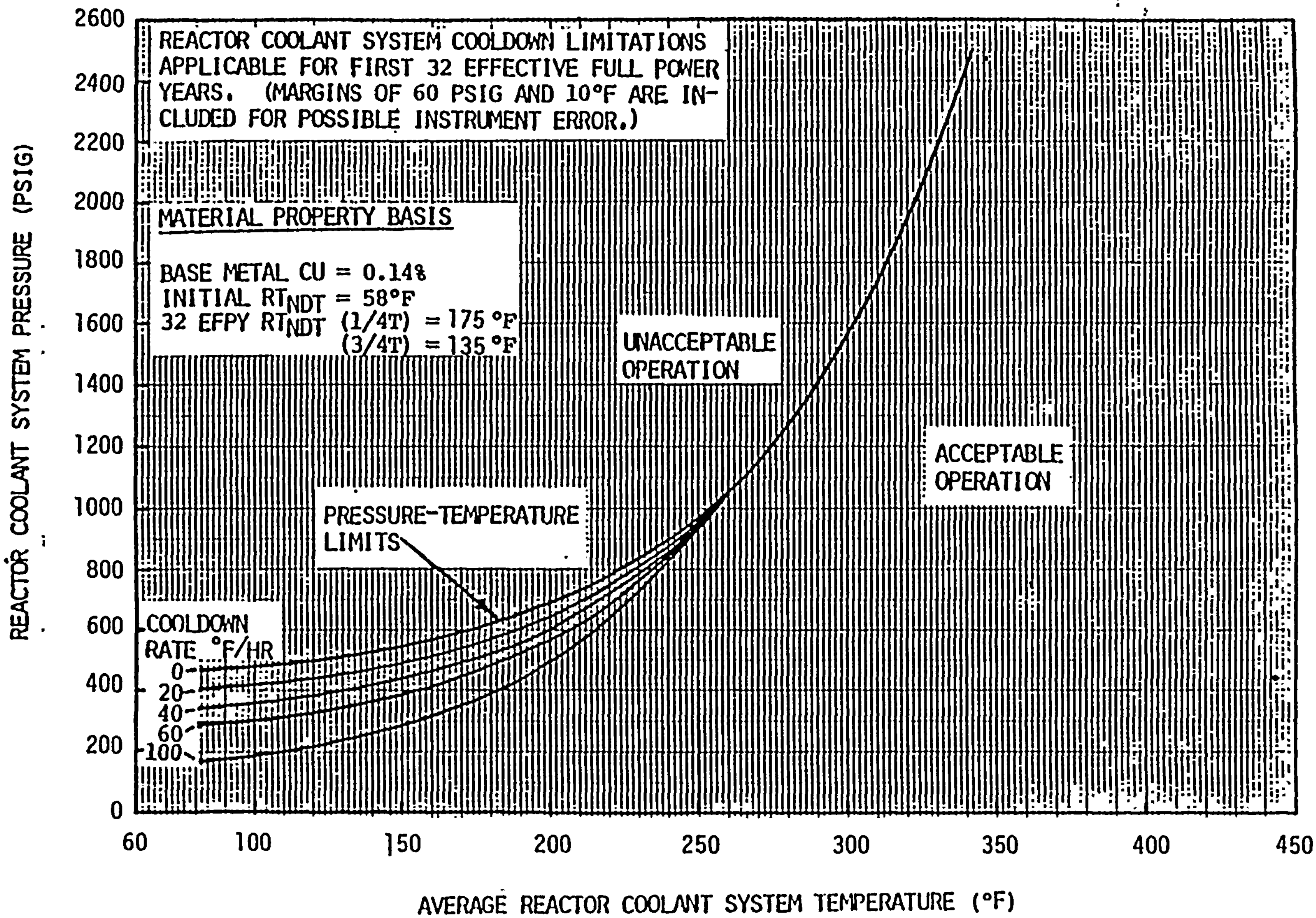


FIGURE 14. REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS COOLDOWN RATES, 32 EFPY (Ref. 17)

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APPENDIX A

Determination of Assembly-Wise Source Distribution for
Donald C. Cook Unit 2, Capsule X Analysis

Appendix A

DETERMINATION OF ASSEMBLY-WISE SOURCE DISTRIBUTION FOR DONALD C. COOK UNIT 2, CAPSULE X ANALYSIS

Surveillance capsule X was in the reactor for cycles 1-5. Table A.1 shows the cycle-average relative assembly-wise power distribution for each of these five cycles. These values were obtained by averaging BOC, MOC, and EOC power distributions provided for each cycle. The resulting assembly-wise relative power distribution shown in the last column of Table A.1 formed the basis of the space-dependent source used in the transport calculations. The relative power values shown in this table were multiplied by a value of 17.6 MW_{th} per assembly to obtain the absolute power produced by each assembly. Table A.2 shows the final absolute power produced by each assembly. Table A.2 shows the final absolute assembly-wise power distribution for a quarter core model (note that some assemblies appear as fractions in the quarter core, which reduces their absolute power produced). The absolute power values are converted to a neutron source by multiplying by the conversion factor of 8.163×10^{16} neutrons/s per MW. A pin-wise intra-assembly distribution was used to represent the spatial power variation within each of the peripheral assemblies, while a flat distribution is used for interior assemblies. The relative pin-power distribution was provided by the Donald C. Cook Unit 2 support staff. The normalized, space-dependent source distribution is then transformed to the DOT RΘ mesh by using a computer program which performs the necessary interpolation and renormalization calculations. The output of this source routine, which includes a listing of the final DOT RΘ spatial source distribution, is included. The source energy distribution corresponds to an ENDF/B-V Watt fission spectrum.

Table A.1. Cycle-Average Assembly Relative Power Distribution for Donald C. Cook Unit 2

	CYCLE ——>					
Zone	1	2	3	4	5	Average
1**	1.146	0.861	0.854	0.850	1.013	0.945
2*	1.188	1.037	1.060	0.962	1.139	1.077
3*	1.151	0.968	1.117	0.987	1.183	1.081
4*	1.205	1.135	1.206	1.038	1.250	1.165
5*	1.117	0.988	1.113	0.982	1.171	1.074
6*	1.123	1.073	1.079	1.070	1.186	1.106
7*	0.972	0.931	1.084	1.015	1.023	1.005
8*	0.731	0.944	0.873	0.855	0.944	0.869
9*	1.192	1.031	1.047	0.974	1.146	1.078
10	1.151	0.964	1.083	1.064	1.187	1.090
11	1.184	1.053	1.213	1.182	1.215	1.169
12	1.140	1.077	1.114	1.066	1.153	1.110
13	1.173	1.218	1.181	1.185	1.239	1.199
14	1.069	1.088	1.145	0.999	1.138	1.088
15	1.039	1.166	1.120	1.106	1.156	1.117
16	0.751	0.928	0.851	0.759	0.955	0.849
17*	1.167	0.980	1.122	0.997	1.187	1.091
18	1.189	1.066	1.216	1.183	1.220	1.175
19	1.143	1.012	1.110	1.089	1.234	1.118
20	1.199	1.237	1.196	1.074	1.278	1.197
21	1.108	1.015	1.098	1.110	1.219	1.110
22	1.097	1.194	1.180	1.225	1.250	1.189
23	0.929	0.905	1.048	1.047	1.106	1.007
24	0.656	0.829	0.752	0.826	0.853	0.783
25*	1.224	1.127	1.211	1.042	1.257	1.172
26	1.165	1.077	1.119	1.076	1.163	1.120
27	1.201	1.242	1.199	1.104	1.292	1.208
28	1.139	1.011	0.970	1.098	1.233	1.090
29	1.134	1.178	1.125	1.244	1.216	1.179
30	1.036	0.942	1.034	1.073	1.183	1.054
31	0.965	1.081	0.999	1.118	1.119	1.056
32	0.545	0.556	0.423	0.563	0.459	0.509
33*	1.169	1.004	1.119	0.994	1.195	1.096
34	1.199	1.233	1.193	1.198	1.265	1.218
35	1.127	1.026	1.017	1.121	1.226	1.103
36	1.146	1.184	1.127	1.249	1.258	1.193
37	1.166	0.912	1.052	1.038	1.216	1.077
38	0.983	0.984	0.955	1.173	1.215	1.062
39	0.814	0.901	0.781	0.767	0.773	0.807
40*	1.095	1.045	1.075	1.062	1.182	1.092
41	1.085	1.096	1.151	0.994	1.173	1.100
42	1.148	1.194	1.191	1.217	1.253	1.201
43	1.070	0.956	1.039	1.067	1.203	1.067
44	1.019	0.986	0.941	1.182	1.210	1.068
45	0.973	1.051	0.893	1.014	1.007	0.988
46	0.497	0.547	0.401	0.404	0.389	0.448

*1/4 assembly in 1/4 core.
**1/2 assembly in 1/4 core.
NOTE: Circled values correspond to peripheral assemblies.

Table A.2. Absolute Assembly (i.e., Zone) Power for Donald C. Cook Unit 2

Total Power = 3391 MW_{th}
No. of assemblies = 193

$$\therefore \overline{P} \text{ per assembly} = \frac{3391}{193} = 17.57 \frac{MW}{\text{assembly}}$$

Zone	Relative Power	Absolute Power (MW)
1**	0.945	4.151
2*	1.077	9.461
3*	1.081	9.497
4*	1.167	10.252
5*	1.074	9.435
6*	1.106	9.716
7*	1.005	8.829
8*	0.869	7.634
9*	1.078	9.470
10	1.090	19.151
11	1.169	20.539
12	1.110	19.503
13	1.199	21.066
14	1.088	19.116
15	1.117	19.626
16	0.849	14.917
17*	1.091	9.584
18	1.175	20.645
19	1.118	19.643
20	1.197	21.031
21	1.110	19.503
22	1.189	20.891
23	1.007	17.693
24	0.783	13.757
25*	1.172	10.296
26	1.120	19.678
27	1.208	21.224
28	1.090	19.151
29	1.179	20.715
30	1.054	18.519
31	1.056	18.554
32	0.509	8.943
33*	1.096	9.628
34	1.122	19.710
35	1.103	19.380
36	1.193	20.961
37	1.077	18.923
38	1.062	18.659
39	0.807	14.179
40*	1.092	9.593
41	1.100	19.327
42	1.201	21.102
43	1.067	18.747
44	1.068	18.765
45	0.988	17.359
46	0.448	7.871

**1/4 assembly in 1/4 core.
*1/2 assembly in 1/4 core.

NOTE: Circled values correspond to peripheral assemblies.

Figure A.1. Identification of Assembly Nomenclature
Used in Source Determination

40	41	42	43	44	45	46	
33	34	35	36	37	38	39	
25	26	27	28	29	30	31	32
17	18	19	20	21	22	23	24
9	10	11	12	13	14	15	16
1	2	3	4	5	6	7	8

APPENDIX B

Description of the 3-D Flux Synthesis Method

Appendix B

DESCRIPTION OF THE 3-D FLUX SYNTHESIS METHOD

A 3-D (RΘZ) flux distribution is synthesized using the following well established approximation:

$$\phi(R, \theta, Z) = \phi_{R\theta}(R,\theta)\frac{\phi_{RZ}(R,Z)}{\phi_R(R)} = \phi_{R\theta}A(R,Z) \qquad \text{B.1}$$

where $\phi_{R\theta}$ is the flux obtained from the RΘ DOT calculation; and

$A(R,Z) = \frac{\phi_{RZ}}{\phi_R} =$ axial distribution function obtained by representing the RZ flux = (ϕ_{RZ}) distribution and dividing it by the integral over Z of the RZ flux; i.e.,

$$\phi_R = \int_Z \phi_{RZ}\, dZ.$$

In some previous studies, the RZ flux distribution was represented by the results obtained from a DOT RZ calculation, while the radial flux ϕ_R was obtained from a one-dimensional calculation. However, it has been discovered that a simpler approximation gives similar results (within a few percent) as the result of these transport calculations for locations not outside of the RPV and near the reactor midplane. In this approach, we represent

$$A(R,Z) = \frac{\phi_{RZ}(R,Z)}{\int_Z \phi_{RZ}\, dZ} \cong \frac{P(Z)}{\int_Z P(Z)dZ} \qquad \text{B.2}$$

where P(Z) is the average axial distribution of power in the core. The function P(Z) has been represented by 61 discrete nodal values provided by American Electric Power. These values, which are shown in Table B.1 and B.2, correspond to the average relative power for 61 six-centimeter nodes defined over the core height. Table B.1 is the MOC axial distribution for a twice-burned peripheral assembly, while Table B.2 is for a fresh peripheral assembly.

Employing the expression in Eq. B.2, we find

$$A(R,Z) \cong A(Z) \rightarrow A_K = \frac{P_K}{\sum_{i=1} P_K \Delta Z} \; ; \; K=1, 61$$

Evaluating the denominator by summing the values in Tables B.1 and B.2, and multiplying by $\Delta Z=6$ gives

$A_K = \frac{P_K}{163}$ = axial flux factor for node K for burned assembly (P_K taken from Table B.1)

$A_K = \frac{P_K}{150.8}$ = axial flux factor for node K for fresh assembly (P_K taken from Table B.2)

The axial factors (A_K) used in synthesizing the RθZ fluxes are also shown in Tables B.1 and B.2. Note from these tables that the axial flux factors have different axial variations for the fresh and burned assemblies (indicating a difference in the relative flux shape). However, the peak value in each case is nearly identical (~3.1 E-3), and occurs at approximately the same location (~35 inches below the midplane). The axial distribution is fairly flat in both cases, and varies by only about 10% over the middle 9 feet of the core. Since surveillance capsule X as well as the peak RPV flux are located opposite a twice-burned assembly, the axial distribution factors in Table B.1 are more appropriate for this analysis.

In order to compute the 3-D flux or activity at some axial node i (corresponding to a height Z in Tables B.1 and B.2), for some Rθ location one must

1. find the flux or activity at the appropriate (R_I, θ_J) location in the DOT Rθ run
2. find the axial flux factor at the appropriate node K
3. compute the 3-D value using expression

$$\phi(R_I, \theta_J, Z_I) = \phi_{R\theta}(R_I, \theta_J) * A_K$$

(*) For example, the reactor midplane corresponds to node 31. From Table B.1, it can be seen that the axial flux factor for node 31 is equal to 3.063×10^{-3}. Therefore, all activities and fluxes in the DOT Rθ output should be multiplied by this factor in order to obtain the corresponding midplane values. All of the dosimeter results given in the tables presented previously correspond to midplane values obtained in this manner. The maximum values occur below the midplane and are obtained by using an axial factor of 3.143×10^{-3}.

Table B.1. Axial Distribution Factors for Burned Peripheral Assembly in Donald C. Cook Unit 2

	Node	Z_k (cm)	P_k (relative power)	A_k (axial flux factor)
Top	1	3.0	0.212	1.301E-3
	2	9.0	0.212	1.301E-3
	3	15.0	0.268	1.645E-3
	4	21.0	0.318	1.952E-3
	5	27.0	0.359	2.204E-3
	6	33.0	0.386	2.369E-3
	7	39.0	0.368	2.259E-3
	8	45.0	0.411	2.523E-3
	9	51.0	0.444	2.725E-3
	10	57.0	0.456	2.799E-3
	11	63.0	0.463	2.842E-3
	12	69.0	0.474	2.910E-3
	13	75.0	0.477	2.928E-3
	14	81.0	0.479	2.940E-3
	15	87.0	0.470	2.885E-3
	16	93.0	0.413	2.535E-3
	17	99.0	0.470	2.885E-3
	18	105.0	0.483	2.965E-3
	19	111.0	0.488	2.995E-3
	20	117.0	0.494	3.032E-3
	21	123.0	0.496	3.045E-3
	22	129.0	0.498	3.057E-3
	23	135.0	0.494	3.032E-3
	24	141.0	0.462	2.836E-3
	25	147.0	0.444	2.725E-3
	26	153.0	0.488	2.995E-3
	27	159.0	0.491	3.014E-3
	28	165.0	0.496	3.045E-3
	29	171.0	0.499	3.063E-3
	30	177.0	0.501	3.075E-3
Midplane	31	183.0	0.499	3.063E-3
	32	189.0	0.493	3.026E-3
	33	195.0	0.438	2.689E-3
	34	201.0	0.476	2.922E-3
	35	207.0	0.496	3.045E-3
	36	213.0	0.498	3.057E-3
	37	219.0	0.499	3.063E-3
	38	225.0	0.504	3.094E-3
	39	231.0	0.504	3.094E-3
	40	237.0	0.503	3.088E-3
	41	243.0	0.491	3.014E-3
	42	249.0	0.438	2.689E-3
	43	255.0	0.497	3.051E-3
	44	261.0	0.507	3.112E-3
	45	267.0	0.512	3.143E-3
	46	273.0	0.512	3.143E-3

Table B.1. (continued)

	Node	Z_k (cm)	P_k (relative power)	A_k (axial flux factor)
	47	279.0	0.511	3.137E-3
	48	285.0	0.507	3.112E-3
	49	291.0	0.499	3.063E-3
	50	297.0	0.462	2.836E-3
	51	303.0	0.442	2.713E-3
	52	309.0	0.484	2.971E-3
	53	315.0	0.482	2.959E-3
	54	321.0	0.477	2.928E-3
	55	327.0	0.466	2.860E-3
	56	333.0	0.449	2.756E-3
	57	339.0	0.422	2.590E-3
	58	345.0	0.381	2.339E-3
	59	351.0	0.332	2.037E-3
	60	357.0	0.266	1.632E-3
Bottom	61	363.0	0.133	8.160E-4

Table B.2. Axial Distribution Factors for Fresh Peripheral Assembly in Donald C. Cook Unit 2

	Node	Z_k (cm)	P_k (relative power)	A_k (axial flux factor)
Top	1	3.0	0.174	1.154E-3
	2	9.0	0.183	1.214E-3
	3	15.0	0.238	1.578E-3
	4	21.0	0.283	1.877E-3
	5	27.0	0.320	2.122E-3
	6	33.0	0.347	2.301E-3
	7	39.0	0.348	2.308E-3
	8	45.0	0.373	2.474E-3
	9	51.0	0.403	2.673E-3
	10	57.0	0.416	2.759E-3
	11	63.0	0.427	2.832E-3
	12	69.0	0.432	2.865E-3
	13	75.0	0.434	2.878E-3
	14	81.0	0.435	2.885E-3
	15	87.0	0.428	2.839E-3
	16	93.0	0.405	2.686E-3
	17	99.0	0.431	2.858E-3
	18	105.0	0.436	2.892E-3
	19	111.0	0.438	2.905E-3
	20	117.0	0.442	2.931E-3
	21	123.0	0.444	2.945E-3
	22	129.0	0.445	2.951E-3
	23	135.0	0.444	2.945E-3
	24	141.0	0.420	2.786E-3
	25	147.0	0.425	2.819E-3
	26	153.0	0.450	2.984E-3
	27	159.0	0.457	3.031E-3
	28	165.0	0.458	3.038E-3
	29	171.0	0.460	3.051E-3
	30	177.0	0.459	3.044E-3
Midplane	31	183.0	0.461	3.057E-3
	32	189.0	0.454	3.011E-3
	33	195.0	0.427	2.832E-3
	34	201.0	0.451	2.991E-3
	35	207.0	0.461	3.057E-3
	36	213.0	0.464	3.077E-3
	37	219.0	0.466	3.091E-3
	38	225.0	0.467	3.097E-3
	39	231.0	0.467	3.097E-3
	40	237.0	0.465	3.084E-3
	41	243.0	0.447	2.965E-3
	42	249.0	0.436	2.892E-3
	43	255.0	0.465	3.084E-3
	44	261.0	0.473	3.137E-3
	45	267.0	0.476	3.157E-3
	46	273.0	0.478	3.170E-3

Table B.2. (continued)

	Node	Z_k (cm)	P_k (relative power)	A_k (axial flux factor)
	47	279.0	0.478	3.170E-3
	48	285.0	0.478	3.170E-3
	49	291.0	0.473	3.137E-3
	50	297.0	0.442	2.931E-3
	51	303.0	0.461	3.057E-3
	52	309.0	0.466	3.091E-3
	53	315.0	0.458	3.038E-3
	54	321.0	0.450	2.984E-3
	55	327.0	0.434	2.878E-3
	56	333.0	0.413	2.739E-3
	57	339.0	0.382	2.533E-3
	58	345.0	0.342	2.268E-3
	59	351.0	0.286	1.897E-3
	60	357.0	0.207	1.373E-3
Bottom	61	363.0	0.207	1.373E-3

APPENDIX C

Tensile Test Data Records

Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. M111-8 Project No. 06-8888-001

Test Temperature 210°F Machine Ident. #4

Strain Rate .005 in/in/min Date of Test 4/21/87

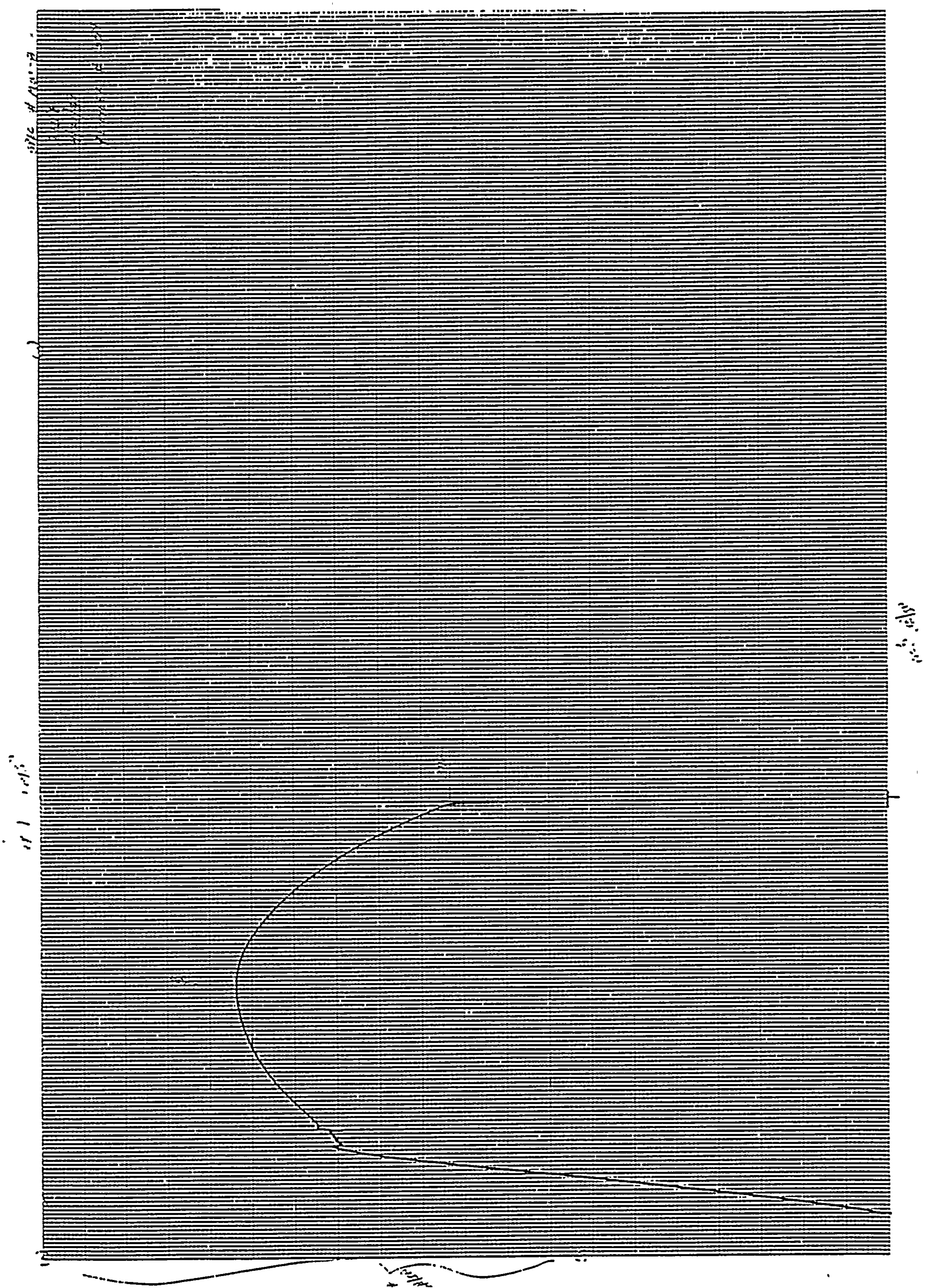
Initial Diameter	.250 in	Final Diameter	.147 in
Initial Area	.049 in²	Final Area	.017 in²
Initial Gage Length	1.0 in	Final Gage Length	1.214 in
Specimen Temperature:		Maximum Load	4630#
Top T.C.	212°F	0.2% Offset Load	3917#
Middle T.C.	211	Fracture Load	3112#
Bottom T.C.	210°F	Elong. to Max. Load	.139 in

