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EFFECT OF RADIATION DAMAGE ON SM-1, SM-1A AND PM-2A REACTOR VESSELS

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October 14, 1961

Nuclear Power Engineering Department
Alco Products, Inc.
Schenectady, New York

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REACTOR TECHNOLOGY

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ABSTRACT

The report covers the status of the SM-1, SM-1A and PM-2A reactor vessels with regard to effect of irradiation on nil-ductility transition temperature and the associated problem of brittle fracture. The three vessels are reported not subject to brittle fractures at present because of restrictions placed on operation during startup and shutdown. At temperatures where brittle fracture could occur, stresses were limited to 18% of the yield strength of the vessel material. Vessel stresses were limited by controlling the primary system pressure, cooling and heating rates and core power level.

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1.0 INTRODUCTION

Early in September 1961, the Army Nuclear Power Program received a questionnaire from the Division of Licensing and Regulation, U. S. Atomic Energy Commission, in regard to the status of the radiation damage problem in the reactor pressure vessels of the SM-1, SM-1A and PM-2A. By separate correspondence, the Commission posed an additional question, raised by the Advisory Committee for Reactor Safeguards (ACRS), in regard to the hazards presented by a potential brittle fracture failure of a plant pressure vessel.

ALCO Products was requested by the AEC New York Operations Office to prepare the answers to these questions, based on the results obtained thus far from the radiation damage study being conducted under Task 6 of the PWR Support Program for the Army plants.

The questions are restated in this section, for reference. Questions 1 - 5 are those of the Division of Licensing and Regulation; Question 6 was added by the Army Nuclear Power Program to provide a basis for responding to the other questions; Question 7 was originally posed by the ACRS. The information on radiation damage is presented in a different sequence than that in which the questions were listed, in order to provide continuity to the report. However, all questions have been answered.

QUESTIONS CONCERNING IRRADIATION EFFECTS ON REACTOR PRESSURE VESSELS

REPLY CONTAINED IN SECTION

3.0

1. The calculated yearly integrated neutron dose (nvt) in each of the following regions of the reactor vessel:

   a. The belt region i.e. the region of the vessel which is exposed to the maximum neutron flux.

   b. The nozzle or penetration region of the vessel.

   c. The region of the vessel support brackets.

   d. Any other region where the integrated neutron dose in conjunction with a high stress level may be of concern when evaluating the long term integrity of the vessel.

QUESTIONS (CONT'D)

REPLY CONTAINED IN SECTION

When calculating the integrated neutron dose, a cut-off point of 1 Mev should be used in order to correlate the results which are presently available. However, the integrated neutron dose for the energy band between 0.1 Mev and 1 Mev should be given as a separate number in each case.

3.0 2. The calculated integrated neutron dose from beginning of reactor to date in all of the above mentioned regions of the vessel.

5.0 3. The results of the stress analysis of the vessel taking into account the following loadings and conditions:

a. The internal pressure.

b. The pipe reactions at those nozzles which receive a significant amount of integrated neutron dose, nvt.

c. Thermal conditions under steady operations at full power.

d. Thermal conditions resulting from start-up and shut-down.

e. Thermal conditions resulting from abnormal conditions such as emergency shut-down (scram) and emergency cooling conditions.

Stresses need only be given for the regions identified under question (1) above, and where the calculated integrated neutron flux for 20 years of operation is above 10^{18} nvt.

4.0 4. Significant data and information about the materials of construction of the vessel, such as:

a. Its chemical analysis.

b. Its physical test data including its initial NDT

c. Method of cladding the stainless steel overlay to the base metal.

d. Sensitivity of radiography and other non-destructive inspection tests used.

QUESTIONS (CONT'D)

REPLY CONTAINED IN SECTION

7.0 — 5. Information concerning surveillance and inspection of the vessel which should include:

a. Provisions and procedures for inspecting and testing the vessel during its operational lifetime.

b. Proposed provisions and procedures for monitoring the vessel during its operational lifetime.

c. Limitations which have been imposed or are intended to be imposed on the allowable pressure during conditions of transient temperatures which may result in reactor vessel temperatures below the FTE (fracture transition elastic) of the material, and an evaluation of the adequacy of these procedures in preventing a drastic failure of the reactor vessel at the present time and as damage to the vessel material increases through its operational lifetime.

ADDITIONAL QUESTIONS

6.0 — 6. The design criteria employed to relate basic NDT data and stress analyses, to establish operating limitations for the pressure vessel to include:

a. The basis or justification for such criteria, in the absence of formal codes.

b. The degree of conservatism believed to exist in such criteria.

8.0 — 7. An analysis of the environmental effects of a complete brittle failure of the pressure vessel originating in that portion of the vessel in which the integrated neutron dose in conjunction with the expected levels during normal or abnormal operations indicates the greatest likelihood of initiation of failure; to include:

a. The pressure rise within the containment vessel.

b. The ability of the containment vessel to remain intact under such a pressure loading.

c. The effect of possible missiles on containment vessel integrity.

QUESTIONS (CONT'D)

REPLY CONTAINED IN SECTION

d. An evaluation of the amount of core melt-down and amount and kind of fission product release to the containment vessel.

e. An estimate of the amount and kind of fission product release from the containment to the environs.

f. A discussion of the resultant hazards to on-site and off-site personnel.

2.0 SUMMARY - EFFECT OF RADIATION DAMAGE ON THE ARMY REACTORS

Radiation damage as it is used here, refers to the large increase in the nil-ductility transition temperature (NDT) experienced by ferritic steels subjected to sustained irradiation by neutrons of energies greater than 1 Mev. The NDT is the lower boundary of the descending temperature range over which many classes of normally ductile steels exhibit a transition to completely brittle fracture behavior. The relationship of fracture characteristics to the NDT temperature and the FTE (fracture transition for elastic loading) temperature, which lies about 60° above the NDT near the middle of the transition range, was clearly established by the Naval Research Laboratory during the investigation of welded ship failures in the early 1940's. It was concluded that a structure operated at or below its NDT temperature could fail in a brittle manner at customary engineering stress levels provided that a stress concentration sufficient to cause local yielding was present.

The consequence of a radiation-induced rise in NDT temperature to a level approaching reactor operating temperatures is that during startup and shutdown the vessel is operating at temperatures where brittle fracture is a possibility. The concept that was used to preclude brittle fracture in the Army vessels was to control the vessel stresses within the limits established in the design criteria discussed in Section 3.4 of this report. Stresses were limited by controlling the primary system pressure, cooling and heating rates, and core power level as a function of temperature.

The status of the SM-1, SM-1A and PM-2A vessels with regard to radiation damage and the associated problem of brittle fracture is summarized briefly in Table 1. It is apparent from this summary and from the detailed information presented in the following sections of this report that the three vessels are not subject to brittle fracture at the present time because of the restrictions that have been placed on their mode of operation during startup and shutdown.

The satisfactory status of the Army reactor vessels with respect to radiation damage is not a coincidence. The possible detrimental effect of irradiation on NDT temperature has been recognized by the Army Nuclear Power Program for some time. In the initial design of the SM-1, Alco included a group of subsize Izod impact specimens (used to measure NDT) prepared by ORNL and mounted above the core support ring inside the vessel. In the Alco report on the selection of the vessel material for the SM-2, the early data on transition temperature shifts caused by irradiation were recorded and discussed. Finally, a joint Army-Alco Products Naval Research Laboratory program has been in existence since September, 1960, to define the extent of radiation-caused changes in the Army vessel steels and to determine what controls and corrective action were required to assure safety of operation. During the past year, the Naval Research Laboratory oriented their existing research program on radiation damage to obtain data on the upward shift in transition temperature from irradiated samples of Army vessel materials. Alco defined the engineering criteria for applying these data and used them to establish operating boundaries for the existing vessels. The Army NPFO then established operating curves for each vessel within the prescribed boundaries. To summarize, the FY' 61 program defined the problem and established the initial conservative operating restrictions. The FY' 62 program is directed towards refining the current restrictions and exploring the effects of neutron radiation on other materials properties such as resistance to fatigue and strain-cycling failure.