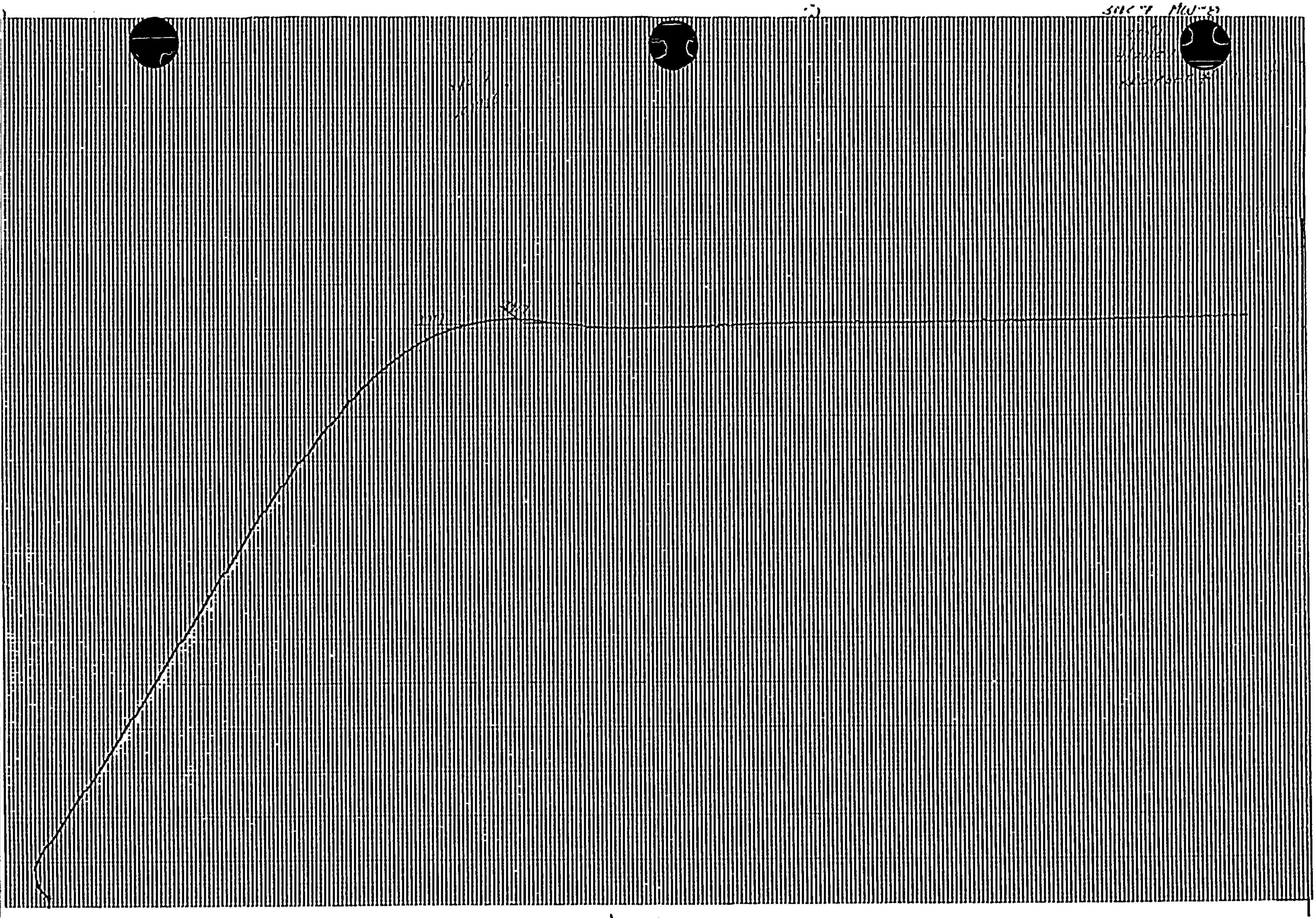
U.T.S. = Maximum Load/Initial Area	=	94,490
0.2% Y.S. = 0.2% Offset Load/Initial Area	=	79,939
Frature Stress = Fracture Load/Final Area	=	183,059
% R.A. = 100 (Init. Area-Final Area)/Init. Area	=	65.31
% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L.	=	21.40
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L.	=	13.90

Test Performed by: Victor D. Dana

Calculations Performed by: T.E. MASDEN (Date) 4/27/87

Calculations Checked by: [signature] (Date) 5/7/87





Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. M10-7

Project No. 06-8888-001

Test Temperature 550°F

Machine Ident. #4

Strain Rate .005 in/in/min

Date of Test 4/22/87

Initial Diameter .250 in
Initial Area .049 in²
Initial Gage Length 1.0 in
Specimen Temperature:
Top T.C. 552°F
Middle T.C. N/A
Bottom T.C. 549°F

Final Diameter .155 in
Final Area .0189 in²
Final Gage Length 1.188 in
Maximum Load ~~2944~~ 4534#
0.2% Offset Load 3612#
Fracture Load 3148#
Elong. to Max. Load .114 in

U.T.S. = Maximum Load/Initial Area = 92,531 ~~59,427~~

0.2% Y.S. = 0.2% Offset Load/Initial Area = 73 714

Frature Stress = Fracture Load/Final Area = 166,561

% R.A. = 100 (Init. Area-Final Area)/Init. Area = 61.43

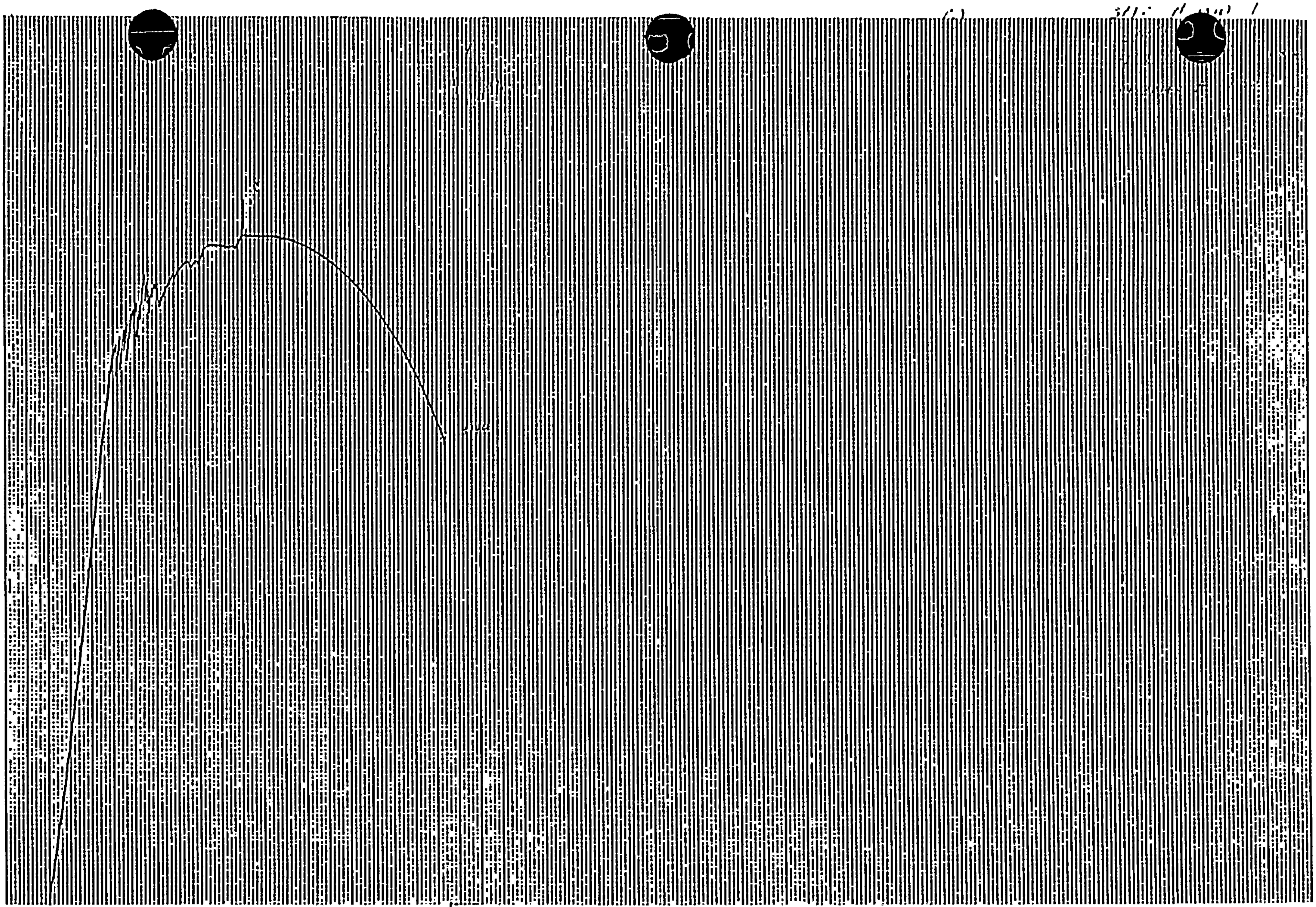
% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L. = 18.80

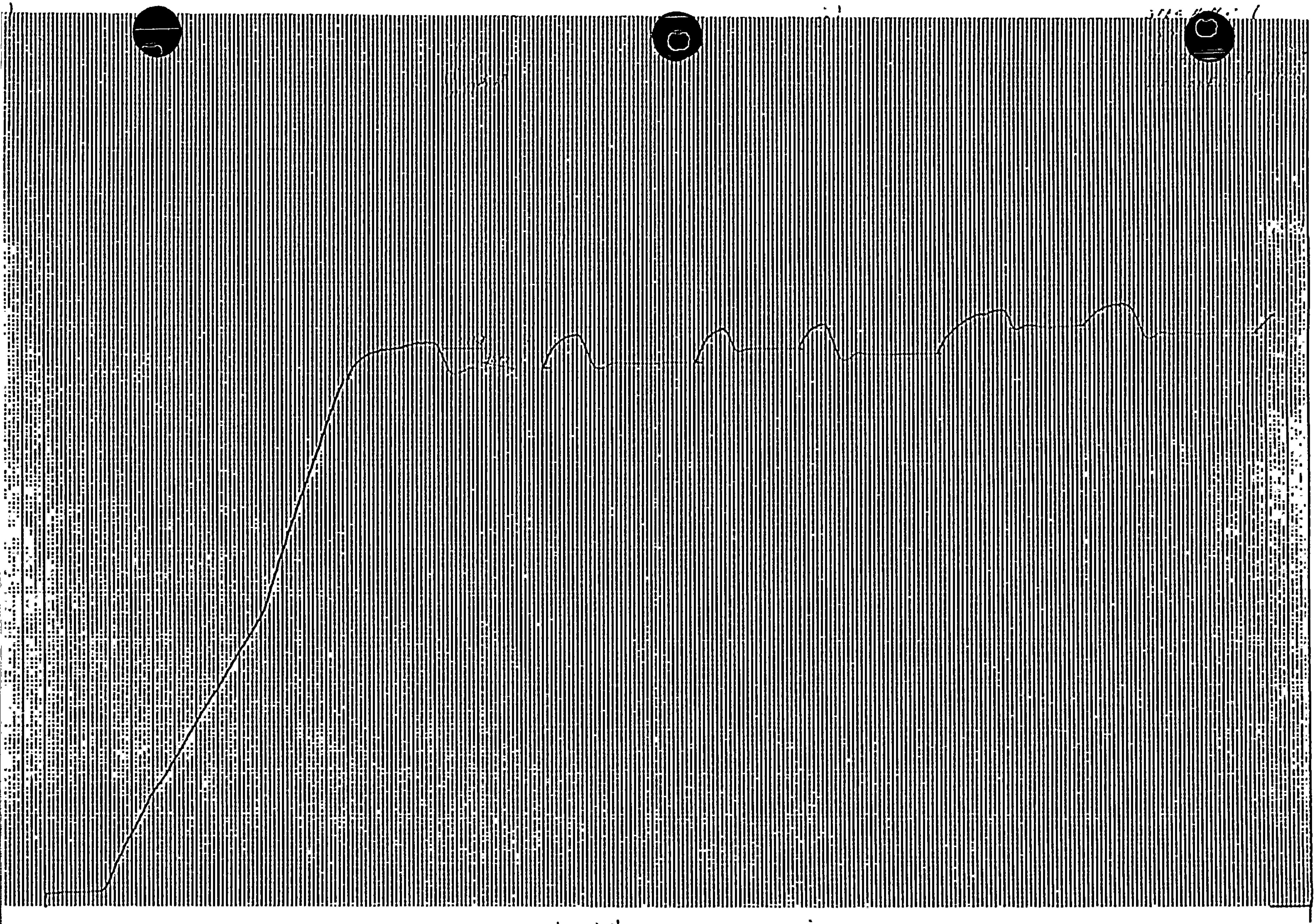
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L. = 11.40

Test Performed by: Victor D. Oliva

Calculations Performed by: TE MASDEN (Date) 4/27/87

Calculations Checked by: [signature] (Date) 5/7/87





Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. MT-8

Test Temperature 250°F

Strain Rate .005"/in/min

Project No. 06-8285-001

Machine Ident. #4

Date of Test 4/23/87

Initial Diameter .249 in
Initial Area .0487 in²
Initial Gage Length 1.0 in
Specimen Temperature:
Top T.C. 250°F
Middle T.C. N/A
Bottom T.C. 247°F ~~250°F~~

Final Diameter .171 in
Final Area .023 in²
Final Gage Length 1.187 in
Maximum Load 4572 #
0.2% Offset Load 3702 #
Fracture Load 3588 #
Elong. to Max. Load .150 in

U.T.S. = Maximum Load/Initial Area = 93881

0.2% Y.S. = 0.2% Offset Load/Initial Area = 76,016

Frature Stress = Fracture Load/Final Area = 156000

% R.A. = 100 (Init. Area-Final Area)/Init. Area = 52.77

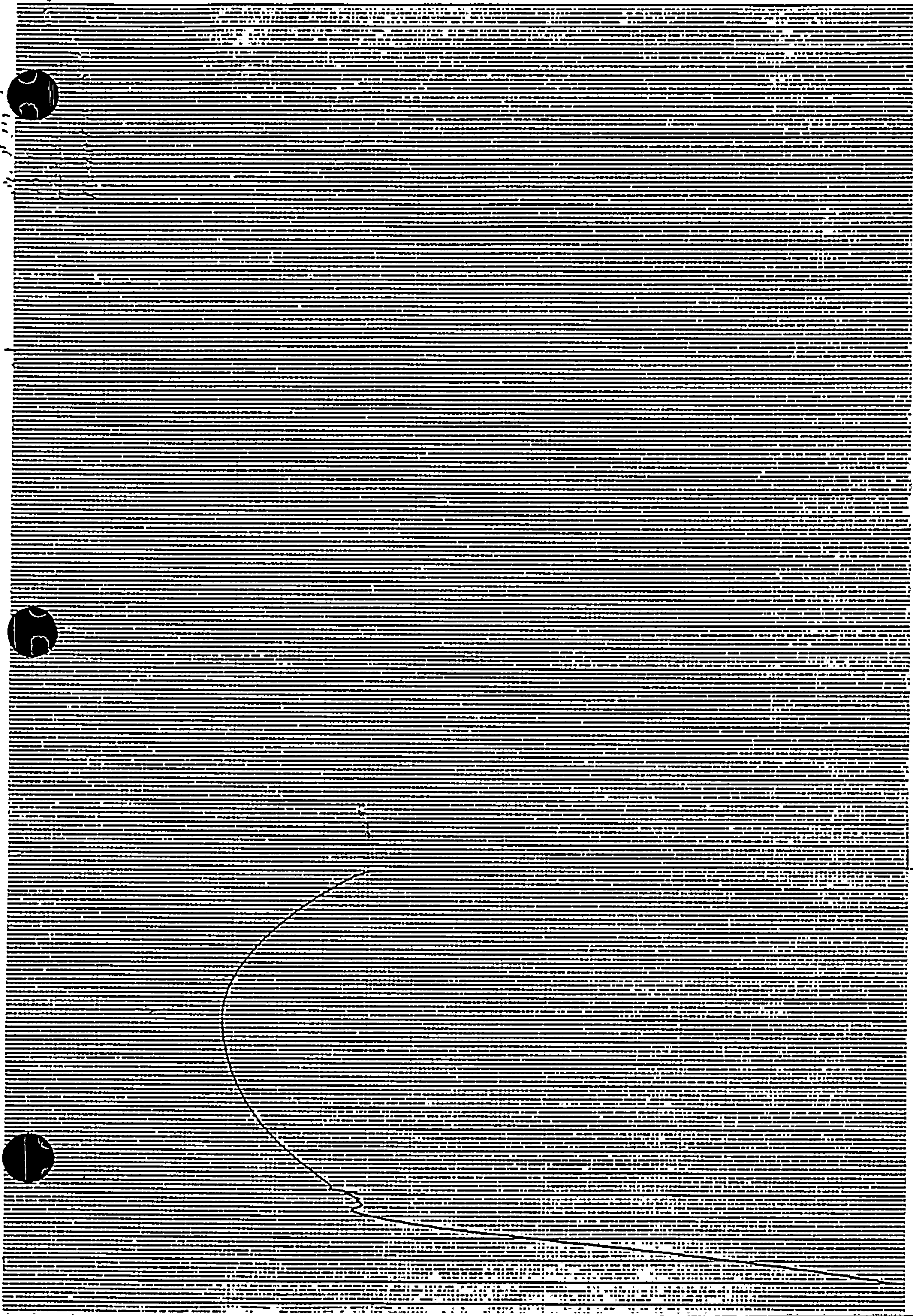
% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L. = 18.70

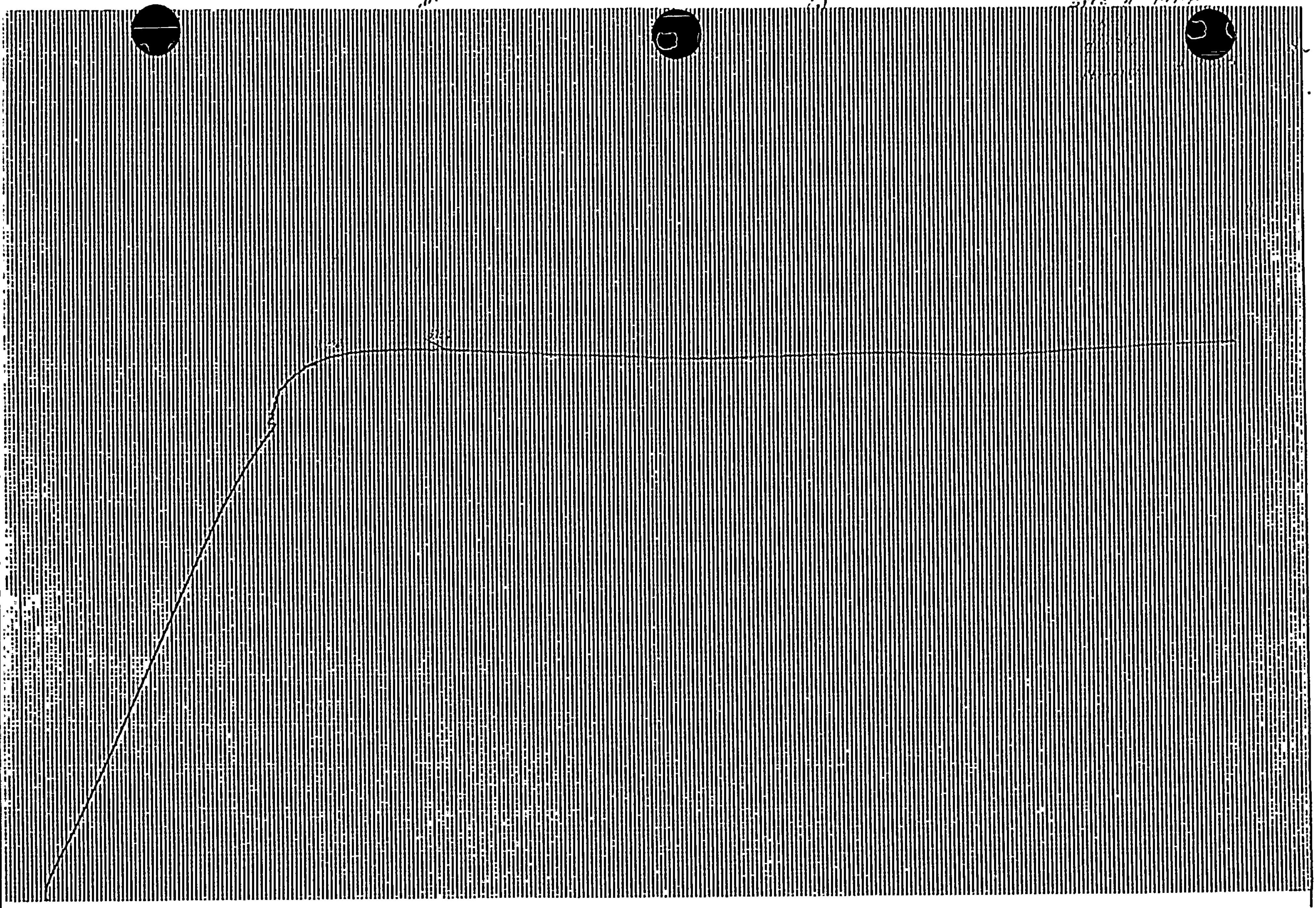
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L. = 15.00

Test Performed by: [signature]

Calculations Performed by: T.E. MARSDEN (Date) 4/27/87

Calculations Checked by: [signature] (Date) 5/7/87





Southwest Research Institute

Department of Materials Sciences

TENSILE TEST DATA SHEET

Specimen No. MT-7

Project No. 06-8888-001

Test Temperature 550°F

Machine Ident. #4

Strain Rate .005 in/in/min

Date of Test 4/22/87

Initial Diameter .249 in
Initial Area .0487 in²
Initial Gage Length 1.0 in
Specimen Temperature:
Top T.C. 550°F
Middle T.C. N/A
Bottom T.C. 547°F

Final Diameter .169 in
Final Area .0224 in²
Final Gage Length 1.173 in
Maximum Load 4493#
0.2% Offset Load 3510#
Fracture Load 3672#
Elong. to Max. Load .146 in

U.T.S. = Maximum Load/Initial Area = 92,259

0.2% Y.S. = 0.2% Offset Load/Initial Area = 72074

Frature Stress = Fracture Load/Final Area = 163,929

% R.A. = 100 (Init. Area-Final Area)/Init. Area = 54.00

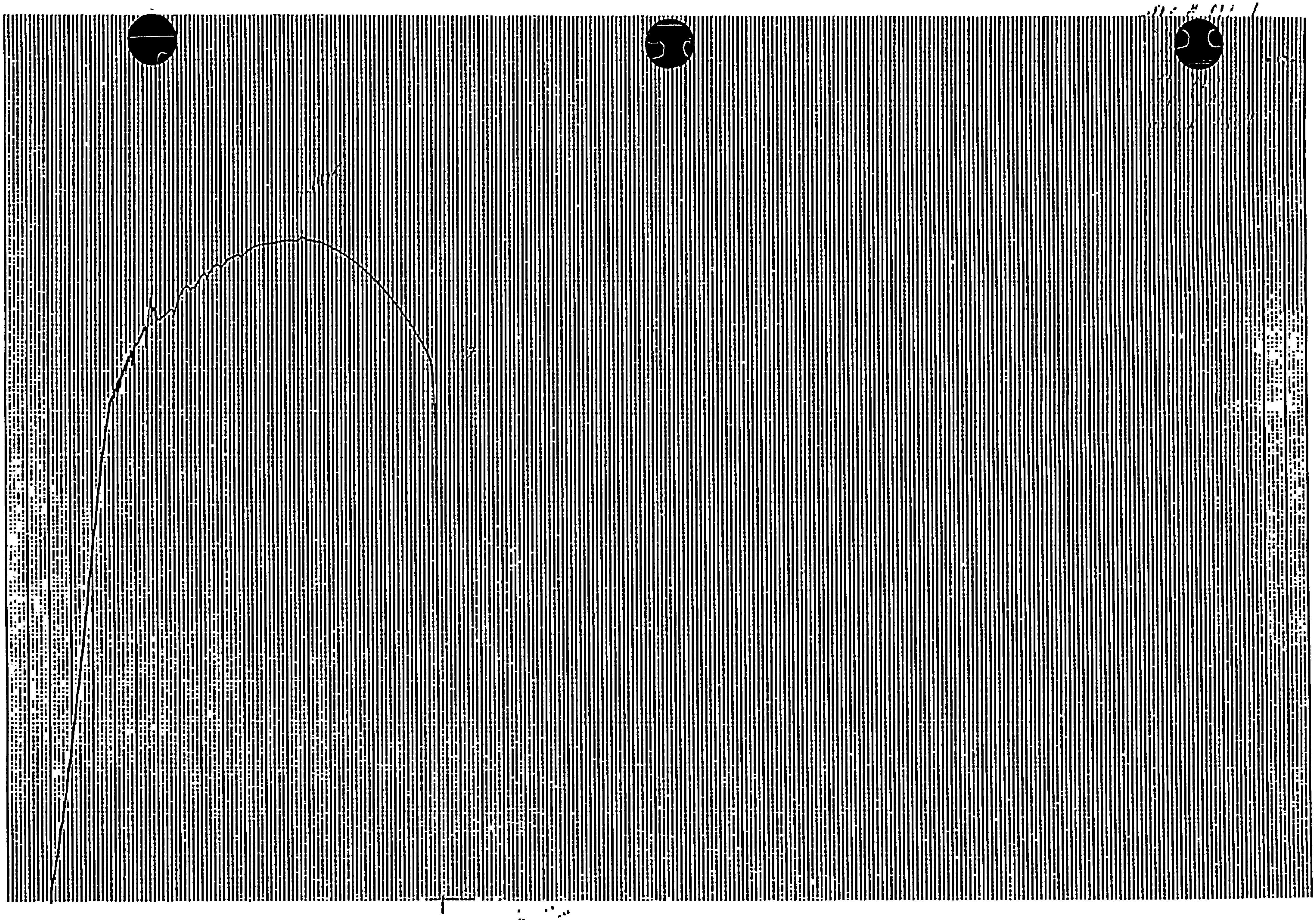
% Total Elong. = 100 (Final G.L.-Init. G.L.)/Init. G.L. = 17.30

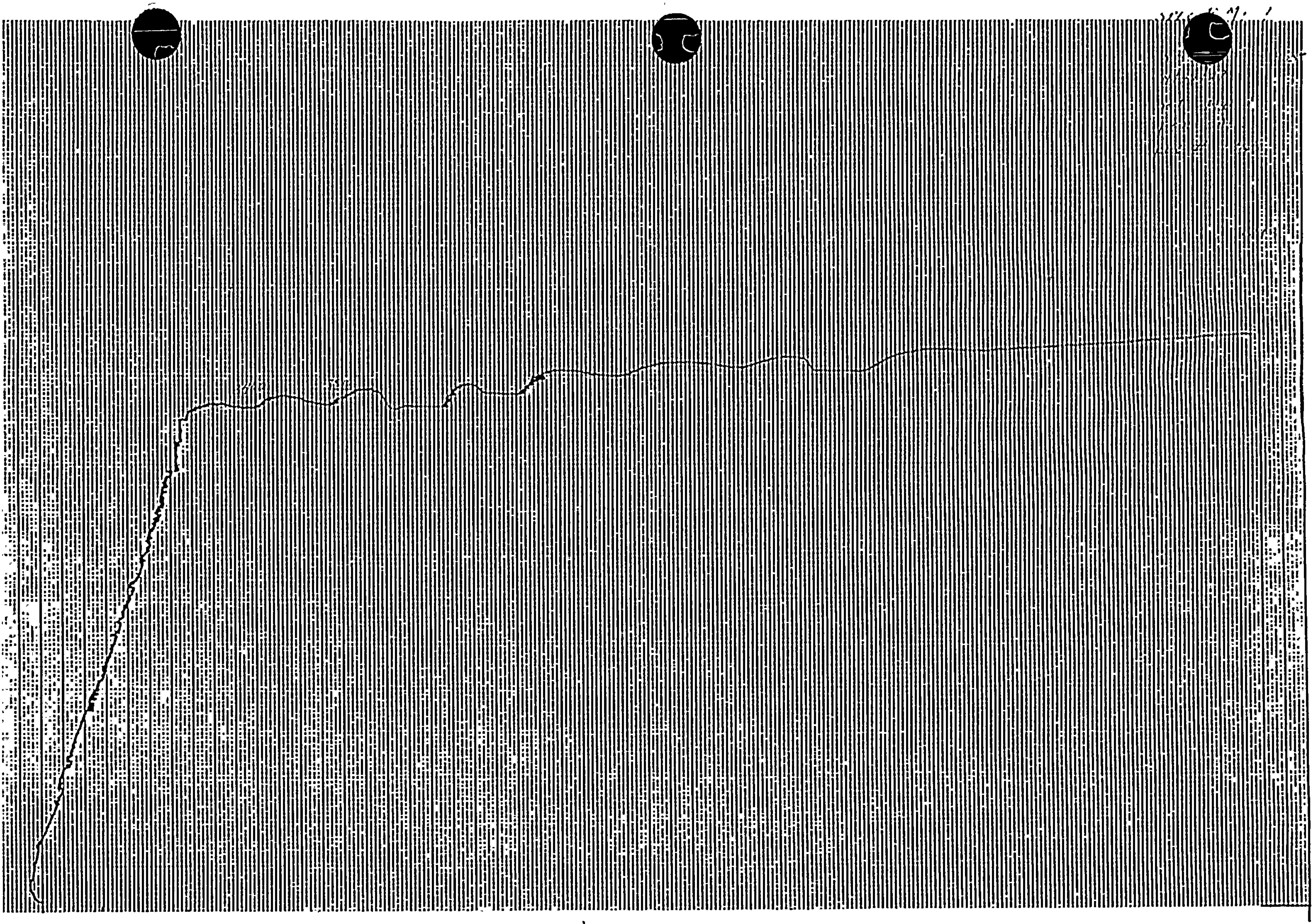
% Uniform Elong. = 100 (Elong. to Max. Load)/Init. G.L. = 14.60

Test Performed by: Victor D. Adams

Calculations Performed by: H. Madsen / H. MADSEN (Date) 4/27/87

Calculations Checked by: [illegible] (Date) 5/7/87





Attachment 5
page 1 of 2

TABLE 4.1-12

REACTOR COOLANT SYSTEM CODES

Component	Code	Unit 1 Addenda and Code Cases
Reactor Vessel	ASME III* Class A	1965 Ed. through 1966 Winter Addenda, Code Cases 1332-2, 1358, 1339-2, 1335, 1359-1, 1338-3, 1336
Full Length Control Rod Drive Mechanisms	ASME III* Class A	1965 Ed. through 1966 Winter Addenda
Steam Generators	ASME III* Class A	1965 Ed. through 1966 Winter Addenda
Reactor Coolant Pump Casings	No Code (Designed with ASME III* Article 4 as a Guide)	1968 Edition
Pressurizer	ASME III* Class A	1965 Ed. through Winter 1966 Addenda, Code Cases 1401, 1459
Pressurizer Safety Valves	ASME III*	1968 Edition
Power Operated Relief Valves	B-16.5	
Main Reactor Coolant System Piping	B31.1**	1967 Edition
Reactor Coolant System Valves	B-16.5 or MSS-SP-66, and ASME III, 1968 Edition*	

*ASME Boiler and Pressure Vessel Code, Section III-Nuclear Vessels

**Repairs and replacements are conducted in accordance with ASME Section XI

TABLE 4.1-12 (cont'd.)

Component	Code	Unit 2 Addenda and Code Cases
Reactor Vessel	ASME III* Class A	1968 Ed. (1968 Summer Addenda) Code Case 1335-4
Full Length Control Rod Drive Mechanisms	ASME III* Class A	1968 Ed. (No Add.)
Steam Generators	ASME III* Class A	1968 Ed. through Winter 1968 Addenda, Code Cases 1401, 1498 for upper assemblies and 1983 Ed. through Summer 1984 for replacement lower assemblies
Reactor Coolant Pump Casings	No Code (Designed with ASME III* Article 4 as a Guide)	1968 Edition through Summer 1969 Addenda
Pressurizer	ASME III* Class A	1965 Ed. through Winter 1966 Addenda
Pressurizer Safety Valves	ASME III*	1968 Edition
Power Operated Relief Valves	B-16.5	
Main Reactor Coolant System Piping	B31.1**	1967 Edition
Reactor Coolant System Valves	B-16.5 or MSS-SP-66, and ASME III, 1968 Edition*	

*ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels

**Repairs and replacements are conducted in accordance with ASME Section XI

Attachment 6
page 1 of 9

November 7, 1977

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
DPR No. 58

COPY

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Case:

This letter responds to Mr. Don K. Davis' letter of [illegible] 20, 1977 requesting reactor vessel material property information for the Donald C. Cook Nuclear Plant. In our letter dated July 25, 1977, we informed you that we would need additional time to provide the requested information.

Enclosed herewith are three (3) copies of a document entitled, "D. C. Cook Unit No. 1 Reactor Vessel Material Surveillance Program" which supplies the information requested.

Very truly yours,

John Tillinghast
John Tillinghast
Vice President

JT:mam

Sworn and subscribed to before me
on this 7th day of November 1977
in New York County, New York

Gregory M. [illegible]

Notary Public

GREGORY M. [illegible]
Notary Public, State of New York
No. 31-4643431
Qualified in New York County
Commission Expires March 30, 19.75

Attachment 6
page 2 of 9

cc: G. Charnoff
P. W. Steketee
R. J. Vollen
R. C. Callen
R. Walsh
D. V. Shaller - Bridgman
R. W. Jurgensen

bc: S. J. Milioti/P. W. Daley
J. G. Feinstein
M. H. Fletcher - NRC
M. M. Mlynczak - NRC
DC-N-6015.1
DC-N-6079

COPY

D. C. COOK UNIT NO. 1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

1.) The estimated maximum fluence (E >.1 Mev) at the inner surface of the reactor vessel as of March 31, 1977 is 8.38×10^{17} n/cm^2.

2.) The effective full power years (EFPY) of operation accumulated as of March.31, 1977 is 1.34 EFPY.

3.) Fabrication of the reactor vessel was performed by Combustion Engineering, Inc.

4.) a.) Sketch of the reactor vessel showing materials in the beltline region is shown in Figure 1.

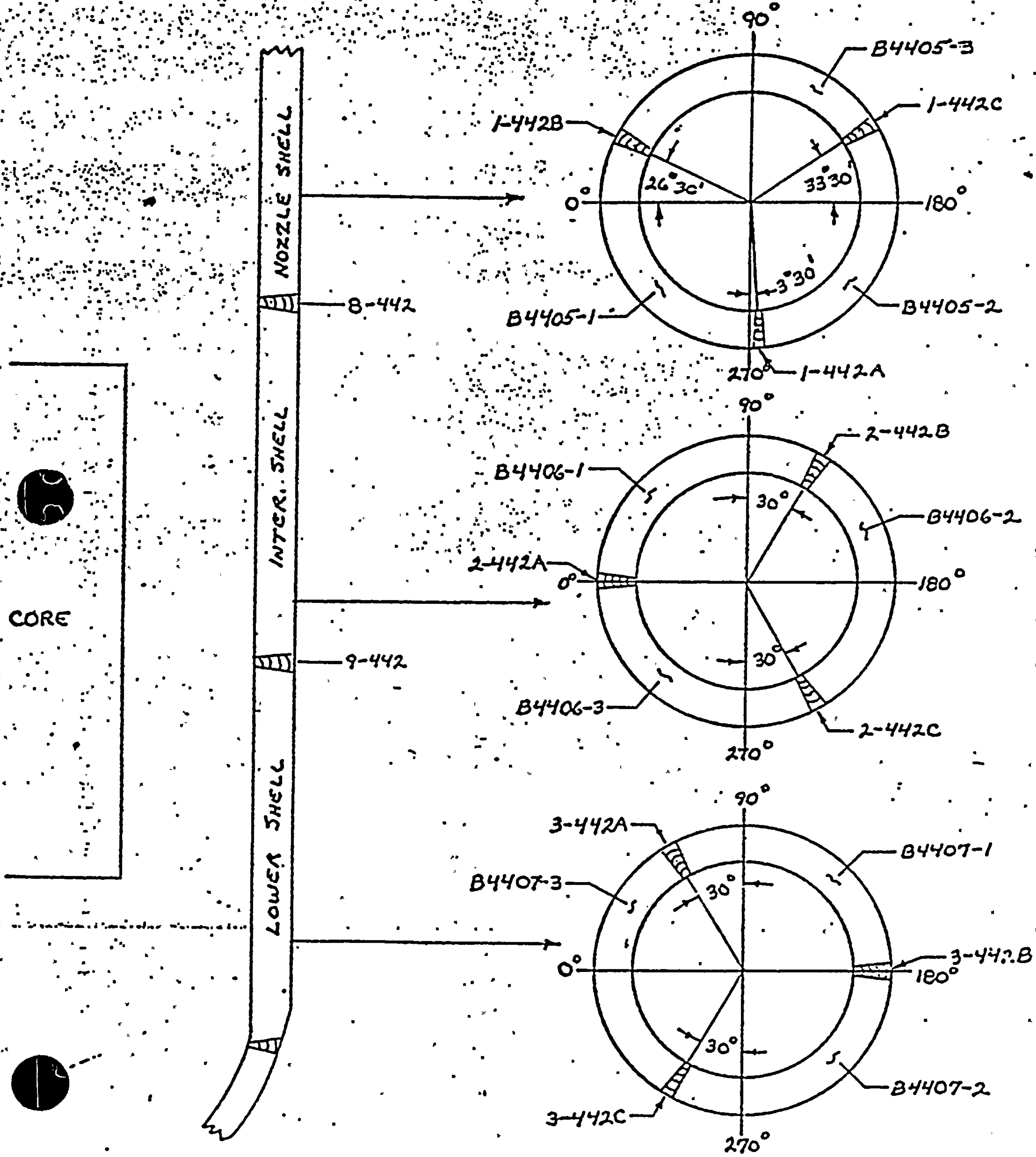
b.) Information on each of the welds in the beltline region is shown in Tables 1 through 4.

c.) Information on each of the plates in the beltline region is shown in Tables 4 through 8.

5.) Information relative to the weld and plate material in the material surveillance program is shown in Tables 1 through 3 and 5 through 8.

FIGURE 1

IDENTIFICATION AND LOCATION OF D. C. COOK UNIT NO. 1 REACTOR VESSEL BELTLINE REGION WELD AND PLATE MATERIAL



TA

IDENTIFICATION AND LOCATION OF D. C. COOK UNIT NO. 1 VESSEL BELTLINE REGION WELD METAL

Weld Location	Welding Process	Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Post Weld Heat Treatment
Nozzle Shell Vertical Seams 1-442 A, B & C	Submerged Arc (Tandem Wire)	M1.14	B-4 Mod. B-4 Mod.	13253 12008	Linde 1092	3791	1125-1175°F-40HR-FC
Nozzle Shell to Inter Shell Circle Seam 8-442	Submerged Arc	M1.18	B-4 Mod.	20291	Linde 1092	3833	1125-1175°F-40HR-FC
Inter. Shell Vertical Seams 2-442 A, B & C	Submerged Arc (Tandem Wire)	M1.14	B-4 Mod. B-4 Mod.	13253 12008	Linde 1092	3791	1125-1175°F-40HR-FC
Inter. to Lower Shell Circle Seam 9-442	Submerged Arc	M1.42	B-4 Mod.	IP3571	Linde 1092	3958	1125-1175°F-40HR-FC
Lower Shell Vertical Seams 3-442 A, B & C	Submerged Arc (Tandem Wire)	M1.14	B-4 Mod. B-4 Mod.	13253 12008	Linde 1092	3791	1125-1175°F-40HR-FC
Surveillance Weld	Submerged Arc		B-4 Mod.	13253	Linde 1092	3791	1125-1175°F-40HR-FC

Attachment 6
page 5 of 9

TA[illegible]

CHEMICAL COMPOSITION OF VESSEL BELTLINE REGION WELD METAL

Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	C	Mn	P	S	Si	Ni	Mo	Cr	Cu	V	
B4 Mod.	13253	Linde 1092	3791	.15	1.83	.013	.015	.06	.72	.45	.04	.07	--	*
B4 Mod.	12008	Linde 1092	3791	.13	1.92	.010	.015	.05	.99	.51	.06	.13	--	*
B4 Mod.	20291	Linde 1092	3833	.16	1.92	.008	.009	.03	.74	.51	--	--	--	
B4 Mod.	IP3571	Linde 1092	3958	.12	1.38	.017	.009	.21	.82	.54	--	.40	--	
	Surveillance Weld			.26	1.33	.023	.014	.18	.74	.44	.02	.27	.001	

Weight Percent: columns C through V.

* Wire Analysis - No As Deposited Weld Analysis was Performed

TABLE 3

MECHANICAL PROPERTIES OF VESSEL BELTLINE REGION WELD METAL

Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	T_{NDT} °F	Energy at 10°F Ft-Lbs	RT_{NDT} °F	Shelf Energy Ft-Lbs	YS KSI	UTS KSI	Elong. %	RA %
B4 Mod. B4 Mod.	13253 12008	Linde 1092	3791	0*	84,74,70	0*	--	63.3	80.1	27.5	69.7
B4 Mod.	20291	Linde 1092	3833	0*	35,50,48	0*	--	70.5	88.0	25.5	67.1
B4 Mod.	IP3571	Linde 1092	3958	0*	40,46,46	0*	--	69.0	84.0	28.0	69.4
Surveillance Weld		CE Tests		-70	54,54,73	-56	115.5	--	--	--	--
Surveillance Weld		W Tests		--	83,84,92	-70	111	67.1	81.9	26.8	69.2

* Estimated per NRC Standard Review Plan Section 5.3.2

Attachment 6
page 6 of 9

TABLE 4

MAXIMUM END-OF-LIFE FLUENCE AT VESSEL INNER WALL LOCATIONS

Plate or Weld Seam Location	Plate or Seam No.	Fluence N/CM^2
Nozzle Shell Vertical Seam	1-442A	2.4×10^{17}
" " " "	1-442B	3.9×10^{17}
" " " "	1-442C	4.9×10^{17}
Nozzle Shell to Inter. Shell Circle Seam	8-442	7.3×10^{17}
Inter. Shell Vertical Seam	2-442A	6.2×10^{18}
" " " "	2-442B	1.1×10^{19}
" " " "	2-442C	1.1×10^{19}
Inter. Shell to Lower Shell Circle Seam	9-442	2.0×10^{19}
Lower Shell Vertical Seam	3-442A	1.1×10^{19}
" " " "	3-442B	6.2×10^{18}
" " " "	3-442C	1.1×10^{19}
Nozzle Shell Plate	B4405-1	7.3×10^{17}
" " "	B4405-2	7.3×10^{17}
" " "	B4405-3	7.3×10^{17}
Inter. Shell Plate	B4406-1	2.0×10^{19}
" " "	B4406-2	2.0×10^{19}
" " "	B4406-3	2.0×10^{19}
Lower Shell Plate	B4407-1	2.0×10^{19}
" " "	B4407-2	2.0×10^{19}
" " "	B4407-3	2.0×10^{19}

Attachment 6
page 7 of 9

TABL[illegible]

IDENTIFICATION AND LOCATION OF VESSEL BELTLINE REGION PLATE MATERIAL

					Heat Treatment		
Component	Plate No.	Heat No.	Mat'l. Spec. No.	Supplier	Austenitize	Temper	Stress Relief
ozzle Shell	B4405-1	C3594	A533B Cl. 1	Lukens	1600°F±50°F-4HR-WQ	1225°F±25°F-4HR-AC	1150°F±25°F-40HR-FC
" "	B4405-2	C3594	A533B Cl. 1	Lukens	"	"	"
" "	B4405-3	C3872	A533B Cl. 1	Lukens	"	"	"
nter. Shell	B4406-1	C1260	A533B Cl. 1	Lukens	"	"	"
" "	B4406-2	C3506	A533B Cl. 1	Lukens	"	"	"
" "	B4406-3*	C3506	A533B Cl. 1	Lukens	"	"	"
ower Shell	B4407-1	C3929	A533B Cl. 1	Lukens	"	"	"
" "	B4407-2	C3932	A553B Cl. 1	Lukens	"	"	"
" "	B4407-3	C3929	A553B Cl. 1	Lukens	"	"	"

Surveillance Material same as Inter. Shell Plate B4406-3

TABLE 6

CHEMICAL COMPOSITION OF VESSEL BELTLINE REGION PLATE MATERIAL

	Weight Percent							
Plate No.	C	Mn	P	S	Si	Ni	Mo	Cu
B4405-1	.21	1.42	.007	.018	.26	.46	.47	.14
B4405-2	.20	1.41	.006	.018	.25	.45	.47	.14
B4405-3	.24	1.30	.008	.013	.30	.48	.46	.14
B4406-1	.25	1.17	.016	.025	.29	.52	.49	.12
B4406-2	.24	1.41	.008	.015	.28	.50	.47	.15
B4406-3	.21	1.40	.009	.015	.25	.49	.46	.15
B4407-1	.21	1.35	.010	.014	.29	.55	.53	.14
B4407-2	.20	1.25	.012	.014	.22	.59	.54	.12
B4407-3	.22	1.32	.010	.014	.24	.50	.55	.14
B4406-3*	.24	1.40	.009	.015	.25	.49	.46	.14

* Surveillance Plate Analysis Performed by Westinghouse

Attachment 6
page 8 of 9

TABLE 7

MECHANICAL PROPERTIES OF VESSEL BELTLINE REGION PLATE MATERIAL

Plate No.	T_{NDT} °F	RT_{NDT}* °F	Shelf Energy Ft-Lbs MWD	NMWD*	YS KSI	UTS KSI	Elong. %	RA %
B4405-1	10	2	134	87	56.3	81.3	29.5	68.1
B4405-2	0	34	142	92	62.9	85.8	28.5	66.8
B4405-3	0	40	123.5	80	64.4	86.4	25.5	66.5
B4406-1	-10	- 8	123	80	63.3	86.3	27.0	67.1
B4406-2	-10	17	124	80.5	67.2	89.7	26.2	68.0
B4406-3	-10	27	121	78.5	66.8	88.8	26.2	68.0
B4407-1	-20	5	133	85.5	64.1	86.7	28.0	69.6
B4407-2	-20	-15	149	97	62.1	84.1	27.2	70.6
B4407-3	0	0	139	90.5	63.7	86.4	27.2	69.7

* Estimated from Data in the Major Working Direction (MWD) per NRC Standard Review Plan Section 5.3.2

TABLE 8

MECHANICAL PROPERTIES OF SURVEILLANCE PLATE & OTHER BELTLINE PLATES PERFORMED BY WESTINGHOUSE

Plate No.	T_{NDT} °F	RT_{NDT} °F	Shelf Energy Ft-Lbs MWD	NMWD	YS KSI	UTS KSI	Elong. %	RA %
B4406-1	---	5	---	83	--	--	--	--
B4406-2	---	33	---	96	--	--	--	--
B4406-3	---	40	130	98	68.4	90.4	27.5	70.0
B4407-1	---	28	---	103	--	--	--	--
B4407-2	---	-12	---	126	--	--	--	--
B4407-3	---	38	---	108	--	--	--	--

Attachment 6
page 9 of 9

QUESTION 121.2

Provide the following information for the pressure vessel:

1. A schematic of the reactor vessel showing all welds in the beltline region. Welds should be identified by a shop control number (such as a procedure qualification number) and the heat of filler metal, type and batch number of flux, etc.