The operating limits for each reactor are valid for a specified period of time. The means of extending that period are being explored for each reactor as part of a planned program to have the necessary corrective action defined for each plant by the time it is required. No corrective action is required for the SM-1 during its 20 year design life unless plant utilization above the current 46% level is desired. Corrective action for the SM-1A consists primarily of obtaining data on the increase in yield strength of the vessel steel as a function of integrated neutron flux, then revising the operating limits on the basis of this data. Another primary effort involves experimental measurement of the axial distribution of flux at the vessel wall through analysis of the flux monitors which have been installed in the vessel. Preliminary studies on corrective action for PM-2A indicate that an extension of service life can be realized by utilizing increased yield strength data and by a reduction in system operating pressure supported by data from planned tests (Task XIV) on core boiling using an instrumented fuel element. However, these steps will not be sufficient to extend the operating life to a full 20 years. Further action in the form of in-place annealing of the pressure vessel will probably be required and is being investigated.

TABLE 1

RADIATION DAMAGE STATUS OF SM-1, SM-1A, AND PM-2A VESSELS

Reactor	Location	Vessel Material	Operating Conditions		Maximum nvt (> Mev)		Operating Limits Established ?	Service Life Based On Belt Line Nvt		Requirements For Extending Service Life
			Pressure, psia	Temperature, °F	nvt/MWYR	nvt For 20 Yrs. at Load Factor		MWYR	Calendar Yrs @ Load Factor	
SM-1	Ft. Belvoir, Va.	A-212B	1200	430	1.18×10^{17}	1.09×10^{19}* @ 46%	Yes See Fig. 11	92.5	20 @ 46%	None Required
SM-1A	Ft. Greeley, Alaska	SA-350LF1 (modified)	1200	430	3.35×10^{17}	8.05×10^{19}* @ 60%	Yes See Fig. 12	62.6	5.2 @ 60%	Measure increase in irradiated yield strength. Life can be extended to 626 MWYRS.
PM-2A	Camp Century, Greenland	SA-350LF3	1750	510	2.67×10^{18}	2.13×10^{20} @ 40%	Yes See Fig. 13	12.3	3.1 @ 40%	1. Measure increase in irradiated yield strength. 2. Reduce pressure and/or power. 3. Anneal vessel. 1 + 2 can extend life to 10 yrs. Further extension may be obtained by 3.

* Calculated value normalized to SM-1 mockup measurements with 30% added to account for probable error.

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3.0 DETERMINATION OF INTEGRATED FAST NEUTRON FLUX

3.1 INTRODUCTION

Structural damage within a reactor vessel is related to both the spatial and energy dependent neutron flux.[1] Accordingly, a comprehensive experimental and analytical program was initiated under Task 6.7 of the PWR Research and Development Program to determine the best estimates of the integrated fast neutron flux (nvt) distribution for the SM-1, SM-1A, and PM-2A reactors. This program was initiated during FY-61 and will be continued during FY-62.

Prior to this program, estimates of the nvt values were based only upon one-dimensional analytical calculations, employing two-group theory. The availability of experimental data upon the various reactors was negligible. The lack of data, together with the crudeness of the analytical models, made the task of estimating the errors in the calculated nvt distributions nearly impossible. Any attempt to estimate the accuracy of the calculations was based primarily upon engineering judgment and experience; in general, very conservative assumptions were built into the calculations, resulting in overestimates of the nvt distribution, as was verified during the work performed under Task 6.7. The following sections describe the experimental and analytical programs being performed to obtain more accurate nvt estimates for the three Army vessels, and the results obtained to date.