2. For each of the above welds, and for welds in the vessel material surveillance programs, an identification of the welding process (sub arc, electroslag, manual metal arc, etc.). Also, a listing of the following information on each of these welds: chemical composition (particularly Cu, P and S content), drop weight T_{NDT}, RT_{NDT}, upper shelf Charpy energy and tensile properties.

3. The maximum end of life fluence at the vessel I.D. for each weld in the beltline.

Reference

NRC letter dated May 20, 1977 to Mr. John Tillinghast, Vice President, Indiana and Michigan Electric Company on the above subject and additional requested information.

ANSWER

For Donald C. Cook Unit 2 reactor vessel the response to the above question and to the additional requested information in the referenced letter is provided below:

1. Not Applicable.

2. Not Applicable.

3. Chicago Bridge and Iron.

4. a. A sketch of the reactor vessel showing all material welds in the beltline region is shown in Figure 1.

   b. Information relative to each of the welds in the beltline region is shown in Tables 1 through 4.

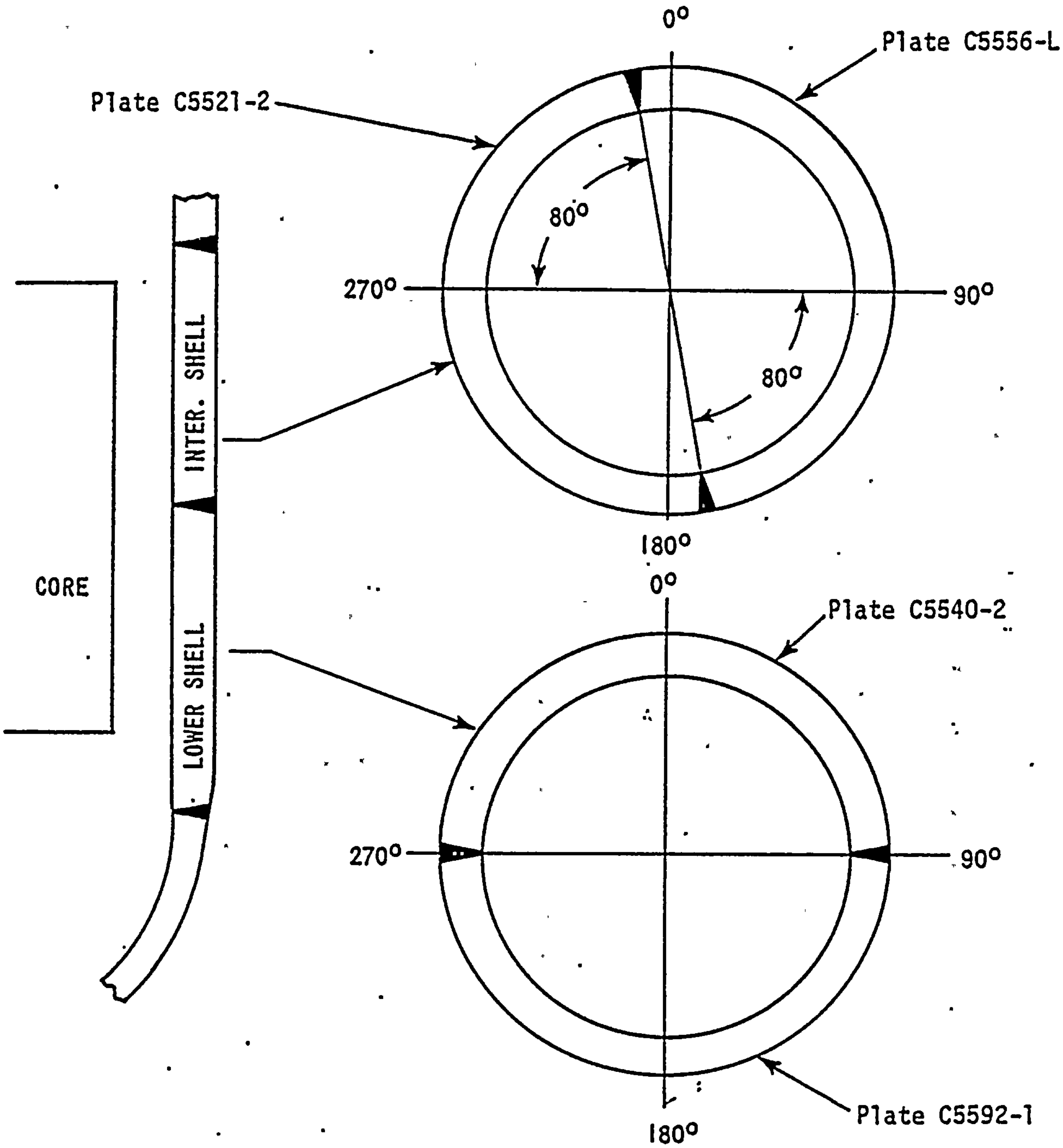
Attachment 7
page 2 of 10

c. Information relative to each of the plates in the beltline region is shown in Tables 4 through 7.

5. Information relative to the weld and plate material included in the vessel material surveillance program is shown in Tables 1 through 3 and 5 through 7.

Figure Q121.2-1

Reactor Vessel Beltline Region Welds (D. C. Cook Unit 2)



WELD ORIENT.	WELD LOCATION	MAX. END OF LIFE FLUENCE N/cm^2
VERTICAL	170° & 350°	7.7×10^{18}
VERTICAL	90° & 270°	6.3×10^{18}
CIRCUMFERENTIAL	INTER. TO LOWER SHELL	2.0×10^{19}

TABLE 1

IDENTIFICATION OF REACTOR VESSEL BELTLINE REGION WELD MATERIAL

	Welding Process	Weld Qual. No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Post Weld Heat Tr.
Inter. Shell (Vertical Seams)	Sub. Arc*	WPS-1323-2F4F6	ADCOM INMM	S3986	LINDE 124	934	1125-1150°F-62 1/2 HRS-FC
Inter. to Lower Shell (Circle Seam)	"	"	"	"	"	"	" " "
Lower Shell (Vertical Seams)	"	"	"	"	"	"	" " "
Surveillance Weld	"	"	"	"	"	"	1115-1165°F-9 HRS-FC

*Welds fabricated using both single and tandem wires

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page 4 of 10

TABLE 2

BELTLINE REGION WELD MATERIAL CHEMICAL COMPOSITION

WELD WIRE		FLUX			WEIGHT PERCENT								
TYPE	HEAT NO.	TYPE	LOT NO.		C	Mn	P	S	Si	Ni	Mo	Cr	Cu
ADCOMINMM	S3986	Line 124	934	(Single Wire)	.080	1.42	.019	.016	.36	.96	–	.07	.05
"	"	"	"	(Tandem Wire)	.092	1.46	.019	.015	.35	.97	.53	.07	.06
SURVEILLANCE WELD					.110	1.33	.022	.012	.44	.97	.54	.07	.055

Attachment 7
page 5 of 10

TABLE 3

MECHANICAL PROPERTIES OF BELTLINE REGION WELD MATERIAL

WELD WIRE TYPE	WELD WIRE HEAT NO.	FLUX TYPE	FLUX LOT NO.		T_{NDT} °F	RT_{NDT} °F	SHELF ENERGY FT-LBS	YS KSI	UTS KSI	ELONG %	RA %
ADCOMINMM	S3986	LINDE 124	934	(Single Wire)	–	27*	77*	71.8	86.5	30.0	68.6
"	"	"	"	(Tandem Wire)	–	27*	77*	74.7	91.2	25.5	66.0
SURVEILLANCE WELD					-40	27	77	76.3	92.3	24.2	66.7

*Estimated from surveillance weld data

121.2-6

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page 7 of 10

TABLE 4

MAXIMUM END-OF-LIFE FLUENCE AT INNER WALL REACTOR VESSEL LOCATIONS

	FLUENCE (n/cm^2)
Inter. Shell (Vertical Seams)	7.7×10^{18}
Inter. Shell to Lower Shell (Circle Seam)	2.0×10^{19}
Lower Shell (Vertical Seams)	6.3×10^{18}
Inter & Lower Shell Plates	2.0×10^{19}

TABLE 5

IDENTIFICATION OF BELTLINE REGION PLATE MATERIAL

COMPONENT	PLATE CODE NO.	HEAT NO.	MAT'L SPEC	SUPPLIER	HEAT TREATMENT
Inter. Shell	10-1	C5556-2	A533B,CL.1	LUKENS	1650-1750°F-5HR-WQ 1550-1650°F-4 3/4 HR-WQ 1200-1300°F-5HR-AC 1100-1175°F-62 1/2 HR-FC
Inter. Shell	10-2	C5521-2	A533B,CL.1	LUKENS	1650-1750°F-4 1/2 HR-WQ 1550-1650°F-5HR-WQ 1200-1300°F-4 1/2 HR-AC 1100-1175°F-62 1/2 HR-FC
Lower Shell	9-1	C5540-2	A533B,CL.1	LUKENS	1650-1750°F-4 1/2 HR-WQ 1550-1650°F-5HR-WQ 1200-1300°F-4 1/2 HR-AC 1100-1175°F-62 1/2 HR-FC
Lower Shell	9-2	C5592-1	A533B,CL.1	LUKENS	1650-1750°F-4 1/2 HR-WQ 1550-1650°F-4 1/2 HR-WQ 1200-1300°F-4 1/2 HR-AC 1100-1175°F-62 1/2 HR-FC
Surveillance	Plate	C5521-2	A533B,CL.1	LUKENS	1650-1750°F-4 1/2 HR-WQ 1550-1650°F-5HR-WQ 1200-1300°F-4 1/2 HR-AC 1125-1175°F- 51 1/2 HR-FC

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TABLE 6

CHEMICAL COMPOSITION OF BELTLINE REGION PLATE MATERIAL

PLATE CODE NO.	HEAT NO.	PLATE LOCATION	C	MN	P	S	Si	Ni	Mo	Cu
			WEIGHT PERCENT							
10-1	C5556-2	TOP	.24	1.34	.012	.015	.19	.56	.55	.14
"	"	BOT.	.21	1.38	.014	.014	.18	.58	.55	.15
10-2	C5521-2	TOP	.22	1.28	.012	.016	.18	.57	.54	.14
"	"	BOT.	.21	1.29	.013	.015	.16	.58	.50	.14
9-1	C5540-2	TOP	.21	1.31	.015	.014	.20	.64	.57	.11
"	"	BOT.	.19	1.34	.011	.015	.18	.63	.56	.10
9-2	C5592-1	TOP	.20	1.35	.010	.015	.19	.60	.53	.14
"	"	BOT.	.20	1.25	.012	.014	.18	.57	.50	.14
SURVEILLANCE PLATE			.22	1.28	.017	.014	.27	.58	.55	.11

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page 9 of 10

TABLE 7

MECHANICAL PROPERTIES OF BELTLINE REGION PLATE MATERIAL

PLATE CODE NO.	HEAT NO.	T_{NDT} °F	RT_{NDT} °F	SHELF ENERGY FT-LBS	YS KSI	UTS KSI	ELONG. %	RA %
10-1	C5556-2	0	58	90	67.2	87.3	25.5	–
10-2	C5521-2	10	38	86	64.5	85.2	25.5	–
9-1	C5540-2	-20	-20	110	65.8	85.7	26.5	–
9-2	C5592-1	-20	-20	103	70.0	88.1	24.5	–
SURVEILLANCE PLATE		10	38	86	66.4	86.6	25.2	60.6

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Page 10 of 10

Attachment 8
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INDIANA & MICHIGAN POWER COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

July 3, 1979
AEP:NRC:00097C

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
License No. DPR-58

Mr. James G. Keppler, Director
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

References: (1) NRC IE BULLETIN NOS. 78-12, 78-12A, 78-12B "ATYPICAL WELD MATERIAL IN REACTOR PRESSURE VESSELS"

(2) "COMBUSTION ENGINEERING REPORT IN COMPLIANCE WITH NRC IE BULLETIN 78-12, DATED JUNE 8, 1979

Dear Mr. Keppler:

This letter and its attachments are in response to the above referenced I.E. Bulletins as they apply to Unit 1 of the Donald C. Cook Nuclear Plant.

Combustion Engineering, manufacturer of the reactor vessel for Unit 1 has submitted to the NRC, on June 8, 1979, a generic report (reference 2) providing the required weld material information on all reactor vessels fabricated by them. Westinghouse and American Electric Power have reviewed the above referenced report and concluded that it represents adequately the data for the weldment material used in the reactor vessel of

Attachment 8
page 2 of 14

Unit 1 of the Donald C. Cook Nuclear Plant. Westinghouse has noted some discrepancies in the Combustion Engineering report. These are editorial in nature and will be submitted to the NRC as a revision by Combustion Engineering, Inc.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

Attachments:

1) Combustion Engineering letter to NRC dated June 8, 1979
2) Combustion Engineering review certification letter dated June 8, 1979
3) Westinghouse letter to AEP dated 6/25/79

cc: R. C. Callen
G. Charnoff
D. V. Shaller - Bridgman
R. S. Hunter
R. W. Jurgensen

Attachment 8
page 3 of 14

bc: S. J. Milioti/J. I. Castresana/T. Satyan
R. F. Hering/S. H. Steinhart/J. A. Kobyra
H. N. Scherer, Jr.
R. F. Kroeger
J. F. Stietzel - Bridgman
D. Wigginton - NRC
Cook Plant Region III Resident Inspector
AEP:NRC:00097C
R. C. Kopelow/J. R. Jensen
DC-N-6015.3.1

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel 203/688-1911
Telex 9-9297

Attachment 1
AEP:NRC:00097C

POWER
SYSTEMS

Attachment 8
page 4 of 14

June 8, 1979
LD-79-036

Mr. Harold D. Thornburg
Division of Reactor Construction Inspection
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: I&E Bulletin 78-12, "Atypical Weld Material in Reactor Pressure Vessel Welds"

Dear Mr. Thornburg:

Enclosed please find three (3) copies of a document entitled "Information Requested by I&E Bulletin 78-12, Atypical Weld Material in Reactor Pressure Vessel Welds."

This report is being submitted directly to the NRC by Combustion Engineering as permitted by Supplement A to the Bulletin. It is expected that holders of Construction Permits and Operating Licenses will reference this report in responding to the Bulletin on their individual dockets.

Should you have any questions, please feel free to call me or Mr. E. H. Kennedy of my staff at (203)688-1911, extension 2828.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Licensing Manager

AES:dag

Enclosure

C-E Power Systems
Combustion Engineering, Inc
911 W. Main Street
Chattanooga, Tennessee 37402

Tel. 615/265-4631

Attachment 2
AEP:NRC:00097C

Attachment 8
Page 5 of 14

POWER
SYSTEMS

June 8, 1979

I hereby certify that the record search required by I.E. Bulletin 78-12 and 78-12A has been completed and that, to the best of my knowledge and belief, the report submitted to the NRC on June 8, 1979, entitled, Information Requested by NRC Inspection and Enforcement Bulletin No. 78-12, "Atypical Weld Material in Reactor Pressure Vessels", addresses all of the applicable materials used in the fabrication of the following reactor vessel:

C-E Contract No.: 23366

Utility/Site: Indiana-Michigan Electric Co.

Donald Cook #1

W. A. Stone, Jr.

W. A. Stone, Jr., Manager
Nuclear Quality Assurance
Chattanooga Nuclear Operations

0 ; 1 J ; 3 ; (C 6) AEP:NRC:00097C

Attachment 8
page 6 of 14

Westinghouse Electric Corporation

Power Systems Company

Nuclear Service Division

Box 2728
Pittsburgh Pennsylvania 15230

June 25, 1979

AEP-79-17

Mr. J. R. Jensen
Mechanical Engineering Division
American Electric Power Service Corp.
2 Broadway
New York, NY 10004

Dear Mr. Jensen:

NRC IE BULLETINS #78-12 & #78-12A
"Atypical Weld Material in Reactor Pressure Vessel Welds"

Based upon our technical evaluation of the information contained in the generic report compiled by Combustion Engineering, Inc. to satisfy the requirements presented in the U.S. Nuclear Regulatory Commission IE Bulletins #78-12 & #78-12A, Westinghouse has concluded that the weld material data and other required information pertinent to the D.C. Cook Unit 1 reactor vessel are included in Combustion Engineering, Inc. report.

This report has previously been submitted to the U.S. Nuclear Regulatory Commission, as evidenced by Combustion Engineering, Inc. transmittal letter of June 8, 1979 to the US Nuclear Regulatory Commission, a copy of which is enclosed for your information.

Additionally, we have enclosed for your files a copy of Combustion Engineering, Inc. letter to Westinghouse, dated June 5, 1979 and attached certification stating that the generic report submitted to US Nuclear Regulatory contains data for the D.C. Cook Unit 1 reactor vessel.

Westinghouse audited the content of the subject report against the ASME Code and W E-Spec. requirements for the D.C. Cook Unit 1 reactor vessel built by Combustion Engineering Inc. The report contains data pertaining to the D.C. Cook Unit 1 react. vessel and is considered to be in compliance with the US NRC Bulletins and Westinghouse requirements. However, some apparent errors were noted in the report. These discrepancies were brought to the attention of Combustion Engineering, Inc. and Combustion Engineering, Inc. is currently evaluating them. They have agreed to resolve the comments to Westinghouse satisfaction and will submit revised pages for the report to the Nuclear Regulatory Commission and Westinghouse at a later date.

In addition to the data supplied by Combustion Engineering, Inc. in the subject report, Westinghouse has developed surveillance weldment data. This data is contained in the following report, which has previously been transmitted to you:

D.C. Cook Unit 1, WCAP-8047, dated March, 1973

As stated in their report Combustion Engineering, Inc. does not maintain archive material for the welds represented by this report. In addition, Westinghouse inventoried our archive surveillance weldment material and none exists for the D.C. Cook Unit 1 reactor vessel.

In conclusion, this letter provides assurance that the D.C. Cook Unit 1 reactor vessel is covered in the subject report, and fulfills Westinghouse's obligations relative to the Reactor Vessel Weld Material Program contracted for by American Electric Power Service Corporation.

A copy of the Combustion Engineering, Inc. generic report applicable to the D.C. Cook Unit 1 reactor vessel is submitted for your records.

Sincerely,

F. Noon, Manager
Eastern Region & WNI Support

JDC/ej
attachments

cc: D. V. Shaller*
R. W. Jurgensen*
J. G. Kern*

*without attachment

C-E Power Systems
Combustion Engineering, Inc.
911 W. Main Street
Chattanooga, Tennessee 37402.

Tel. 615/265-4631

Attachment 8
pg 8 of 14

~~Satyan of Nuclear Div. keeping Cook 2 book.~~

E.N July '81

got back 4/27/81 JRJ

POWER SYSTEMS

INFORMATION REQUESTED BY

NUCLEAR REGULATORY COMMISSION

INSPECTION & ENFORCEMENT

BULLETIN NO. 78-12

"ATYPICAL WELD MATERIAL IN REACTOR

PRESSURE VESSEL WELDS"

Attachment 8
pg 9 of 14

INFORMATION REQUESTED BY

NUCLEAR REGULATORY COMMISSION

INSPECTION & ENFORCEMENT

BULLETIN NO. 78-12

"ATYPICAL WELD MATERIAL IN REACTOR

PRESSURE VESSEL WELDS"

Prepared by

COMBUSTION ENGINEERING, INC.

NUCLEAR POWER SYSTEMS

June 8, 1979

Attachment 8

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REACTOR PRESSURE VESSELS
FABRICATED BY COMBUSTION ENGINEERING, INC.

C-E CONTRACT NO.	CUSTOMER	ASME CODE	OWNER	SITE
164	General Electirc	I & VIII, 1962	Niagara Mohawk	Nine Mile Point #1
264	General Electric	I & VIII, W-63	Jersey Central	Oyster Creek
17765	Westinghouse	III, W-65	Consolidated Edison Co.	Indian Point #2
19865	General Electric	III, S-65	Northeast Utilities	Millstone #1
2966A	CENPD - Windsor	III, 1965	Consumers Public Power	Palisades
3266	Westinghouse	III, W-65	Public Service of N. J.	Salem #1
3366	Westinghouse	III, W-65	Consolidated Edison Co.	Indian Point #3
6866	Westinghouse	III, W-65	Carolina P&L	Robinson #2
21366	General Electric	III, W-66	Consumers Public Power	Cooper Site
[illegible]66	General Electric	III, W-66	Boston Edison Co.	Pilgrim
21566	General Electric	III, W-66	Power Authority State N.Y.	Fitzpatrick
23066	Westinghouse	III, W-66	Pacific Gas & Electric	Diablo Canyon #1
23366	Westinghouse	III, W-66	Indiana-Michigan Elec. Co.	Donald Cook #1
71166	CENPD - Windsor	III, W-67	Omaha	Ft. Calhoun
2067	Westinghouse	III, W-66	Public Service of N. J.	Salem #2
2167	Westinghouse	III, S-71	Duke Power Company	McGuire #1
2667	General Electric	III, S-69	Detroit Edison	Fermi
2867	General Electric	III, W-69	Commonwealth Edison	LaSalle
3067	General Electric	III, S-68	Long Island Lighting Co.	Shoreham
3167	General Electric	III, W-66	Southern Services	Hatch #1
[illegible]167	CENPD - Windsor	III, W-67	Baltimore Gas & Electric	Calvert Cliff
73167	CENPD - Windsor	III, W-67	Baltimore Gas & Electric	Calvert Cliff
74167	CENPD - Windsor	III, W-67	Florida Power & Light	St. Lucie I

SUMMARY OF WELD MATERIALS AND TEST

WIRE/FLUX

WELDING MATERIALS						NUMBER AND DATES OF TESTS		C-E CODE NO.	REFER ATTACHED NON-CONFORM. REPORT
WIRE/ELECTRODE			FLUX			WIRE/FLUX OR ELECTRODE WELD DEPOSIT TEST PLATES			
VENDOR	TYPE	HEAT/LOT NO.	VENDOR	TYPE	LOT NO.	NO. OF TESTS	DATE(S)		
ADCOM	HMM	12008	LINDE	1092	3947	1)	4-1-70	M1.37	
RACO 3	HMM	305414	LINDE	1092	3947	1)		M1.37	
RACO 3	HMM	33A277	LINDE	1092	3947	1	4-8-70	M1.38	
Reid-Avery	HMM	305424	LINDE	1092	3947	1	4-10-70	M1.39	
Reid-Avery	HMM	305414	LINDE	1092	3951	1	5-4-70	M1.40	
ADCOM	HMM	12008	LINDE	1092	3951	1)	5-11-70	M1.41	
Reid-Avery	HMM	305414	LINDE	1092	3951	1)		M1.41	
Reid-Avery	HMM	305414	LINDE	1092	3958	1	6-2-70	M1.42	
Reid-Avery	HMM	1P3571	LINDE	1092	3958	1	NA	M1.42	
Reid-Avery	HMM	1P3571	LINDE	1092	3958			M1.43	
Reid-Avery	HMM	1P3571	LINDE	1092	3958	1	6-9-70	M1.43	
Reid-Avery	HMM	1P3571	LINDE	1092	3958	1	6-3-70	M1.44	
Reid-Avery	HMM	305414	LINDE	1092	3958	1	6-3-70	M1.44	
ADCOM	HMM	27204	LINDE	124	3687	1)	7-11-67	E1.01	
NA	HMM	51989	LINDE	124	3687	1)		E1.01	
ADCOM	HMM	27204	LINDE	124	3687	1	10-10-67	E1.02	
Reid-Avery	HMM	34B009	LINDE	124	3687	1	2-28-68	E1.03	
Reid-Avery	HMM	34B009	LINDE	124	3688	1	2-7-69	E1.04	
NA	HMM	A-8746	LINDE	124	3688	1	5-7-69	E1.05	
NA	HMM	A-8746	LINDE	124	3878	1	9-10-69	E1.06	
Reid-Avery	HMM	33A277	LINDE	124	3878	1	10-29-69	E1.07	

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WIRE/FLUX INDEX

Heat of Wire	Flux Type	Lot	Test Results
646B428	Linde 80	8174	Page 1
661H577	Linde 80	8174	Page 1
86054-B	Arcos B-5	4D4F 4D5F	Page 2
1248	Arcos B-5	4K13F	Page 3
5458	Linde 80	8208	Page 4
W-5214	Arcos B-5	5G13F	Page 5
39B196	Linde 1092	3617	Page 6
34B009	Linde 80	8405	Page 7
27204	Linde 1092	3724	Page 8
12420	Linde 1092	3724	Page 8
13253	Linde 1092	3724	Page 9
13253 & 12008	Linde 1092	3774	Page 10
20291	Linde 1092	3791	Page 11
7114	Linde 1092	3833	Page 12
8746	Linde 1092	3854	Page 13
IP2809	Linde 1092	3854	Page 14
IP2815	Linde 1092	3854	Pages 15 & 16
21935	Linde 1092	3869	Pages 17 thru 19
33A277	Linde 1092	3869 & 8651	Pages 20 & 21
305424	Linde 1092	3889	Pages 22 & 23
305414	Linde 1092	3947	Pages 24 & 25
IP3571	Linde 1092	3958	Pages 26 & 27
885T40	Linde 0091	3922	Pages 28 & 29
90099	Linde 0091	3922	Pages 30 & 31
35C191	Linde 0091	3922	Page 32
90136	Linde 0091	3977	Pages 33 & 34
10120	Linde 0091	3999	Pages 35 & 36
10137	Linde 0091	3999	Pages 37 & 38
6329637	Linde 0091	3999	Pages 39 & 40
51874	Linde 0091	3458	Pages 41 & 42
51876	Linde 0091	3458	Pages 43 & 44
51907	Linde 0091	3458	Pages 45 & 46
606L40	Linde 0091	3489 & 3458	Pages 47 thru 49
51922	Linde 0091	3489	Pages 50 & 51
51923	Linde 0091	3489	Pages 52 & 53
51912	Linde 0091	3490	Pages 54 thru 56
3P4767	Linde 0091	3490	Pages 57 & 58
83640	Linde 0091	3490	Pages 59 & 60
83642	Linde 0091	3536	Pages 61 & 62
83653	Linde 0091	3536	Pages 63 & 64
83648	Linde 0091	3536	Pages 65 & 66
4P5174	Linde 0091	1122	Pages 67 & 68
83637 & 83650	Linde 0091	1122	Pages 69 thru 71
5P5622	Linde 0091	1122	Pages 72 & 73
83646	Linde 0091	1122	Pages 74 & 75
2P5755	Linde 0091	1122	Pages 76 & 77
4P6052	Linde 0091	0145	Pages 78 & 79
87005	Linde 0091	0145	Pages 80 & 81
87600	Linde 0091	0145	Pages 82 & 83
88118	Linde 0091	0145	Pages 84 & 85

ft

SUBJECT	FROM — DATE
Welding Material Qualification to Requirements of ASME Section III A-32255 810560	Metallurgical Research and Development Department Chattanooga June 9, 1970

The following test data is for 3/16" diameter bare wire, type B-4 MOD., heat number 1P3571 (tandem), flux type 1092, lot number 3958.

A weld deposit was made using the above heat of wire and lot of flux. Welding was done in accordance with C. E. Welding Procedure Specification SAA-33-H3. The completed weldment was given a post weld heat treatment of 1150°F ±25°F for 40 hours and furnace cooled to 600°F.

Charpy V-Notch Impacts

Test Code	Ft/Lbs. @ + 10°F	Requirements
VZ	79, 68, 64	30 Ft./Lbs. @ +10°F

All Weld Metal .505 Tensile

Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation in 2" %	Reduction of Area %
70.5	86.8	27.0	67.0

CEB:sl

Attachment 8

DATE 6-19-70 pg 14 of 14

INFO

CHEMICAL ANALYSIS OF WIRE-FLUX
TEST WELD COUPON

SAMPLE NO.						
LAB NO.	D8698					
TYPE WIRE	B-4 MOD.					
SIZE WIRE	3/16"					
HEAT NO.	1P3571 TANDEM					
[illegible]	1092					
LOT NO.	3958					
SI	.23					
S	.011					
P	.017					
MN	1.31					
C	.12					
MO	.51					
CU	.37					
NI	.75					

Attachment 9
pg 1 of 19

INDIANA & MICHIGAN POWER COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

June 1, 1979
AEP:NRC:00097

Donald C.Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License Nos. DPR-74

Mr. James G. Keppler, Director
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

References: (1) NRC IE BULLETIN NOS. 78-12, 78-12A, 78-12B ATYPICAL WELD MATERIAL IN REACTOR PRESSURE VESSELS

(2) "CHICAGO BRIDGE & IRON COMPANY REPORT IN COMPLIANCE WITH THE NRC BULLETINS 78-12 AND 78-12A", DATED APRIL 24,1979

Dear Mr. Keppler:

This letter and its attachments are in response to the above referenced I.E. Bulletins as they apply to Unit No. 2 of the D.C. Cook Nuclear Plant.

Chicago Bridge & Iron (CB&I), manufacturer of the reactor vessel for Unit 2, has submitted to the NRC, on April 24, 1979, a generic report (reference 2) providing the required weld material information on all reactor vessels fabricated by CB&I. Westinghouse and American Electric Power have reviewed the above referenced report and concluded that it represents adequately the data for the weldment material used in the reactor vessel of Unit No. 2 of the Donald C. Cook Nuclear Plant. Weldment material that might be used for verification purposes, is available in the archives of the Westinghouse Electric Corporation.

As stated in our letter No. AEP:NRC:00097B dated May 21, 1979, the above information for Donald C. Cook Unit No. 1 reactor vessel will be submitted by July 2, 1979

Very truly yours

John E. Dolan

JED:em

John E. Dolan
Vice President

Attachments:

1) CB&I review certification letter to the NRC dated 4/24/79
2) CB&I letter to the NRC dated 4/24/79
3) Westinghouse letter to AEP dated 5/23/79

cc: R. C. Callen
G. Charnoff
D. V. Shaller-Bridgman
R. W. Jurgensen

bc:S.J. Milioti/J. I. Castresana/T.Satyan
R. F. Hering/S. H. Steinhart
H. N. Scherer, Jr.
R. F. Kroeger
J. F. Stietzel-Bridgman
D. Wigginton-NRC
Cook Plant Region III Resident Inspector
AEP:NRC:00097
DC-N-6015.3-1
R. C. Kopelow/J. Jensen

Attachment 9
pg 4 of 19

ATTACHMENT I

Chicago Bridge & Iron Company

8900 Fairbanks north Houston road
p o box 40066
Houston, Texas 77040

telephone 713. 463 7581

The documentation and information required by NRC Bulletins 78-12 and 78-12A, and Westinghouse PO #546-MVC-401945-MN for

CBI Contract # 68-3262

Vessel D. C. Cook II

are contained in the attached report.

Welding consumables were re-reviewed against the original requirements in accordance with the above listed documents. No deviations were found.

Based upon our records, I certify, to the best of my knowledge, this report is correct.

R E Kelley 4-24-79

Ralph E. Kelley Date
Manager, CQA Services

ATTACHMENT 2

Chicago Bridge & Iron Company

8900 Fairbanks north Houston road
p o box 40066
Houston, Texas 77040

telephone 713. 466 7531

April 24, 1979

Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. G. W. Reinmuth

RE: NRC BULLETINS 78-12 & 78-12A

Gentlemen:

In accordance with the above listed Bulletins and requirements from Westinghouse and General Electric, enclosed is one copy of our report.

This report includes information from all completed Reactor Vessels constructed by Chicago Bridge & Iron Co.

Very truly yours,

CHICAGO BRIDGE & IRON CO.

R E Kelley

Ralph E. Kelley, Manager
CQA Services
Houston Operations

REK:mks

Enclosure

0 0 1 0 0 3 1 0 0

ATTACHMENT 3

Attachment 9
Pg 6 of 19

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Service Division

Box 2728
Pittsburgh Pennsylvania 15230

May 23, 1979
AEP-79-10

Received 5/29/79

Mr. J. R. Jensen
Mechanical Engineering Division
American Electric Power Service Corp.
2 Broadway
New York, NY 10004

Dear Mr. Jensen:

NRC IE Bulletins #78-12 & #78-12A
"Atypical Weld Material in Reactor Pressure Vessel Welds"

Based upon our technical evaluation of the information contained in the generic report compiled by Chicago Bridge & Iron Company to satisfy the requirements presented in the U.S. Nuclear Regulatory Commission IE Bulletins #78-12 and #78-12A, Westinghouse has concluded that the weld material data and other required information pertinent to the D.C. Cook Unit 2 reactor vessel are included in Chicago Bridge & Iron's report.

This report has previously been submitted to the U.S. Nuclear Regulatory Commission, as evidenced by Chicago Bridge & Iron Company's transmittal letter of April 24, 1979 to the U.S. Nuclear Regulatory Commission, a copy of which is enclosed for your information.

Additionally, we have enclosed for your files a copy of Chicago Bridge & Iron Company's letter to Westinghouse, dated April 24, 1979, providing further confirmation that the generic report prepared by vendor includes records pertaining to the D.C. Cook Unit 2 reactor vessel. The Chicago Bridge & Iron certifications stating that the report contains data for the D.C. Cook Unit 2 reactor vessel is included in Part 2 of the report.

Westinghouse audited the subject report against the ASME and W E-Spec. requirements for the D.C. Cook Unit 2 reactor vessel built by Chicago Bridge & Iron. The report contains data pertaining to the D.C. Cook Unit 2 reactor vessel and is considered to be in compliance with the U.S. Nuclear Regulatory Commission bulletins and Westinghouse requirements.

In addition to the data supplied by Chicago Bridge & Iron Company in the subject report, Westinghouse has developed surveillance weldment data. This data is contained in the following report, which has previously been transmitted to you:

D.C. Cook Unit 2, WCAP-8512, dated November, 1975

Attachment 9
pg 7 of 19

As stated in their report Chicago Bridge & Iron Company has no archive material for the welds represented by this report. Westinghouse inventoried our archive weldment material which could be used for verification purposes on the D.C. Cook Unit 2 reactor vessel. This material consists of one full thickness weldment made up of weld wire from heat number 53986 and Linde Flux 124 from lot number 934.

In conclusion, this letter provides assurance that the D.C. Cook Unit 2 reactor vessel is covered in the subject report, and fulfills Westinghouse's obligations relative to the Reactor Vessel Weld Material Program contracted for by American Electric Power Service Corporation.

A copy of the Chicago Bridge and Iron generic report applicable to the D.C. Cook Unit 2 is submitted for your records.

Sincerely,

F. Noon, Manager
Eastern Service Region

JDC/pl
Attachments
cc: D.V. Shaller
R.W. Jurgensen
J.G. Kern

Attachment 9
Pg 8 of 19

39.60.26.1

Chicago Bridge & Iron Company

8900 Fairbanks north Houston road
p o box 40066
Houston, Texas 77040

telephone 713. 466 7531

CHICAGO BRIDGE & IRON COMPANY

REPORT IN COMPLIANCE WITH THE

NUCLEAR REGULATORY COMMISSION

BULLETINS 78-12 & 78-12A

Report prepared by R E Kelley 4-24-79
Ralph E. Kelley
Mgr., CQA Services

Date

PART I

LIST OF REACTOR VESSELS INCLUDED

WESTINGHOUSE VESSELS

CBI CONTRACT	VESSEL
68-3262	D C Cook II
68-3780	Trojan
71-2631	Virgil C. Summer I
71-2632	Shearon Harris I
71-2633	Shearon Harris II

GENERAL ELECTRIC VESSELS

9-5624	Monticello
9-6201	Vermont Yankee
68-2471	Brunswick I
68-2472	Brunswick II
68-3331	Susquehanna I
68-3332	Susquehanna II
69-2967	Duane Arnold
69-4824	Quad Cities II (CBI Portion)
69-4962	Peach Bottom II (CBI Portion)
69-5128	Peach Bottom III (CBI Portion)
69-5401	Limerick I
69-5402	Limerick II
69-5571	Zimmer I
73-6735	Clinton I

Attachment 9.5
Pg 10 of 19

Chicago Bridge & Iron Company

CBI

8900 Fairbanks north Houston road
p o box 40066
Houston, Texas 77040

telephone 713. 466 7581

The documentation and information required by NRC Bulletins 78-12 and 78-12A, and Westinghouse PO #546-MVC-401945-MN for

CBI Contract # 68-3262

Vessel D. C. Cook II

are contained in the attached report.

Welding consumables were re-reviewed against the original requirements in accordance with the above listed documents. No deviations were found.

Based upon our records, I certify, to the best of my knowledge, this report is correct.

R E Kelley 4-24-79

Ralph E. Kelley Date
Manager, CQA Services

Attachment 9
Pg 11 of 19

CBI NUCLEAR RECORD INDEX

Document Number	Number of Pages	DESCRIPTION: WIRE / FLUX	WIRE SIZE	WIRE HEAT NO.	FLUX RUN OR LOT	TEST NO.	SPECIFICATIONS
1	1	INMM-124	3/16	12088	859	PT 150	APPENDIX #7
2	2	"	"	1P4217	989	PT 269	" #7
3	1	"	"	1P4218	989	WO 168D	" #7
4	1	"	"	"	"	569C	" #7
5	1	"	"	3994	934	PT 244	" #7
6	2	"	"	3P4000	989	WO 14D	" #7
7	2	"	"	"	"	425D	" #7
8	1	"	"	S3986	934	PT 203A	" #7
9	2	"	"	"	"	WO 377C	" #7
10	2	"	"	"	"	PT 200	" #7
11	2	"	"	1P4217	1214	PT 320 B2B/B1B	" #7
12	2	"	"	3998	0331	PT 411 A1/A2	" #7
13	2	"	"	4P4724	1214	371E	" #8
14	4	"	"	"	989	PT 321 A/B	" #6, 7, & 8
15	2	"	"	3P4966	1214	375E	" #8
16	2	"	"	"	"	670E	" #9
17	2	"	"	"	0331	CN-49	" #9
18	5	"	"	5P2771	0342	CN-161	" #6 & 9
* 19	2	"	"	3P4955	0331	PT 411 B1/B2	" #7
20	2	"	"	"	"	CN-15	" #9
21	4	"	"	3P4755	1214	678E/677E	" & 9
22	2	"	"	5P6214E	0331	83F-5/7	" #9
23	6	"	"	5P5657	0931	CN-68	" #10
24	6	"	"	5P7391	0342	CN-224-5/7	" 9 & 10
25	6	"	"	3P4955	1214	PT 375 A/E	" #6, 7, 8 & 9
26	6	"	"	"	3476	PT 556 A/B	" #10

COPIES of documents covered by this index are certified to be true copies

Date ______

Signature ______

COOK PLANT
MED RECORD - MED COPY
SECTION ______
ENGINEER ______
DATE ______
☐ PLANT LIFETIME
DATE TO PLANT
☐ NON PERMANENT

* See Part 5 Note 3

______ Office Code

______ Classification

Page ____ of ____

______ Contract Number

Folder ____ of ____

GO 1152 JAN 75

Attachment 9
pg 12 of 19

CHICAGO BRIDGE & IRON COMPANY

1500 N. 50TH ST. P.O. BOX 277, BIRMINGHAM, ALABAMA 35202

TWX 810-733-3654
Western Union-WUX
Area Code: 205 595-1191

CERTIFICATE OF ANALYSIS

PURCHASE ORDER NUMBER:

Test Number: PT 200 A
Type Electrode: Adcom 1NMM/Linde 124
Trade Name: Adcom 1NMM Wire
Diameter: 3/16"
Flux Lot Number: 3877-Run 934-Linde 124
Wire Heat Number: S3986

CHEMICAL TESTS

Carbon. . .	.101
Manganese. .	1.49
Chromium. . .	.12
Nickel. . . .√	.92
Silicon. . .√	.41
Columbium. .	.004
Tantalum. . .	--
Molybdenum. . .	.53
Tungsten. . .	--
Copper. . .	.05
Titanium. . .	--
Phosphorus. . .	.022
Sulfur. . . .	.016
Vanadium. . .	--
Iron. . .	--
Schaeffler Ferrite. .	
Cobalt . . .	.033

MECHANICAL TESTS

Heat Treatment 50 Hours @ 1125/1150° Farenheit

Tensile Properties @ Room Temp.
Type: .505" ø
UTS 89,000 PSI
YLP 70,100 PSI
% Elongation in 2 inches = 23.5
% Reduction of Area: = 65

Impact Properties
Type: Charpy Vee Notch
Orientation: ⊥ To Weld Direction
Test Temperature +10° F
Foot- lbs. 67.5, 67.5, 65
% Shear 60, 60, 55
Lateral Expansion 61, 58, 52

This material conforms to Section III of the ASME CODE, Paragraph N511.3.

CHICAGO BRIDGE AND IRON COMPANY
Birmingham Materials Laboratory

Approved by W F Ricketts

By Rick Boone Date 6-4-70
In charge of Testing for Materials Evaluation

Materials Engineer

Date 6/4/70

8

Attachment 9
pg 13 of 19
2078

CHICAGO BRIDGE & IRON COMPANY

P. O. BOX 13308, MEMPHIS, TENNESSEE 38113

901 947-3111

CERTIFICATE OF ANALYSIS

Purchase Order Number: M30506-3262/3780
Test Number: WO #337C (Tandem Wire)
Type Electrode: Adcom 1NMM/Linde 124 (20 x 150) Flux
Trade Name: Adcom 1NMM
Electrode Diameter: 3/16"
Lot Number: -
Heat Number: S3986
Flux Batch Number: Run 934 Lot 3878

CHEMICAL TEST RESULTS

Carbon........	.089
Manganese.....	1.47
Chromium......	.11
Nickel...... √	.90
Silicon...... √	.47
Columbium.....	
Tantalum......	
Molybdenum....	.53
Tungsten......	
Copper........	.06
Titanium......	
Phosphorus....	.028
Sulfur........	.014
Vanadium......	
Iron..........	
Schaeffler Ferrite..	

MECHANICAL TEST RESULTS

Heat Treatment 1150°F +25°-50°F for 62 1/2 Hours

Tensile Properties

Type: .505"ø
UTS 92,000 PSI
YLP 78,800 PSI
% Elongation in 2 inches = 26%
% Reduction of Area = 57.3

Impact Properties

Type: Charpy Vee Notch
Orientation: ⊥ to Weld Direction
Test Temperature +10°F
Foot - Lbs. 39, 53, 38
Lateral Expansion 36, 44, 35
% Shear 40, 50, 40

This material conforms to SECTION III of the ASME CODE, Paragraph N511.3

CHICAGO BRIDGE & IRON COMPANY

BY R. A. [signature] DATE JUNE 15 1970

9

Telecopy to → CHARLES LETCHMAN
c/o RAndy KELLY
CBI Houston

1078

CHICAGO BRIDGE & IRON COMPANY

P. O. BOX 13308, MEMPHIS, TENNESSEE 38113

901 947-3111

Attachment 9
pg 14 of 19

CERTIFICATE OF ANALYSIS

Purchase Order Number: M30506-3262/3780

Test Number: WO #337C (Single Wire)

Type Electrode: Adcom 1NMM/Linde 124 (20 x 150) Flux

Trade Name: Adcom 1NMM

Electrode Diameter: 3/16"ø

Lot Number: -

Heat Number: S3986

Flux Batch Number: Run 934
Lot 3878

CHEMICAL TEST RESULTS

Carbon........	.076
Manganese.....	1.44
Chromium......	.10
Nickel......√.	.81
Silicon......√	.46
Columbium.....	
Tantalum......	
Molybdenum....	.50
Tungsten......	
Copper........	.06
Titanium......	
Phosphorus....	.026
Sulfur........	.017
Vanadium......	
Iron..........	
Schaeffler Ferrite..	

MECHANICAL TEST RESULTS

Heat Treatment 1150°F +25°-50°F for 62 1/2 Hours

Tensile Properties

Type: .505"ø

UTS 89,500 PSI

YLP 74,300 PSI

% Elongation in 2 inches = 27%

% Reduction of Area = 67%

Impact Properties

Type: Charpy Vee Notch

Orientation: _ to Weld Direction

Test Temperature +10°F.

Foot - Lbs.	50, 49, 62
Lateral Expansion	45, 44, 53
% Shear	35, 35, 40

This material conforms to SECTION III of the ASME CODE, Paragraph N511.3

CHICAGO BRIDGE & IRON COMPANY

BY R. A. Kennedy DATE JUNE 15, 1976

...R 26 1979

9

Attachment 9
Pg 15 of 19

CHICAGO BRIDGE & IRON COMPANY

1500 N 50TH ST. P.O. BOX 277, BIRMINGHAM, ALABAMA 35202

TWX 810-733-3654
Western Union-WUX
Area Code: 205 595-1191

CERTIFICATE OF ANALYSIS

PURCHASE ORDER NUMBER:

Test Number: PT#200-Single Wire
Type Electrode: Adcom Inmm/Linde 124
Trade Name: Adcom Inmm Wire
Diameter: 3/16"Ø
Flux Lot Number: 3876- Run 934- Linde 124
Wire Heat Number: S-3986

CHEMICAL TESTS

Carbon. . . . 0.080
Manganese. . 1.42
Chromium. . . 0.07
Nickel. ✓. . . 0.96
Silicon. ✓. . 0.36
Columbium. .
Tantalum. . .
Molybdenum... 0.52
Tungsten. . .
Copper. . . . 0.05
Titanium. . .
Phosphorus... 0.019
Sulfur. . . . 0.016
Vanadium. . .
Iron. .
Schaeffler Ferrite. .