3.2 REVIEW OF WORK PERFORMED IN FISCAL 1961

3.2.1 Experimental Measurements

Flux monitor capsules prepared by NRL were placed in the SM-1 vessel. The first capsule, inserted in one of the subsize Izod specimen holders mounted above the core in the core support ring, has been removed and is being analyzed. A second more complete capsule, installed in April 1961, is scheduled for removal during October 1961. The dosimeters were cobalt, cadmium-shielded cobalt, nickel, and sulfur, which have threshold energies of 0.025 ev, 0.4 ev, 5.0 Mev and 2.9 Mev respectively. They will provide flux measurements which are directly applicable to the subsize Izod specimens and which can be related to the vessel wall flux through correlation with measurements made on the SM-1 core mockup in the Alco Critical Facility.

Neutron activation experiments were performed upon mockups of the SM-1 and PM-2A at the Alco Critical Facility, using various high energy threshold types of detectors. The mockups included the cold, clean core, reflector, thermal shields, and pressure vessel of the SM-1 and PM-2A reactors. Experiments were performed, at room temperature, by irradiating activation foils of sulfur, aluminum, bare and cadmium-covered gold, and uranium-238, to establish the neutron intensity and approximate neutron energy spectra in core regions between the core centerline and the pressure vessel wall. Effective threshold energies for the S^{32} (n, p) P^{32}, Al^{27} (n, p) Mg^{27} (n, p) Mg^{27}, and Al^{27} (n,α) Na^{24} reactions, are 2.9 Mev, 5.3 Mev, and 8.6 Mev, respectively.(2) The cadmium-covered gold foils were used to measure the neutron flux in the resonance energy region.

Limited axial distributions were measured with primary emphasis placed upon the axial plane at which the maximum nvt occurs on the vessel. The experimental program summarized here is reported in APAE No. 95.(3)

3.2.2 Analytical Program

The integrated fast flux above numerous threshold energies was calculated for the cold and hot, clean SM-1, SM-1A and PM-2A reactors. In order to obtain a detailed knowledge of both the spatial and energy dependent neutron flux, two calculational models were employed:

a. P1MG-2(4): A one-dimensional, IBM-704 code, which calculates the spatially dependent multigroup flux, taking into account the slowing down of neutrons from elastic and inelastic scattering, the removal of neutrons from epithermal capture and fission resonances, and the regeneration of fast neutrons from epithermal fissioning or in the one thermal group.

b. PDQ-2(5): A two-dimensional, IBM-704 code, which calculates the few-group fluxes in (r, z) geometry.

The P1MG-2 code was used to determine the multigroup flux distribution along the radial (r) direction. The results would be applicable for an average axial plane. To account for the axial variation of the flux, PDQ (r, z) calculations were employed utilizing modified two-group constants; fast parameters were calculated by the MUFT-3 code(6), while thermal parameters were obtained by P_3 theory.(7)

The calculations for the cold SM-1 mockup were compared to the measured flux distributions to determine the accuracy of the analytical models. A correction factor of 1.14, representing the ratio of cold measured to cold calculated values, was obtained and applied to the calculations for the hot, clean reference SM-1 reactor. Due to the similarity between the SM-1 and SM-1A, the same correction factor was assumed for both reactors, with proper modifications made for the geometrical and power level differences.

At the time of this report, the data for the PM-2A mockup was not fully reduced to allow direct comparisons between analytical and measured data. Therefore, SM-1 correction factors were applied to the PM-2A calculations for the purposes of this report only.

3.2.3 Results

The results of the calculations for the SM-1 mockup are shown in Fig. 1 through 4 and tabulated below. Tabulations of the best estimate of neutron flux

FIGURE I

COMPARISON OF PIMG-2 RESULTS WITH THE Cd COVERED Au FOIL MEASUREMENTS FOR THE SM-I MOCK-UP (68° F, 0 MWYRS)