MECHANICAL TESTS

Heat Treatment 62-1/2hours@1125/1150

Tensile Properties At Room Temperatur
Type: 0.505"Ø
UTS 86,500
YLP 71,800
% Elongation in 2 inches = 30.0%
% Reduction of Area. = 68.6%

Impact Properties
Type: Charpy Vee Notch
Orientation: ⊥ to Weld Direction
Test Temperature Plus 10°F
Foot- lbs. 46-51-49
% Shear 40-40-40
Lateral Expansion 38-44-43

This material conforms to Section III of the ASME CODE, Paragraph N511.3

CHICAGO BRIDGE AND IRON COMPANY
Birmingham Materials Laboratory

By Harold Gray Date 5-12-69
In charge of Testing for Materials Evaluation

10

CHICAGO BRIDGE & IRON COMPANY
1500 N 50TH ST. P.O BOX 277, BIRMINGHAM, ALABAMA 35202

TWX 810-733-3654
Western Union-WUX
Area Code 205 595-1191

CERTIFICATE OF ANALYSIS

PURCHASE ORDER NUMBER:

Test Number: PT #200-Tandem Wire
Type Electrode: Adcom Inmm/Linde 124
Trade Name: Adcom Inmm Wire
Diameter: 3/16"Ø
Flux Lot Number: 3876- Run 934- Linde 124
Wire Heat Number: S-3986

CHEMICAL TESTS

Carbon. . . .	0.092
Manganese. .	1.46
Chromium. . .	0.07
Nickel. . . .	0.97
Silicon. . .	0.35
Columbium. .	
Tantalum. . .	
Molybdenum. . .	0.53
Tungsten. . .	
Copper. . . .	0.06
Titanium. . .	
Phosphorus . . .	0.019
Sulfur. . . .	0.015
Vanadium. . .	
Iron. .	
Schaeffler Ferrite. .	

MECHANICAL TESTS

Heat Treatment 62-1/2 hours @1125/1150°F

Tensile Properties At Room Temperature
Type: 0.505"Ø
UTS 91,200
YLP 74,700
% Elongation in 2 inches=25.5%
% Reduction of Area. =66.0%.

Impact Properties
Type: Charpy Vee Notch
Orientation: ⊥ to Weld Direction
Test Temperature Plus 10°F
Foot- lbs. 41-45-46
% Shear 50-55-55
Lateral Expansion 49-44-41

This material conforms to Section III of the ASME CODE, Paragraph N511.3

CHICAGO BRIDGE AND IRON COMPANY
Birmingham Materials Laboratory

By HAROLD GRAY Date 5-12-69
In charge of Testing for Materials Evaluatio

Attachment 9
pg 17 of 19

CBI NUCLEAR RECORD INDEX

Document Number	Number of Pages	DESCRIPTION BARE WIRE AND FLUX CORED WIRE Type Electrode	Size	Heat No.	Lot No.	Specifications
1	2	ADCOM INMM	3/32	S3986	—	APPENDIX # 14
2	3	OXWELD-65	3/32	915L65	—	" # 15
3	2	OXWELD-65	3/32	L40165	—	" # 16
4	3	RACO INMM	3/32	5P6756	—	" # 17

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Signature ______________________

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Office Code

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Classification

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Contract Number

Page 1 of 1

Folder 4 of ____

Attachment 9
pg 18 of 19

CHICAGO BRIDGE & IRON COMPANY

P. O. BOX 13308, MEMPHIS, TENNESSEE 38113

CERTIFICATE OF ANALYSIS

901 947-3

Purchase Order Number:

Test Number: LS 1016 & W.O. 12D

Type Electrode: GTA Filler Metal

Trade Name: ADCOM 1NMM

Electrode Diameter: 3/32

Lot Number:

Heat Number: S3986

Flux Batch Number:

Shielding Gas: Argon

CHEMICAL TEST RESULTS

Carbon........	.081
Manganese.....	2.0
Chromium......	.08
Nickel........	.97
Silicon.......	.03
Columbium.....	
Tantalum......	
Molybdenum....	.48
Tungsten......	
Copper........	.09
Titanium......	
Phosphorus....	.015
Sulfur........	.014
Vanadium......	
Iron..........	
Schaeffler Ferrite..	

MECHANICAL TEST RESULTS

Heat Treatment

62 1/2 Hours at 1150° +25°-50°F

Tensile Properties

Type: 0.505"ø

UTS 95,700 psi

YLP 95,200 psi

% Elongation in 2 inches = 24%

% Reduction of Area = 66.1%

Impact Properties

Type: Charpy Vee Notch

Orientation: _ to Weld Direction

Test Temperature -20°F

Foot - Lbs. 123,92,158

% Shear 100,100,100

Lateral Expansion 81,73,82

This material conforms to SECTION III of the ASME CODE, Paragraph N511.3

CHICAGO BRIDGE & IRON COMPANY KS 4/5/71

BY [signature] DATE 12/12/69

REVISED 4/5/71

1

Attachment 9
pg 19 of 19

MANUFACTURERS OF TECHNICALLY CONTROLLED WIRE, ROD NICKEL, INCONEL, INCONEL X, INCOLOY, ADCOM STAINLESS STEELS, ALLOY COLD HEADING STEELS, HIGH ALLOY STEELS, LOW ALLOY STEELS, WELDING ALLOYS, LOW, MED, & HIGH WELDING ELECTRODES.

AM

ADCOM METALS COMPANY, INC.
INTERSTATE INDUSTRIAL PARK
I-85 AT BEAVER RUIN RD., ATLANTA, GA.
POST OFFICE BOX 29005 — PHONE 443-1171

CUSTOMER'S ORDER NO.	ADCOM ORDER NO.	DATE SHIPPED	SPECIFICATION
M-102401	761	11-13-69	

SHIPPED TO

Chicago Bridge & Iron
Box 13308
Memphis, Tenn. 38113

MARKED:

ITEM	CONSISTING OF 120#
3/32" x 36" 1NMM	

GENTLEMEN: WE HEREBY CERTIFY THAT MATERIAL REFERRED TO ABOVE CONFORMS TO THE PHYSICAL AND CHEMICAL TESTS AS FOLLOWS AND IS IN ACCORDANCE WITH SPECIFICATIONS:-

HEAT	C.	Mn.	Si.	S.	P.	Cr.	Ni.	Cu.	Mg.	Fe.	Al.	Ti.	Cb. + Ta.	Mo.	Co.
986	.16	1.97	.07	.012	.010	.010	1.07	.03			.006			.55	

HEAT	TENSILE STRENGTH	YIELD STRENGTH	ELON.	GRAIN SIZE	ROCKWELL	SHEAR
	201,700 PSI					

YOU REQUESTED THIS IMPORTANT INFORMATION.

Very truly yours,
ADCOM METALS COMPANY, INC.

NOTARY

AUTHORIZED OFFICIAL

Notary Public, Georgia State at Large
Commission Expires April 22, 1972

PLEASE GIVE TO YOUR PURCHASING AGENT.

1

File
R.V. Materials
Surveillance Program

Attachment 10
page 1 of 2

NUCLEAR OPS. DIVISION
Date Recd: JUN 27'85
Resp. Person: Att. O. Jensen
cc: Satyan-Sharma
Steinhart/Kobyra
Clerk: R. Neff

JUN 24 1985

39.60.9.4

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Services Integration Division

Box 2728
Pittsburgh Pennsylvania 15230-2728

AEP-85-641

June 14, 1985

COOK PLANT
MED RECORD - MED COPY
SECTION HE&V
ENGINEER JRJ
DATE 4/22/87
☒ PLANT LIFETIME
DATE TO PLANT N/A
☐ NON PERMANENT
MINIMUM RETENTION ____ YRS.

Mr. M. P. Alexich, Vice President
and Director Nuclear Operations
American Electric Power Service Corporation
One Riverside Plaza
Columbus, Ohio 43216

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D. C. COOK UNIT 1
Reactor Vessel Beltline Region Weld Chemistry

Dear Mr. Alexich:

A review of the weld wire and flux used to fabricate the weld seams in the core beltline region of the D. C. Cook Unit 1 reactor vessel was conducted per the request of D. Hafer of American Electric Power Service Corporation to determine the as deposited copper, nickel and phosphorous content of the as deposited weld seams.

The circumferential girth seam between the intermediate and lower shell is considered to be the limiting weld seam in the vessel. This seam was fabricated with weld wire heat number 1P3571 and Linde 1092 flux lot number 3958. Eight separate chemical analyses are known to have been performed on this combination of the wire and flux and the results are presented below:

Source	Cu	Ni	P
CE Weld Qualification Test (Single Wire)	.40	.82	.017
CE Weld Qualification Test (Tandem Wire)	.37	.75	.017
Kewaunee Unirradiated Surveillance Weld	.20	.77	.016
Maine Yankee Unirradiated Surveillance Weld	.36	.78	.015
Maine Yankee Irradiated Charpy Specimen	.25	.70	.030
Maine Yankee Irradiated Charpy Specimen	.25	.66	.020
Maine Yankee Irradiated Charpy Specimen	.33	.71	.040
Maine Yankee Irradiated Charpy Specimen	.33	.70	.040
Average	.31	.74	.024

Based upon the above data, it is Westinghouse's recommendation that the average of the above data points be used for the Cu and Ni content, since this would be more realistic than using any single data point. This approach has been accepted by the NRC on other applications.

The phosphorous content reported for the irradiated specimens is considered to be highly suspect. Westinghouse considers the average of the four unirradiated values (.016 WT%) to be a realistic phosphorous content for the weld.

The longitudinal weld seams in the beltline region of the vessel were made with a tandem submerged arc process using weld wire heats 12008 and 13253 with Linde 1092 flux lot 3791. No as deposited weld chemistry exists for this combination of wires and flux. Four other tandem welds which contained wire heat number 12008 showed as deposited copper contents of 0.19 to 27%. The surveillance weld which was made from wire 13253 and Linde 1092 flux lot 3791 and which has a copper content of 0.27% is considered to be highly representative of the longitudinal weld seams and the use of its chemistry for the longitudinal weld seams appears appropriate.

The application of new copper and nickel values to the beltline region girth weld seam of the D. C. Cook reactor vessel will not result in the vessel exceeding the PTS screening limits imposed by the NRC.

Please call should you require more information

Very truly yours,

Casimir C. Suda for

A. P. Suda, Manager
Great Lakes Area
Projects Department

APS/debi
4496f:12

cc: M. P. Alexich, 1L
D. Hafer, 1L
W. G. Smith, 1L
J. Feinstein, 1L

Attachment 11

JUN 19 1989

cc: Mr. P. Alexich
T. O. Argenta
P. A. Barrett
S. J. Brewer
J. G. Feinstein
S. P. Klementowicz
R. F. Kroeger
J. F. Kurgan
D. H. Malin
J. J. Markowsky
R. I. Pawliger
J. B. Shinnock
S. H. Steinhart
D. H. Williams, Jr.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 9, 1989

Docket No. 50-315

Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-58 (TAC NO. 71062)

The Commission has issued the enclosed Amendment No. 126 to Facility Operating License No. DPR-58 for the D. C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 14, 1988 and supplements dated December 30, 1988, and June 5, 1989.

This amendment revises the TSs to allow operation of future reload cycles of D. C. Cook Unit 1 at reduced pimary coolant system temperature and pressure conditions. The reduced temperature and pressure (RTP) conditions will decrease the steam generator U-tube stress corrosion cracking of the type observed at D. C. Cook Unit 2.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Lawrence A. Yandell for

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects -
III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 126 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 14, 1988 as supplemented December 30, 1988, and June 5, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 126, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lawrence A. Yandell

Lawrence A. Yandell, Acting Director
Project Directorate III-1
Division of Reactor Projects -
  III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: June 9, 1989

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. S. Brewer
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgeman, Michigan 49106

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mayor, City of Bridgeman
Post Office Box 366
Bridgeman, Michigan 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, Michigan 48909

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3500 N. Logan Street
Post Office Box 30035
Lansing, Michigan 48909

Attachment 9

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.126 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letter dated October 14, 1988, as supplemented December 30, 1988, and June 5, 1989,the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed amendment would permit the operation of future reload cycles of Unit 1 at reduced primary system temperature and pressure conditions. The reduced temperature and pressure (RTP) conditions will decrease the steam generator U-tube stress corrosion cracking of the type observed at the D. C. Cook Nuclear Plant, Unit 2. The licensee's contractor (Westinghouse) has determined that this RTP program should more than double the time to reach a given level of steam generator U-tube corrosion in comparison to the original temperatures and pressure.

D. C. Cook, Unit 1 is presently licensed to operate at 3250 MWt, which is rated thermal power defined by Definition 1.3 of the Technical Specifications. Some transient and accident analyses are performed at a higher power level to position Unit 1 for a potential power uprating. However, not all of the analyses have been performed at this higher power level. The small break loss-of-coolant accident (LOCA) analysis was, for example performed at a power of level of 3250 MWt with the high head safety injection cross-tie valve shut and at 3588 MWt for all other analyzed plant conditions. The staff's review of the RTP program for Unit 1 did not consider any issues related to a future power uprating.

The licensee performed analyses and evaluations to support the RTP program for D. C. Cook, Unit 1. The licensee's efforts addressed full rated thermal power operation (3250 MWt) with a range of vessel average temperature between 547°F and 576.3°F. Two discrete values of the pressure, 2100 psia and 2250 psia, were used in the analyses and evaluations. The analyses and evaluations support a maximum average tube plugging level of 10%, with a peak steam generator tube plugging level of 15%. The licensee will select the desired operating temperature and the pressure on a cycle-by-cycle basis.

The licensee performed the safety analyses and evaluations at conservatively high power levels and high primary system temperatures in order to position both of the D. C. Cook units for future power uprating and in order to support potential future operation of Unit 2 at reduced temperatures and pressure. The potential uprated power for Unit 1 that is partially supported by this analysis and evaluation is 3425 MWt, which corresponds to a reactor power level of 3413 MWt. The design power capability parameters are given in Table 2.1-1 of Reference 2.

2.0 EVALUATION

2.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

2.1.1 Large and Small Break LOCA Analyses

The licensee performed a large break LOCA analysis using the 1981 version of the Westinghouse ECCS Evaluation Model, which uses the BASH computer code.

The analysis assumptions include a total peaking factor, F_Q, of 2.15, a hot channel enthalpy rise factor, F-delta H, of 1.55, 10% safety injection flow degradation, a reactor power level of 3413 MWt, and 15% uniform steam generator tube plugging level. A range of hot-leg temperatures of 580.7°F to 611.2°F and a range of cold-leg temperatures of 513.3°F to 546.2°F, consistent with the temperature range of the RTP program, were considered in the analysis. In the analysis, the reactor coolant system pressure was varied to justify plant operation at either 2100 psia or 2250 psia. A large-break LOCA analysis was also performed with the RHR cross-tie valve closed. For this case, a reduced core power of 3250 MWt was used to compensate for the reduction in safety injection flow caused by the closed RHR cross-tie valve. For those limiting pressure and temperature conditions which produced the largest peak clad temperature, a full break spectrum of discharge coefficients was performed.

The limiting break size was determined to be a cold-leg guillotine break with a discharge coefficient, C_d, of 0.6, a hot-leg temperature of 611.2°F and a primary system pressure of 2250 psia, assuming maximum safety injection flow. The peak clad temperature was calculated to be 2180.5°F. Based on these results, the requirements of 10 CFR 50.46 have been met for the Unit 1 large-break LOCA analysis.

The licensee performed a small-break LOCA analysis using the Westinghouse small-break ECCS Evaluation Model, which uses the NOTRUMP code. The analysis assumptions included a total peaking factor of 2.32, a hot channel enthalpy rise factor of 1.55, safety injection flow rates based on pump performance curves degraded 10% below design head and including the effect of closure of the high head safety injection cross-tie valve, and a uniform 15% steam generator tube plugging level. The analysis was performed at a core power level of 3250 MWt, a range of operating core average temperatures of 547°F to 581.3°F, and reactor pressure of either 2100 psia or 2250 psia. All other plant conditions were analyzed at a power of 3588 MWt. The licensee analyzed a spectrum of cold-leg breaks at the limiting reactor coolant system temperature and pressure conditions. The limiting break size from this analysis was then analyzed at other temperature and pressure points of the operating range. The limiting case was determined to be a three-inch diameter cold-leg break at a pressure of 2100 psia and at a core average temperature of 547°F. This limiting break resulted in a peak clad temperature of 2122°F. Based on these results, the requirements of 10 CFR 50.46 have been met for the Unit 1 small-break LOCA analysis.

The licensee reviewed the effect of the RTP program on the post-LOCA hot-leg recirculation time to prevent boron precipitation. This time is affected by power level and various systems' water volumes and boron concentrations. Because these systems' water volumes and boron concentrations are not affected by the RTP program, there is no effect on the post-LOCA hot-leg switchover time.

The licensee reviewed the effect of the RTP program on the post-LOCA hydrogen generation rates. The assumption of 120°F maximum normal operations containment temperature bounds, for the analysis of record, the effect of the primary system temperature changes of the RTP program on the post-LOCA hydrogen generation rates.

2.1.2 Non-LOCA Transients and Accidents

The licensee has evaluated the impact of the RTP program on the non-LOCA events presented in Chapter 14 of the D. C. Cook, Unit 1 FSAR. The approved reload core design methodology and design codes were used. The evaluations were performed to support the operation of Unit 1 at a core power of 3250 MWt over a vessel average temperature range between 547°F and 576.3°F at a primary system pressure of either 2100 psia or 2250 psia. The evaluation assumes a steam generator tube plugging level of 10%, with a peak steam generator tube plugging level of 15%. The non-LOCA safety evaluation supports the parameters of the RTP program with the exceptions of the steamline break mass and energy releases outside containment, which were evaluated at a full power vessel average temperature no greater than the current D. C. Cook Unit 1 full power average temperature, T_{avg}, of 567.8°F.

The evaluation performed by the licensee also considered the parameters for a potential uprating of Unit 1 to reactor core power level of 3413 MWt, with a vessel average temperature range between 547°F and 578.7°F at a primary system pressure of either 2100 psia or 2250 psia. The steam generator tube plugging level is assumed to be the same as for the RTP program. Even though the non-LOCA evaluation may have been performed for the uprated core power and its associated parameters, the staff's review of this license amendment does not address a D. C. Cook Unit 1 power uprating.

The licensee revised certain reactor trip and engineered safeguards features (ESF) setpoints to provide adequate operating margins for the RTP operating conditions. Revised reactor trip setpoints were incorporated in the overtemperature-delta T (OTDT) and overpower-delta T (OPDT) trip functions. The revised ESF setpoints affects the low steamline pressure value of the high-high steamline flow coincident with a low steamline pressure actuation logic. The new OPDT and OTDT reactor trip setpoints were developed by the licensee for a new set of core thermal safety limits for the RTP program at a reactor core power level of 3413 MWt. The approved setpoint methodology of Reference 3 was used. For those events analyzed with the approved Improved Thermal Design Procedure (ITDP), Reference 4, a safety-limit value of 1.45 was used for the Departure from Nucleate Boiling Ratio (DNBR). This is conservative compared to the design DNBR value of 1.32 for a thimble cell and 1.33 for a typical cell required to meet the DNB design basis.

In the safety analysis for D. C. Cook, Unit 1, the licensee assumed the high pressurizer water level trip setpoint of 100% (nominal reactor setpoint). Furthermore, the reference average temperature used in the OPDT and OTDT trip setpoint equations are rescaled to the full power average temperature each time the cycle average temperature is changed. Similarly, the appropriate value of primary system pressure of either 2100 or 2250 psia was used in the two trip setpoint equations. For the revised ESF setpoint of the high-high steamline flow coincident with low steamline pressure, the low steamline pressure setpoint was lowered from 600 psig to 500 psig to accommodate the range of conditions of the RTP program and a potential power uprating.

2.1.3 Steamline Break Mass/Energy Releases

The current mass and energy releases for the inside containment analysis is based on analyses performed for Cook Unit 2, which are also applicable to Cook Unit 1. Data are represented in Chapter 14 of the FSAR for Unit 2 at power levels of 0, 30, 70, and 100% power. For the "at power" analyses, the initial primary system temperature and secondary steam pressures of the RTP program are lower than those in the Unit 2 FSAR analyses. The mass blowdown rate is dependent on steam pressure and since the steam pressure will be less than the current analyses, the initial mass blowdown rate will be lower. The lower steamline pressure setpoint (500 psig) of the ESF actuation signal does not significantly impact the analysis because the lead-lag compensation results in a steamline pressure signal which anticipates the rapid decrease in pressure caused by a steamline break. Based on these considerations, the licensee concludes that the RTP program will result in a lower integrated energy release into containment and that the data used in the Unit 2 FSAR remains bounding.

A study was performed for Unit 1 of the mass and energy release outside containment to address equipment qualification issues (Ref. 5). Cases at 70% and 100% power were analyzed. The analysis presented in Reference 5 assumed the full power vessel average temperature to be 567.8°F. Any reduction in full power T_{avg} from the analyzed T_{avg} and the associated reduction in initial steam pressure will result in less limiting releases. The low steamline pressure value assumed in the analysis supports the reduced value of the setpoint to 500 psig. The increased level of steam generator tube plugging is acceptable because the analysis assumed better heat transfer characteristics. The licensee concludes that the current mass and energy release analysis is acceptable for the RTP program as long as the full power T_{avg} is equal to or less than 567.8°F.

2.1.4 Startup of an Inactive Loop

The licensee evaluated the startup of an inactive loop event. This event cannot occur above the P-7 permissive setpoint of 10% power as restricted by the Technical Specifications. The parameters assumed in the FSAR analysis for three-pump operation at 10% power remain bounding for the parameters for 10% power condition. The licensee concludes, therefore, that the conclusions presented in the FSAR remain valid.

2.1.5 Uncontrolled Rod Bank Withdrawal from a Subcritical Condition

The uncontrolled rod bank withdrawal from a subcritical condition transient causes a power excursion. This power excursion is terminated, after a fast power rise, by the negative Doppler reactivity coefficient of the fuel, and a reactor trip on source, intermediate, or power range flux instrumentation. The power excursion results in a heatup of the moderator/coolant and the fuel. The analysis used a reactivity insertion rate of 75 pcm (note that one pcm is equal to a reactivity of 10^{-5} delta K/K). This reactivity insertion rate is greater than for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at the maximum speed of 45 inches/minute. The neutron flux overshoots the nominal full power value; however, the peak heat flux is much less than the full power nominal value because of the inherent thermal lag of the fuel. The analysis, with the reduced system pressure of 2100 psia, yields the minimum value of DNBR. The analysis is performed using the Standard Thermal Design Procedure (STDP). The W-3 DNB correlation was issued to evaluate DNBR in the span between the lower non-mixing vane grid and

the first mixing vane grid. The WRB-1 DNB correlation is applied to the remainder of the fuel assembly. From the analysis performed, the licensee concludes that the DNB design bases are met for all regions of the core, and therefore, the conclusions in the FSAR remain applicable for a reduction in nominal system pressure to 2100 psia.

2.1.6 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

The uncontrolled rod bank withdrawal from a power condition transient leads to a power increase. The transient results in an increase in the core heat flux and an increase in the reactor moderator/coolant temperature. The reduction in pressure for the RTP program is non-conservative with respect to DNB. In addition, a revised Overtemperature Delta-T setpoint equation is being assumed in the Cook Unit 1 analyses. The Power Range High Neutron Flux and Overtemperature Delta-T reactor trips provide the primary protection against DNB. Both minimum and maximum reactivity cases were analyzed over a range of reactivity insertion rates. The licensee provided quantitative results for the maximum reactivity feedback case for power levels of 10%, 60%, and 100% power for a range of reactivity insertion rates. The results indicate that the DNBR limit is met for all the cases.