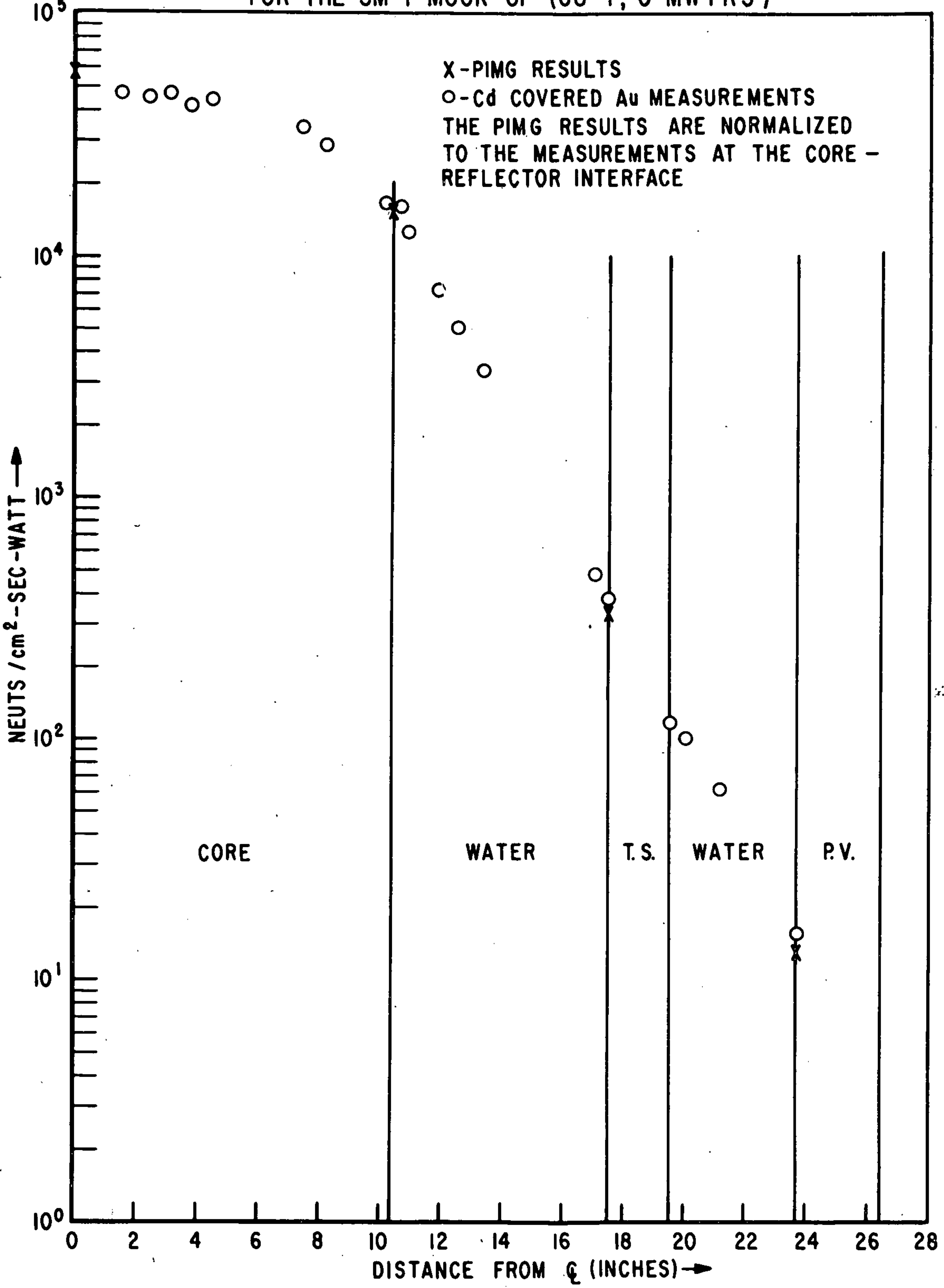


FIGURE 2
COMPARISON OF PIMG-2 CALCULATIONS WITH
THE S (n, p) THRESHOLD MEASUREMENTS (En>2.9 MEV)
FOR THE SM-1 MOCK-UP (68° F, 0 MWYRS)