The licensee examined a number of cases associated with the pressurizer water volume transient caused by an uncontrolled control rod assembly bank withdrawal-at-power event. It was determined that credit for high pressurizer water level reactor trip was required to prevent the pressurizer from filling. The licensee assumed a value of 100% narrow range span (NRS) for the high pressurizer water level reactor trip setpoint. A time delay of 2 seconds was assumed for trip actuation until rod motion becomes adequate to terminate the transient.

Thus the high neutron flux and overtemperature-delta T reactor trips provide adequate protection over the range of possible reactivity insertion rates in that the minimum value of DNBR remains above the safety-limit DNBR value. In addition, the high pressurizer water level reactor trip prevents the pressurizer from filling.

2.1.7 Rod Cluster Assembly Misalignment

The rod cluster control assembly misalignment events consist of three separate events: (1) a dropped control rod, (2) a dropped control bank, and (3) a statically misaligned control rod. These events were reanalyzed because the reduction in pressure for the RTP program is nonconservative with respect to the DNB transient. A dropped control rod or control bank may be detected in the following manner: (1) by a sudden drop in the core power as seen by the nuclear instrumentation system; (2) by an asymmetric power distribution as seen by the excore neutron detectors or the core exit thermocouples; (3) by rod bottom signal; (4) by the rod position deviation monitor; and (5) by rod position indicators. A misaligned control rod may be detected in the following manner; (1) by an asymmetric power distribution as seen by the excore neutron detectors or the core exit thermocouples; (2) by the rod position deviation monitor; and (3) by rod position indicators. The resolution of the rod position indicator channel is ±5 percent or ±12 steps (±7.5 inches). Deviation of any control rod from its group by twice this distance (±24 steps or ±15 inches) will not cause power distribution worse than the design limits. The rod position deviation monitor provides an alarm before a rod deviation can exceed ± 24 steps or ± 15 inches.

The dropped rod event was analyzed using an approved methodology (Ref. 6). A dropped rod or rods from the same group will result in a negative reactivity insertion which may be detected by the negative neutron flux rate trip circuitry. If detected, a reactor trip occurs in about 2.5 seconds. For those dropped rod events for which a reactor trip occurs, the core is not adversely impacted because the rapid decrease in reactor power will reach an equilibrium value dependent on the reactivity feedback or control bank withdrawal (if in automatic control). The limiting case for this class of events is the case with the reactor in automatic control. For this case a power overshoot occurs before an equilibrium power condition is reached. The licensee states that, using the methodology of Reference 6, all analyzed cases result in DNBR values which are within the safety-limit DNBR value.

The licensee states that a dropped rod bank results in a reactivity insertion of at least 500 pcm. This will be detected by the negative neutron flux rate trip circuitry and cause a reactor trip within about 2.5 seconds of the initial motion of the rod bank. Power decreases rapidly and there is, therefore, no adverse impact on the reactor core.

The most severe misalignment cases, with respect to DNBR, are those in which one control rod is fully inserted or where control bank "D" is fully inserted but with one control rod fully withdrawn. Multiple alarms alert the operator before adverse conditions are reached. The control bank can be inserted to its insertion limit with any control rod fully withdrawn without DNBR falling below the safety-limit DNBR value, as shown by analysis. An evaluation performed by the licensee indicates that control rod banks other than the control bank would give less severe results. For the case with one rod fully inserted, DNBR remains above the safety-limit DNBR value. For all cases following identification of a control rod misalignment, the operator is required to perform actions in accordance with plant Technical Specifications and procedures.

2.1.8 Chemical and Volume Control System Malfunction

The boron dilution event was analyzed by the licensee for startup and power operation. The analysis is performed to show that sufficient time is available to the operator to determine the cause of the dilution event and take corrective action before the shutdown margin is lost. The licensee reports that 45 minutes is available for Mode 1 (power operation) and 68 minutes for Modes 2 or 3 (startup or hot standby conditions) (Ref. 7).

2.1.9 Loss of Reactor Coolant Flow

The loss-of-flow transient causes the reactor power to increase until the reactor trips on either a low-flow trip signal or reactor coolant pump power supply undervoltage signal. The reactor power increase causes a reactor moderator/coolant temperature increase. This initial coolant temperature increase causes a positive reactivity insertion because of the positive moderator temperature coefficient. The licensee analyzed both a partial loss-of-flow (loss of one pump with four coolant loops in operation) transient and a complete loss-of-flow transient (loss of four pumps with four coolant loops in operation). For the partial loss-of-flow transient, the reactor is assumed to be tripped on a low-flow signal. For a complete loss-of-flow transient, the reactor is assumed to be tripped on a pump undervoltage signal. For either event, the average and hot channel heat fluxes do not increase significantly above their initial values and the DNBR remains above the safety-limit DNBR value.

2.1.10 Locked Rotor Accident

The locked rotor accident causes a rapid reduction in the fluid flow through the affected loop. The reactor trips on a low-flow signal which rapidly reduces the neutron flux upon control rod insertion. Control rod motion starts 1 second after the flow in the affected loop reaches 87% of its nominal value. The licensee evaluated this accident assuming that offsite power is available. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after reactor trip. The licensee performed an analysis to determine the DNB transient and to demonstrate that the peak system pressure and the peak clad temperature remain below limit values. The peak reactor coolant system pressure of 2588 psia reached during the transient is less than that which would cause stresses to exceed the faulted conditions stress limits. The peak clad temperature reached is 1959°F. Less than 4.5% of the fuel rods in the most limiting fuel assembly reach values of DNBR less than the safety-limit DNBR value. These results indicate that the RTP program assumptions give acceptable consequences for the locked rotor accident.

2.1.11 Loss of External Electrical Load

The loss-of-external-electrical-load event was analyzed by the licensee to show the adequacy of pressure-relieving devices and to demonstrate core protection. This reanalysis was necessary because of changes in reactor pressure and temperature conditions for the RTP program and because of changes to the Overtemperature-Delta T reactor trip setpoint equation. Maximum and minimum reactivity feedback cases were examined, with the case analyzed with and without credit for pressurizer sprays and power-operated relief valves. For the minimum reactivity feedback case with pressurizer pressure control, the reactor trips on a high pressurizer pressure signal. For the maximum reactivity feedback case with pressurizer pressure control, the reactor trips on a low-low steam generator water level signal. For the minimum reactivity feedback case without pressurizer pressure control, the reactor trips on a high pressurizer pressure signal. For all four cases, the minimum value of DNBR remains well above the safety-limit DNBR value and the Overtemperature-Delta T setpoint was not reached. The analysis confirms that the conclusions of the FSAR remain valid for this event for the RTP program.

2.1.12 Loss of Normal Feedwater Flow

The loss-of-normal-feedwater-flow event was analyzed by the licensee to show that the auxiliary feedwater system is capable of removing the stored and decay heat, thus preventing overpressurization of the reactor coolant system or uncovering the core, and returning the plant to a safe condition. The reanalysis was based on a positive moderator temperature coefficient. A conservative decay heat model based on the ANSI/ANS-5.1-1979 decay heat standard (Ref. 8) was used. Pressurizer power operated relief valves and the maximum pressurizer spray flow rate were assumed to be available since a lower pressure results in a greater system expansion. The initial pressurizer water level was assumed to be at the maximum nominal setpoint of 62% narrow range span. Reactor trip occurred when the low-low steam generator water level trip setpoint was reached. The results of the analysis show that a loss of normal feedwater does not adversely affect the reactor core, the reactor coolant system, or the steam system, and that the auxiliary feedwater system is sufficient to prevent water relief through the pressurizer relief or safety valves. The pressurizer does not fill and, therefore, the conclusions of the FSAR remain valid for this event, including RTP conditions.

2.1.13 Excessive Heat Removal Due to Feedwater System Malfunctions

The excessive-heat-removal event due to feedwater system malfunction was analyzed by the licensee to demonstrate core protection. This analysis was necessary because of changes in reactor core temperatures and pressure for the RTP program and because of changes to the OTDT and OPDT trip setpoints. This event is an excessive-feedwater-addition event caused by a control system malfunction or an operator error which allows a feedwater control valve to open fully. The licensee analyzed both full power and hot zero power cases. Both cases assumed a conservatively large negative moderator temperature coefficient. The full power case assumed the reactor was in automatic or manual control. The Improved Thermal Design Procedure (ITDP) of Reference 4 was used in the analysis. For the accidental full opening of one feedwater control valve with the reactor at hot-zero power conditions, the licensee determined that the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in the Uncontrolled-Rod-Cluster-Assembly-Bank-Withdrawal-at-Subcritical-Condition event. Thus, this hot-zero power case is bounded by the results obtained previously for the other event. In addition, if the event were to occur at a hot-zero power and an exactly critical condition, the power range high neutron flux trip (low setting) of about 25% of nominal full power will trip the reactor. The hot-full power case with the reactor in automatic control is more severe than the case with the reactor in manual control. For all excessive feedwater cases, continuous addition of cold feedwater is prevented by automatic closure of all feedwater isolation valves on steam generator high-high level signal. A turbine trip is then initiated and a reactor trip on a turbine trip is then assumed. The results presented by the licensee demonstrate the safe response of Cook Unit 1 to the event, at hot-full power and in automatic control, with the DNBR remaining well above the safety-limit DNBR value.

2.1.14 Excessive Increase in Secondary Steam Flow

The excessive-increase-in-secondary-steam-flow event was analyzed by the licensee to demonstrate core protection. This event is an overpower transient for which the fuel temperature will rise. It was analyzed because of reactor core temperature and pressure changes for the RTP program and because of changes to the OTDT and OPDT setpoints. The Cook Unit 1 reactor control system is designed to accommodate a 10% step load increase and a 5%-per-minute ramp load increase over the range of 15 to 100 percent of full power. Load increase in excess of these rates would probably result in a reactor trip. Four cases were analyzed by the licensee. These included minimum and maximum reactivity feedback cases with each case analyzed for both manual and automatic reactor control. For the minimum reactivity feedback cases, a zero moderator temperature coefficient was assumed to bound the positive moderator temperature coefficient. For all the cases, no credit was taken for the pressurizer heaters. The analyses used the ITDP of References 4. The studies show that the reactor reaches a new equilibrium condition for all the cases studied, with DNBR remaining well above the safety-limit DNBR value. The operators would follow normal plant procedures to reduce power to an acceptable value to conclude the event.

2.1.15 Loss of all AC Power to the Plant Auxiliaries

The loss-of-all-AC-power-to-the-plant-auxiliaries event was analyzed to demonstrate the adequacy of the heat removal capability of the auxiliary feedwater system. This transient is the limiting transient with respect to the possibility of pressurizer overfill. This event is more severe than the loss-of-load event because the loss of AC power results in a flow coastdown due to the loss of all four reactor coolant pumps. This results in a reduced capacity of the primary coolant to remove heat from the core. A positive moderator temperature coefficient was assumed in the analysis. A conservative decay heat model based on the ANSI/ANS-5.1-1979 decay heat standard (Ref. 8) was used. No credit is taken for the immediate release of the control rods caused by the loss of offsite power. Instead a reactor trip is assumed to occur on a steam generator low-low level signal. Pressurizer power operated relief valves and the maximum pressurizer spray flow rate was assumed to be available since a lower pressure results in a greater system expansion. The initial pressurizer water level is assumed to be at the maximum nominal setpoint of 62% narrow range span plus uncertainties of 5% narrow range span. The results demonstrate that natural circulation flow is sufficient to provide adequate decay heat removal following reactor trip and reactor coolant pump coastdown. The pressurizer does not fill. Thus, the loss of AC power does not adversely affect the core, the reactor coolant system, or the steam system, and the auxiliary feedwater system is sufficient to prevent water relief through the pressurizer relief or safety valves.

2.1.16 Steamline Break

The steamline break accident was analyzed by the licensee to assess the impact of the reduced reactor coolant system pesssure of the RTP program and the low steam pressure setpoint (lowered from 600 psig to 500 psig) of the coincidence logic with high-high steam flow for steamline isolation and safety injection actuation. An end-of-life shutdown margin of 1.6% delta K/K for no load, equilibrium xenon conditions, with the most reactive control rod stuck in its fully withdrawn position, was assumed. A negative moderator temperature coefficient corresponding to the end-of-line rodded core was assumed. The licensee evaluated four combinations of break sizes and initial plant conditions to determine the core power transient which can result from large area pipe breaks. The first case was the complete severance of a pipe downstream of the steam flow restrictor with the plant at no-load conditions and all reactor coolant pumps running. The second case was the complete severance of a pipe inside the containment at the outlet of the steam generator with the plant at no-load conditions and all reactor coolant pumps running. The third case is the same as the first case with the loss of offsite power simultaneous with the generation of a Safety Injection Signal (loss of offsite power results in reactor coolant pump coastdown). The fourth case is the same as the second case with loss of offsite power simultaneous with the generation of a Safety Injection Signal. A fifth case was performed to show that the DNBR remains above the safety-limit DNBR value in the event of the spurious opening of a steam dump or relief valve. The licensee determined that the first case was the limiting case, that is, the double-ended rupture of a main steam pipe located upstream of the flow restrictor with offsite power available and at no-load conditions. The results indicate that the core becomes critical with the control rods inserted (however, with the most reactive control rod stuck out) before boron solution at 2400 ppm enters the reactor coolant system. The core power peaks at less than the nominal full core power. The DNB analysis showed that the

minimum DNBR remained above the safety limit DNBR value, even though this event is classified as an accident with fuel rods undergoing DNB not precluded. The analysis performed by the licensee demonstrates that a steamline break accident will not result in unacceptable consequences.

2.1.17 Rupture of Control Rod Drive Mechanism Housing (Rod Ejection Accident)

The rod ejection accident is analyzed at full power and hot, zero-power conditions for both beginning-of-cycle (BOC) and end-of-cycle (EOC). The analysis used ejected rod worth and transients peaking factors that are conservative. Reactor protection for a rod ejection is provided by neutron flux trip, high and low setting, and by the high rate of neutron flux increase trip. The analysis modeled the high neutron flux trip only. The maximum fuel temperature and enthalpy occurred for hot, full-power BOC case. The peak fuel enthalpy was, however, below 200 cal/gm for all the cases analyzed. For the hot, full-power cases, the amount of fuel melting in the hot pellet was less than 10%. Because fuel and clad temperatures and the fuel enthalpy do not exceed the FSAR limits, the conclusions of the FSAR remain valid.

Based on a review of the licensee's evaluation and analysis of the non-LOCA transients and accidents (2.1.3 through 2.1.17) for the reduced temperature and pressure operation (the RTP program), the staff concludes that they are acceptable because (1) approved methodologies and computer codes have been used, and (2) all applicable safety criteria have been met. This review is based on (1) a full power vessel average temperature of less than or equal to 567.8°F, (2) a steam generator tube plugging level of 10% with a peak tube plugging level of 15%, and (3) the minimum measured flow requirement of 91,600 gpm per loop is met.

2.1.18 Steam Generator Tube Rupture (SGTR) Accident

The licensee analyzed the steam generator tube rupture (SGTR) event for Cook Unit 1 using methodology and assumptions consistent with those used for the Cook FSAR SGTR analysis. The range of parameters associated with a future rerating program and the RTP program were used in sensitivity analyses to assess the impact of these programs on the primary-to-secondary break flow and the steam released to the atmosphere by the affected steam generator. These two factors affect the radiological consequences of an SGTR accident. In addition, the licensee's evaluation of the radiological doses considers the effect of the noble gas concentrations. The licensee states that the results of the analyses show that the doses remain within a small fraction (10%) of the 10 CFR Part 100 guidelines for both the thyroid and whole body doses. Since the worst case doses are within the 10 CFR Part 100 guidelines, the staff concludes that the analysis of the SGTR is acceptable

2.1.19 Fuel Structural Evaluation

The fuel assembly lift and buoyancy forces are increased for the RTP program at Cook Unit 1 because a reduction in reactor coolant system temperature of about 20°F will increase the coolant density by about 3%. The licensee evaluated this force increase against the fuel assembly allowable holddown load. The results of the evaluation show that the increased force is well within the minimum spring holddown force design margin. In addition, the licensee determined that the cold-leg break remains the most limiting pipe rupture transient with respect to lateral and vertical hydraulic forces. Based on the licensee's review, the staff concludes that the 15x15 fuel assembly design remains acceptable.

The fuel rod design was evaluated to assess the impact of future rerating. The licensee determined that the rod internal pressure criterion will continue to be the more important factor in fuel burnup capabilities. The fuel will also undergo more severe fuel duty because of the uprated power. The licensee plans to perform cycle-specific verification for each reload to assure that all fuel rod design criteria are met.

2.1.20 Justification for Pressurizer Level

The purpose of the Pressurizer High Level Limit is to ensure that a steam bubble is present in the pressurizer prior to power operation to minimize the consequences of overpressure transients and the possibility of passing water through the relief and safety valves. The safety analysis assumes a maximum water volume which corresponds to about 65% indicated level. This nominal indicated level is maintained during normal operation by the pressurizer level control system.

The licensee (and the fuel supplier - Westinghouse) recommends the use of 92% for the Pressurizer High Level trip limit. They state that this new trip limit will still ensure the presence of a steam bubble in the pressurizer. The pressurizer level will, however, be controlled to the nominal value. For normal operations (Condition I event), the reactor parameters, including the pressurizer level, do no significantly deviate from their nominal values. The licensee concludes that, for the pressurizer level to exceed the nominal level, a transient or accident must occur for which protective action is provided by the Reactor Protection System. Any other possible conditions for which the nominal level would be exceeded before and during a transient would require a transient or transients beyond those usually considered for an FSAR type of analysis. The staff concludes on the basis of the licensee's evaluation that a Pressurizer High Level Trip of 92% is acceptable.

2.2 BALANCE OF PLANT SYSTEMS

The licensee states that balance of plant (BOP) systems and components were analyzed for the effects of operation at reduced temperature and pressure conditions. The secondary side conditions for these analyses were determined using the Performance Evaluation and Power System Efficiencies (PEPSE) heat balance data (14.20 E6 lb/hr main steam flow and main feed flow). The systems reviewed were the non safety-related secondary side power generating and nonpower generating systems. Included in the licensee's analysis were portions of the main feedwater, main steam, steam generator blowdown (SGBS), component cooling water (CCWS), auxiliary feedwater (AFS), heating, ventilation, and air conditioning (HVAC), service water, waste disposal, fire protection, radiation monitoring, and spent fuel pool (SFP) cooling and cleanup systems.

The performance of the above BOP systems was evaluated at the reduced temperature and pressure by using the new primary side NSSS data (14.20E6 lb/hr main steam and main feed flow, and 434°F main feed temperature) furnished by Westinghouse. The licensee states that the impact on containment pressures and temperatures following a postulated design basis main steam line break was evaluated and its effect on equipment qualification was verified. The flooding analysis in safety-related areas of the plant as a result of a postulated pipe break was reevaluated due to the slight increase in flow rates in the main feed, condensate, and main steam systems. The turbine-generator system was also evaluted to confirm its integrity and performance at the increased steam volumetric flow rate and to verify that the original turbine missile analysis remains valid.

The licensee's analysis of BOP system performance provided the following findings concerning the RTP conditions at the present licensed power level of 3250 MWt NSSS power:

(a) The capability of the safety-related portion of the main feedwater system will not be affected and will continue to perform its safety function because the proposed RTP conditions are bounded by the existing main feedwater system design. The licensee's analysis of the pressure/temperature rating conditions for the system confirms that pressure boundary integrity will not be affected. In addition, the main feedwater system isolation valve closure time is not affected by the RTP-imposed conditions.

(b) The capability of the steam generator blowdown system to remove impurities from the secondary side remains essentially the same for the RTP-imposed conditions during normal operation based on the exsisting design.

(c) The reactor makeup water system's (MSW) capability to provide demineralized water for makeup and flushing operations throughout the NSSS auxilliaries, the radwaste systems, and fuel pool cooling and cleanup system is not challenged because the existing system design is based on the worst case demand which bounds the RTP conditions.

(d) The licensee confirmed that safety-related equipment will not be affected by changes in the flooding analysis due to the RTP conditions. Flooding in the auxiliary building due to failure of nonseismic Class I piping has been reviewed. The licensee analyzed systems having access to large water volumes and/or potentially large flowrates were considered as discussed in the FSAR. The only such system is the main feedwater system. Since the changes in flow in the main feedwater system are still within the design limits, the results concerning flooding discussed in the FSAR are still applicable.

Flooding in the containment is slightly increased due to the larger initial water mass in the reactor coolant system because of the higher density at the reduced temperature. This change was found to be within the volume margins used to determine the maximum flood-up elevation. The containment flooding evaluation in the FSAR remains valid at the RTP-induced conditions.

(e) The adequacy of the AFW system for accident mitigation was demonstrated in the Westinghouse accident analysis performed in support of the RTP program under the following scenarios:

1. Loss of main feedwater
2. Loss of offsite power
3. Main steam line rupture

Each accident analysis demonstrated acceptance criteria such as system overpressure limits or DNB limits. The AFW system's ability for design basis accident decay heat removal calculated in the RTP analysis is unaffected.

(f) As evaluated in the RTP analysis, the heat loads in both the primary and secondary systems due to reactor decay heat remain unchanged. Therefore, the Component Cooling Water System (CCWS) analysis and service water system (SWS) analysis in the FSAR remain valid.

(g) For main steam line breaks inside the containment structure, the pressure and temperature will remain within the bounds of the peak pressure and temperature used in the evaluation of containment performance. The initial primary temperatures and secondary steam pressures under the RTP conditions will be lower than those used in the FSAR analysis. The licensee has confirmed that containment environmental qualification of equipment inside containment is not affected.

(h) The superheated mass and energy release analysis outside containment was evaluated to address equipment qualification issues. The primary temperatures and secondary steam pressures resulting from the RTP conditions will be lower than those used in the FSAR analysis. The mass and energy release will be lower and operation with RTP will result in lower temperatures in the break areas. As such, the current superheat mass and energy release analysis outside containment remains bounding provided the full power vessel average temperature is restricted to the currently-licensed 567.8°F and below.

(i) The secondary pressure conditions assumed in the high energy steam line break analysis will be lower than those presented in the FSAR. These bound the proposed RTP conditions and therefore the current analysis is sufficient.

(j) The primary function of the spent fuel pool cooling system (SFPCS) is to remove decay heat that is generated by the elements stored in the pool. Decay heat generation is proportional to the amount of radioactive decay in the elements stored in the pool which is proportional to the reactor power history. Since the plant's rated power level of 3250 MWt remains unchanged, the demand on the SFPCS is not increased. The purification function is controlled by SFPCS demineralization and filtration rates that are not affected by the RTP conditions.

(k) The fire protection systems and fire hazards are independent of the plant operating characteristics with the exception of the slightly increased current requirements for the electric motor driven pumps in the primary system. The increased load is due to the more dense water being pumped under the RTP conditions. The increased current required is small and therefore is not considered to be a fire hazard.

(l) The licensee confirmed that BOP systems have the capability to maintain plant operation under the RTP-induced conditions without modification to the existing design.

The staff has reviewed the FSAR and licensee submittals in order to verify that safety-related BOP system performance capability, as analyzed, bounds the

changes in design basis accident assumptions created by the RTP operation. The staff has confirmed that safety-related BOP system design capability, flooding protection, and equipment qualifications are bounded for the proposed rerating and therefore are considered acceptable as is.