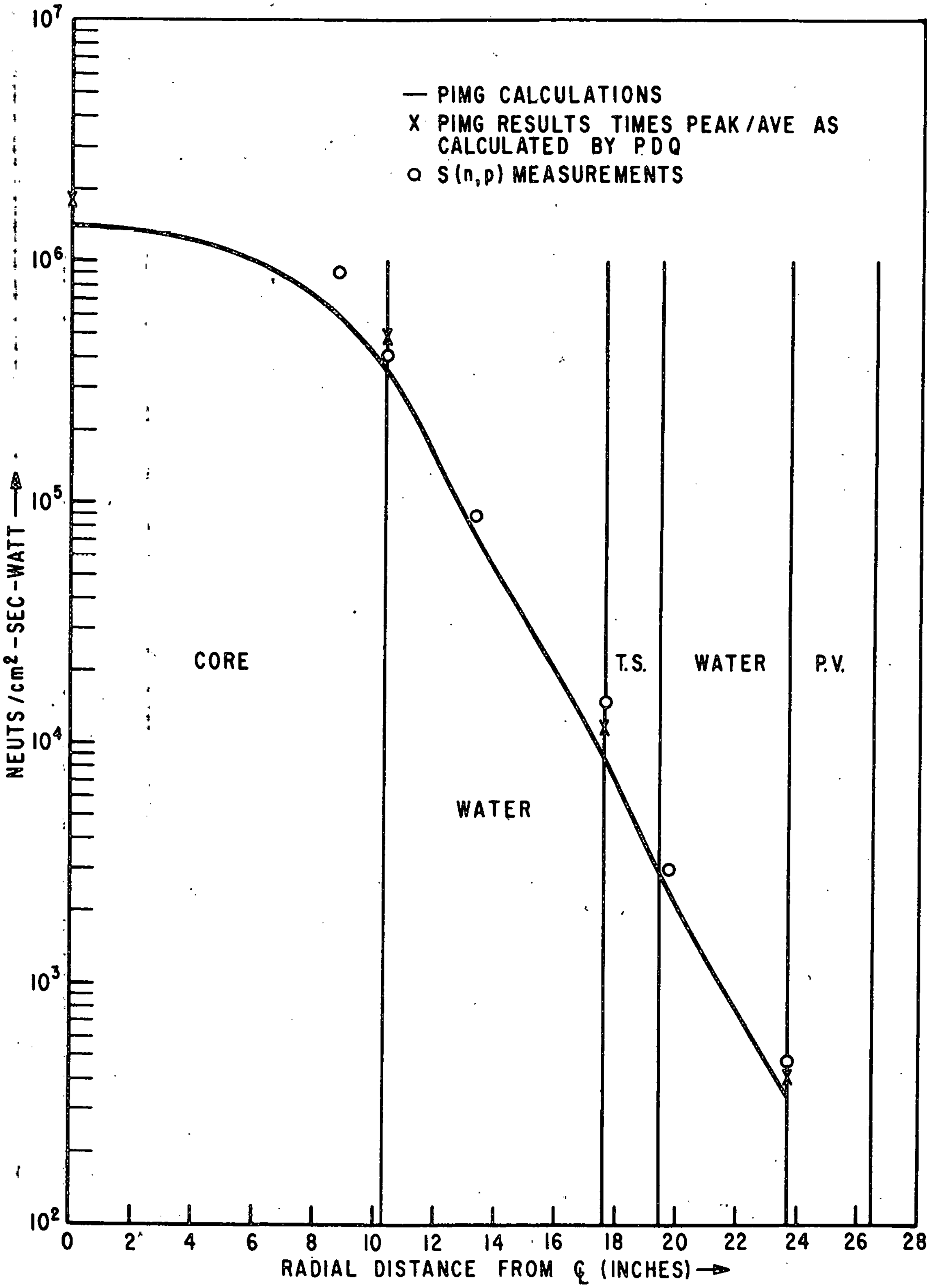


FIGURE 3
COMPARISON OF PIMG-2 CALCULATIONS WITH THE Al (n,p) THRESHOLD MEASUREMENTS (En>5.3 MEV) FOR THE SM-1 MOCK-UP (68° F, 0 MWYRS)