Based on the above, the staff concludes that the proposed license amendment for the D.C. Cook Nuclear Plant Unit 1 concerning the Reduced Temperature and Pressure is within the existing safety-related BOP system design capability for design basis accident mitigation and, therefore, the staff's previous approval against the applicable licensing criteria for the main steam system, main feed system, CCWS, SWS, AFS, MSW, SGBS, SFPCS, flooding protection, containment performance, and equipment qualifications remain valid. The staff, therefore, finds the BOP systems concerned acceptable for continued operation at the proposed reduced temperature and pressure.

2.3 REACTOR VESSEL AND VESSEL INTERNALS

The reactor vessel is designed to the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition with addenda through the winter 1966). The licensee has determined that the operation of the reactor vessel under the most limiting conditions of the RTP rerating is acceptable for its original 40-year design objective. All of the stress intensity and usage factor limits of the applicable code for the Unit 1 reactor vessel are still satisfied when the RTP is incorporated, with the exception of the 3Sm limit for the Control Rod Drive Motor (CRDM) housings and outlet nozzle safe end. However, the code permits exceeding the 3Sm limit provided plastic or elastic/plastic analysis criteria are met.

The licensee's review of the reactor vessels internals for the RTP program included three seperate areas: a thermal/hydraulic assessment, a RCCA drop time evaluation, and a structural assessment. Force increases were calculated for the upper core plate, across the core barrel, and in the upper internals near the outlet nozzles. In these areas the existing margin was determined to be sufficient to accommodate the increased stresses. The results of this review indicate that the original reactor internals components remain in compliance with the current design require-ments when operating at the new range of primary temperatures and pressures.

The PTS rule requires that at the end-of-life of the reactor vessel, the projected reference temperature (calculated by the method given in 10 CFR 50.61(b)(2), RT/pts) value for the materials in the reactor vessel beltline be less than the screening criterion in 10 CFR 50.61(b)(2). The RT/pts value is dependent upon the initial reference temperature, margins for uncertainty in the initial reference temperature and calculational procedures, the amounts of nickel and copper in the material, and the neutron fluence at the end-of-life of the reactor vessel. Of these properties, only neutron fluence is affected by rerating with RTP. Since the colder coolant in the downcomer region is more dense and thus provides for a more efficient neutron shield for the reactor vessel, fluence estimates are lower than those at current operating conditions. All other properties are independent of the RTP-induced conditions.

The effects of NRC Generic letter 88-11, dated July 12, 1988, regarding Regulatory Guide 1.99 Rev. 2 were evaluated by Westinghouse and determined to not be significant for RTP. The effect of RTP will be incorporated by the licensee in future PTS submittals.

An evaluation was performed to determine the impact of RTP rerating on the applicability of the PTS screening criteria in terms of vessel failure. A probabilistic fracture mechanics sensitivity study of limiting PTS transient characteristics, starting from a lower operating temperature, showed that the conditional probability of reactor vessel failure will not be adversely affected. Therefore, the overall risk of vessel failure will not be adversely impacted, meaning that the screening criteria in the PTS Rule are still applicable for the D.C. Cook Nuclear Plant Unit 1 reactor vessel relative to rerated conditions.

Analysis of the CRDM housings and the outlet nozzle safe end shows the maximum range of primary plus secondary stress intensity exceed the 3Sm limit. The licensee, however, performed a simplified elastic/plastic analysis in accordance with paragraph NB-3228.3 of the ASME Boiler and Pressure Vessel Code, Section III (1971 or later edition) and the higher range of stress intensity is justified.

Therefore, based on the licensee's reviews and analysis of the above portions of the reactor vessel and internals, the staff concludes that the conditons imposed on the reactor vessel and internals by the RTP rerating are acceptable.

2.4 TURBINE MISSILES

The FSAR turbine missile analysis is based on a low pressure turbine failure. The licensee's analysis of the slightly changed steam conditions entering the low pressure turbine shows that the probabilty of a low pressure turbine missile is virtually unaffected.

The factors that directly or indirectly cause stress corrosion cracking in the low pressure turbine wheels are steam pressure and temperature, mass flow rate, steam moisture content, water chemistry, oxygen level, and turbine speed. The licensee reported that changes in these factors are negligible due to the RTP-induced conditions. The only noticeable change that the staff can determine is a 1.0% increase in the steam flow rate.

The staff's conclusion, based on the licensee's review, is that the turbine missile hazard is neglibily affected by the RTP conditons and is, therefore, acceptable.

2.5 PLANT STRUCTURAL AND THERMAL DESIGN

The NSSS review consisted of comparing the existing NSSS design with the performance requirements at the rerated RTP conditions.

The current components of the Cook Unit 1/model 51 steam generators continue to satisfy the requirements of the ASME B&PV Code, Section III,(the code applicable for the design of the Cook Nuclear Plant Unit 1), for this program. In addition, thermal hydraulic evaluations of the steam generators show acceptable stability and circulation ratios at the RTP rerated conditions. Circulation ratio is primarily a function of power, which is unchanged, therefore is itself virtually unchanged. The dampening factor characterizes the thermal and hydraulic stability of the steam generator. Westinghouse has determined that all dampening factors are negative at nearly the same value as the current operating conditions. A negative dampening factor indicates a stable device. Since the code requirements continue to be satisfied, and since stability and circulation ratios have been determined by Westinghouse to be

within the design criteria, the staff concludes that RTP operation is acceptable for the Model 51 steam generators.

The pressurizer structural analysis was performed by modifying the original D.C. Cook Nuclear Plant Pressurizer analysis ("Model 51 Series Pressurizer Report"). The analysis was performed to the requirements of the ASME Code 1968 Edition, which is the design basis for the D.C. Cook Nuclear Units. The only ASME Code requirement affected by the transient modifications was fatigue. The limiting components for fatigue usage factors are the upper shell and the spray nozzle, which are calculated to be 0.97 and 0.99 respectively. These remain, however, within the ASME acceptance criteria of 1.0 and are, therefore, acceptable to the staff.

Reactor coolant pump hydraulics and motor adequacy were reviewed for the proposed RTP conditions by Westinghouse. The increased hot horsepower and stator temperature conditions are within the NEMA Class B limits. A review of generic Reactor Coolant Pump stress reports for model 93A pumps by Westinghouse finds that all the design requirements provide adequate bounding of the RTP-induced conditions and, therefore, the staff finds this acceptable.

Due to lower temperatures from the RTP program, the RCS will not expand as much as currently designed. This will result in support gaps being present in locations that were previously zero. The small gaps in the support structure may result in increased dynamic loading (both seismic and LOCA) in localized areas. The overall LOCA loadings on the RCS, however, remain approximately the same for the following reasons:

1. The lower RCS temperatures yield lower thermal loadings.

2. The D. C. Cook Nuclear Plant has a leak before break design methodology which allows the faulted condition evaluation to proceed without having to consider loadings from postulated breaks in the primary loop piping.

The seismic margin available for this plant is also significant which means that there are no components in the system which are close to their allowable stresses. Based on the above, the temperatures associated with the RTP rerating are, therefore, acceptable to the staff for the loop piping, the loop supports, and the primary equipment nozzles.

The effects of the D.C. Cook Nuclear Plant RTP rerating on the operability and design basis analysis of the CRDM's of Unit 1 were reviewed. The RTP rerating does not affect the operability or service duration of the CRDM latch assembly, drive rod, or coil stack. The CRDM latch assembly and drive rod were originally designed for 650°F, and the design basis stress and fatigue calculations remain representative for these components since the components are exposed to the hot leg temperature, which has not increased. The coil stack is located on the outside of the pressure housing which is subject to ambient containment temperatures, which have not changed. An evaluation was performed on the impact of the RTP rerated operating conditions on the structural analysis of the CRDM pressure housing. The component of the pressure housing which experiences the greatest stress range and has the highest fatigue usage factor is the upper canopy. This is the pressure housing seal weld between the rod travel housing and the cap. Westinghouse provided a review on the impact of the differences

between the original normal and upset condition transients and those of the RTP on the code allowable stress levels and fatigue usage factors. The results of the evaluation are:

1. The maximum stress intensity range is equal to 109,960 psi, which is less than the maximum allowable range of thermal stress of 127,105 psi which was previously found to be acceptable.

2. The total fatigue usage factor is equal to 0.672, which is less than the allowable limit of 1.0 (ASME Section III, 1971 Edition).

The staff concludes, based on licensee evaluations, that the impact of the RTP program on the CRDM's is within design criteria and, therefore, is found to be acceptable.

2.6 CONTAINMENT EVALUATION

Short-Term Containment Response

As part of the analysis to support RTP operation, the reactor cavity and loop subcompartments short-term pressurization in the event of a break of large coolant piping or a steam line was reanalyzed by Westinghouse. In some of those areas, the analyzed pressure exceeded the structural limits as expressed in the FSAR. These structures were reevaluated using the peak pressures obtained from the RTP analysis, WCAP 11902 (ref.2), to confirm that the acceptance criteria of Section 5.2.2.3 of the updated FSAR, titled "Containment Design Stress Criteria," were met.

The original design of the containment included a number of considerations of which the subcompartment pressures were but one. For example, radiation shielding requirements may have dictated a thicker concrete slab than was necessary from a structural perspective. The actual capacity is generally greater than the design pressures stated in the FSAR, and is further increased due to the fact that the materials used are stronger than the required minimum design strengths. In the RTP structural review, advantage was taken of these greater capacities by performing manual or finite element evaluations of the affected structural elements. The greater material strengths were used in the analysis where appropriate.

Loop Subcompartments

The containment building subcompartments are the fully or partially enclosed spaces within the containment which contain high energy piping. The subcompartments are designed to limit the adverse effects of a postulated high energy pipe rupture.

The results of the short term containment analyses and evaluations for the D.C. Cook Nuclear Plant Unit 1 demonstrate that, for the pressurizer enclosure, the fan accumulator room, and the steam generator enclosure, the resulting peak pressures remain below the allowable design peak pressures. For the loop compartments, the peak calculated pressures at the RTP rerated conditions are higher than the FSAR design allowables. For these areas, structural evaluations were performed as discussed above for the revised peak pressures, and the structural adequacy of the containment subcompartments have been confirmed (Ref. 10) as follows:

Differential Pressure, Node 1 or 6 to Node 25

This is the differential pressure from the reactor coolant loop compartments adjacent to the refueling canal nodes 1 or 6 across the operating deck to the upper containment.

Original Design pressure	16.6 psi
Original Calculated pressure	14.1 psi
New Calculated pressure	18.7 psi

The licensee demonstrated the increased differential pressure to be acceptable by review of existing computer analysis of the reactor coolant pump hatch covers and reevaluation of the operating deck load carrying capacity.

Differential Pressure, Node 2 or 5 to Node 25

This is the differential pressure across the operating deck from the reactor coolant loop compartments located 90 degrees from the refueling canal to the upper containment.

Original Design pressure	12.0 psi
Original Calculated pressure	10.6 psi
New Calculated pressure	13.0 psi

The licensee demonstrates the increased differential pressure to be acceptable by comparison to Node 1 and Node 6 areas. The slabs in both areas are the same.

Peak Shell Pressure

This is the differential pressure across the containment shell to the outside, for nodes located in the ice condenser inlet areas closest to the refueling canal.

Original Design pressure	12.0 psi
Original Calculated pressure	10.8 psi
New Calculated pressure	14.0 psi

The licensee demonstrates the increased pressure to be acceptable by evaluation on a localized basis. The containment shell can handle pressures well in excess of the overall 12 psi design pressure. The average pressure over the structurally significant portion of the containment shell surrounding and including these nodes is smaller than the 12 psi containment shell design pressure.

Reactor Cavity

The reactor cavity is the structure surrounding the reactor with penetrations for the main coolant piping. This structure is designed to limit the adverse effects of the initial pressure response to a loss of coolant accident. The results of the reactor cavity analysis and evaluations for the D. C. Cook Nuclear Plant Unit 1 demonstrate that, for the reactor vessel annulus and pipe annulus, the resulting peak pressures at the RTP rerated conditions are within the FSAR design allowables. For the upper and lower reactor cavities the peak calculated pressures under RTP conditions exceeded the structural design pressures (Ref. 2, Sections 3.7.2 and 3.7.3) as stated in the FSAR. For these

areas, structural evaluations were performed for the revised peak pressures, and the structural adequacy of the containment subcompartment has been confirmed (Ref. 10) as follows:

Missile Shield, Refueling Canal Bulkhead Blocks, and Upper Reactor Cavity Wall Differential Pressures

The upper reactor cavity walls surround the reactor head. The missile shields and the refueling canal bulkheads are blocks separating the upper reactor cavity from upper containment. The missile shield is bolted down during operation, and is removable for refueling. The refueling canal bulkheads fit snugly in grooves in the upper reactor cavity walls.

	Cavity Wall	Missile Shield and Bulkheads
Original Design pressure	48.0 psi	48.0 psi
Original Calculated pressure	44.1 psi	44.1 psi
New Calculated pressure	48.4 psi	54.3 psi

The licensee demonstrates the increased pressure for the cavity wall to be acceptable by finite element analysis of the entire upper reactor cavity wall.

The licensee has demonstrated the increased pressure for the missile shields and the bulkheads to be acceptable by manual calculation. The test cylinder break strength of the concrete, which is higher than the design strength, was also taken into consideration.

Peak Lower Cavity Pressure

This is the cavity located under the reactor vessel. The peak pressure is used in the structural analysis rather than the differential pressure since most of the cavity walls are in the foundation mat.

Original Design pressure	15.0 psi
Original Calculated pressure	13.8 psi
New Calculated pressure	18.5 psi

The licensee demonstrated that the increased pressures are acceptable by manual calulation.

The staff concludes, based on the licensee's demonstration, that the D. C. Cook Nuclear Plant's design basis pertaining to containment short term response, as stated in Chapter 5.2.7.3 of the FSAR, is adequate for RTP operation, and therefore, is acceptable. The licensee must update the FSAR to reflect the higher structural design values.

Long Term Containment Pressure

The long term peak containment pressure analysis supports operation with the RHR crosstie valves closed at a power level of 3425 MWt for both Units 1 and 2 containment structure. This analysis contained additional justification for operation under the RTP conditions (Ref. 11) and was approved by the staff Safety Evaluation dated January 30, 1989 (Ref. 12).

2.7 NUCLEAR, PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

The Nuclear Sampling System (NSS) is designed to provide representative samples for laboratory analyses used to guide the operation of various primary and secondary systems throughout the plant during normal operation. Since reduction of sample pressure and temperature, when necessary, is already being done by heat exchangers and needle valves, the parameters associated with the RTP program do not affect the performance of the NSS. With no power uprating, the source term remains unchanged. Therefore, the staff concludes that operation under RTP conditions is acceptable for the NSS.

The staff finds that, since no power uprating is being proposed at this time, there is an insignificant effect on the post-accident containment thermal conditions and therefore the existing post-accident sampling system remains adequate and is acceptable.

Operation under RTP conditions results in slight reductions in secondary side temperatures and pressures with no change in the source term. The staff concludes that the change can be accommodated by the process sampling system without causing degradation of their performance, and is, therefore, acceptable.

2.8 ELECTRIC SYSTEMS DESIGN

Operation under RTP conditions results in minor changes to the heat balance. The only impact noted on the electrical systems is the slight increase in motor current for the motors used as prime movers of primary coolant. The required power is increased by the higher densities encountered due to the RTP program. The licensee has reviewed cable penetrations, busses, and motor ratings to conclude that there is sufficient design margin to handle the increased load. The staff finds, based on the licensee's evaluation, that the proposed RTP program minimally affects the electric power system and associated loads and is therefore, acceptable.

3.0 TECHNICAL SPECIFICATIONS

1. Definition 1.38 on design thermal power is being deleted on page 1-7 of the Technical Specifications (TS's) because there is no longer a single design thermal power at which all the transient and accident analyses have been performed. The licensed power level for Cook 1 remains 3,250 MWt. This change is acceptable.

2. Table 1-3 on page 1-10 is being deleted because it previously gave information on the analyses performed at the design thermal power. This change is acceptable because the definition of design thermal power is being deleted also.

3. Figure 2.1-1 on page 2-2 is being revised to reflect the revised DNBR safety limit of 1.45. This change is acceptable because it is supported by the safety analysis.

4. The pressurizer pressure low setpoint (Item 9 of Table 2.2-1 on page 2-5) is increased by 10 psig. This is acceptable because it was assumed in the large- and small-break LOCA analyses.

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3. Figure 2.1-1 on page 2-2 is being revised to reflect the revised DNBR safety limit of 1.45. This change is acceptable because it is supported by the safety analysis.

4. The pressurizer pressure low setpoint (Item 9 of Table 2.2-1 on page 2-5) is increased by 10 psig. This is acceptable because it was assumed in the large- and small-break LOCA analyses.

5. The Overtemperature-Delta T trip setpoint equation (pages 2-7 and 2-8) is being revised in terms of rated thermal power rather than design thermal power. In addition, this revised OTDT trip setpoint protects the core safety limits of Figure 2.1-1. This change is acceptable because it is supported by the non-LOCA safety analyses.

6. The Overpower-Delta T trip setpoint equation (page 2-9) is being revised to reflect the revised core safety limits of Figure 2.1-1. This equation is also being defined in terms of the indicated T_{avg} at rated thermal power. These changes are acceptable because they are supported by the safety analysis for the RTP program.

7. Technical Specification 3.2.2 on page 3/4 2-5 is being revised from a maximum F_Q of 2.10 to 2.15. This change is acceptable because it is supported by the large-break LOCA analysis. The F_Q values for Exxon fuel are being deleted because this fuel will no longer be used at Cook Unit 1.

8. The K(Z) curve applicable to Exxon fuel (page 3/4 2-7) is being deleted. This is acceptable because Exxon fuel will no longer be used at Cook Unit 1.

9. The K(Z) curve for Westinghouse fuel (page 3/4 2-8) is being revised. This is acceptable because it is supported by the new LOCA analysis for Cook Unit 1.

10. The F-Delta H limit applicable to Exxon fuel (page 3/4 2-9) is being deleted. This is acceptable because Exxon fuel will no longer be used at Cook Unit 1.

11. Table 3.2-1 on page 3/4 2-14 on DNB parameters is being revised. T_{avg} must be less than or equal to 570.9°F, the pressurizer

pressure must be less than or equal to 2050 psig, and the reactor coolant system total flow rate must be greater than or equal to 366,400 gpm. These changes are acceptable because they reflect the safety analysis for the RTP program.

12. Technical Specification 3.2.6 on page 3/4 2-15 is being revised to change F_Q in the APL limit to 2.15. This change is acceptable because it reflects the new F_Q limit of Specification 3.2.2. The limits on APL applicable to Exxon fuel are being deleted because Exxon fuel will no longer be used at Cook Unit 1.

13. Functional Units 2 and 11 of Table 3.2-2 on page 3/4 3-10 are being changed. Functional Unit 2 incorporates an editorial change to indicate that the response time is applicable to both the high and low setpoints of the Power Range Neutron Flux trip. This change is acceptable because it is editorial in nature. Functional Unit 11 is being changed from a response time of "not applicable" to "equal to or less than 2 seconds." This is acceptable because this trip on pressurizer water level-high was modeled in the analysis of the control rod withdrawal-at-power event.

14. Functional Units 1.f and 4.d of Table 3.3-4 on pages 3/4 3-24 and 3/4 3-26 are being changed to decrease the steamline pressure low setpoint by 100 psig. These changes are acceptable because they are supported by the steamline break analysis and the steamline break mass and energy evaluations.

15. Technical Specification 3.4.4 on page 3/4 4-6 is being revised to 92% of span. This change is acceptable because it is supported by the safety analysis.

16. Technical Specification 3.5.1.b on page 3/4 5-1 is being revised from an accumulator borated minimum water volume of 929 to 921 cubic feet. This change is acceptable because it is consistent with the LOCA analysis for Cook Unit 1.

17. Surveillance Requirement 4.5.2.f is being revised to reduce the discharge pressure of the safety injection pump and the residual heat removal pump. These changes are acceptable because they are consistent with the LOCA analyses.

18. Surveillance Requirement 4.5.2.h is being revised by adding a requirement to verify that the charging pump discharge coefficient is within a specified range following ECCS modifications. The footnote is broken into four parts for clarity. This change is acceptable because it ensures that the flow delivered to the core by the charging pumps in the event of a LOCA is within the analyzed values.

19. Surveillance Requirement 4.7.1.2 on page 3/4 7-5 is being revised to change the discharge pressure requirements of the motor and turbine driven auxiliary feedwater pumps to 1375 psig and 1285 psig, respectively. This corresponds to a 5% degradation of the pumps

from the manufacturer's pump head curve. These changes are acceptable because they are consistent with the changes for the RTP program.

20. Basis page B 2-1(a) is being changed to incorporate the design limit and safety analysis limit DNBR values. The DNB limits for Exxon fuel are being deleted since Exxon fuel is no longer used at Cook Unit 1. The design limit and safety analysis limit DNBR values are acceptable because they are consistent with the RTP program.

21. Basis page B 2-2 is being revised to delete reference to F-Delta H for Exxon fuel and to design thermal power. These changes are acceptable because references to both items have been deleted in the Specifications.

22. Bases page B 2-4 is being revised to reflect the changes to the Overtemperature-Delta T trip function. The changes are acceptable because they reflect changes made to the Specifications.

23. Bases page B 2-5 is being revised to reflect the changes to the Overpower-Delta T trip function and the pressurizer water level-high trip. These changes are acceptable because they reflect changes to the Specifications.

24. Bases page B 3/4 2-1 is being revised to replace the minimum DNBR value of 1.69 by the words "the safety limit DNBR". This change is acceptable because it will avoid changes to the Bases if the safety limit DNBR value is changed.

25. Surveillance Requirement 4.1.1.5.b is being changed to require T_{avg} determination of T_{avg} every 30 minutes when the reactor is critical and T_{avg} is less than 545°F. This change is supported by Reference 9 and allows a full power T_{avg} of 550°F for Cook Unit 1 Cycle 11 without requiring a monitoring every 30 minutes while at full power, which the previous value of 551°F would have required. This change is acceptable because the intent of maintaining the minimum coolant temperature for criticality of Specification 3.1.1.5 is preserved.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on June 9, 1989(54 FR 24774). Accordingly, based upon the environmental assessment, we have determined that the issuance of he amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The staff has reviewed the request by the Indiana and Michigan Power Company to operate the Donald C. Cook Nuclear Plant Unit 1 at the reduced temperatures and pressures of the RTP program. Reactor operation is restricted to an upper limit on T_{avg} of 567.8°F because the steamline break mass and energy release inside containment was not reanalyzed as part of the RTP program. Although the safety analysis was performed at power ratings which would support a possible power uprating for Cook Unit 1, power uprating is not addressed in the staff's review. The power of D.C. Cook Nuclear Plant Unit 1 is limited to the present rated thermal power of 3250 MWt. Based on its review, the staff concludes that appropriate material was submitted and that normal operation and the transients and accidents that were evaluated and analyzed are acceptable. The Technical Specifications submitted for this license amendment suitably reflect the necessary modifications for the operation of Cook Unit 1.

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 9, 1989

Principal Contributors: Dan Fieno
John Stang, NRR
Anthony Gody, NRR

6.0 REFERENCES

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8. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

9. Letter (AEP:NRC:1067A) from M. P. Alexich (Indiana and Michigan Power Company) to the USNRC, dated December 30, 1988.

10. Letter (AEP:NRC:1067C) from M. P. Alexich (Indiana and Michigan Power Company) to the USNRC, dated March 14, 1989.

11. Letter (AEP:NRC:1024D) from M. P. Alexich to T. E. Murley (NRC), dated August 22, 1988. Includes WCAP-11908, "Containment Integrity Analysis for Donald C. Cook Nuclear Plants, Units 1 and 2."

12. Letter, J. F. Stang (NRC) to M. P. Alexich (IMECo), dated January 30, 1989.