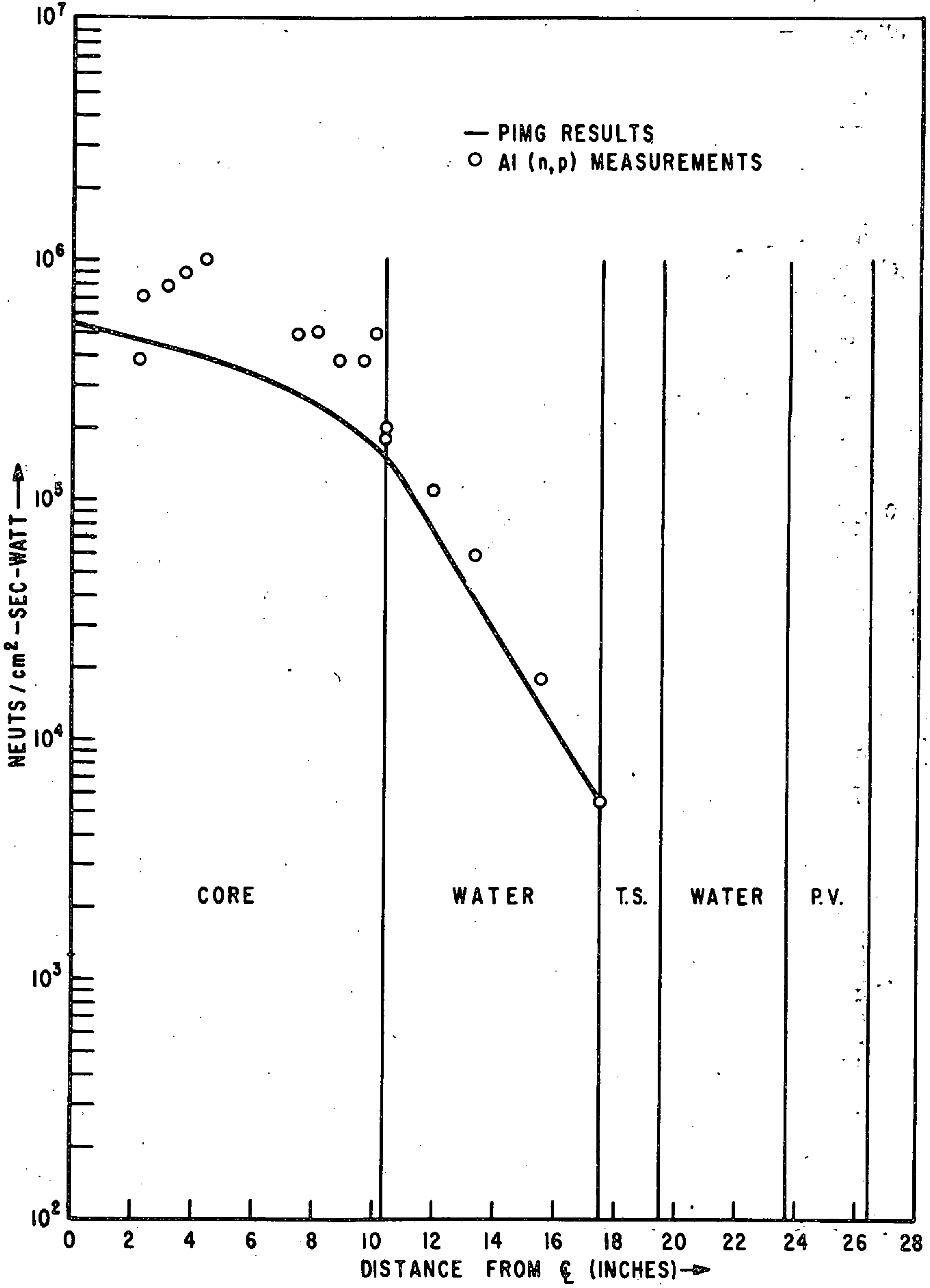
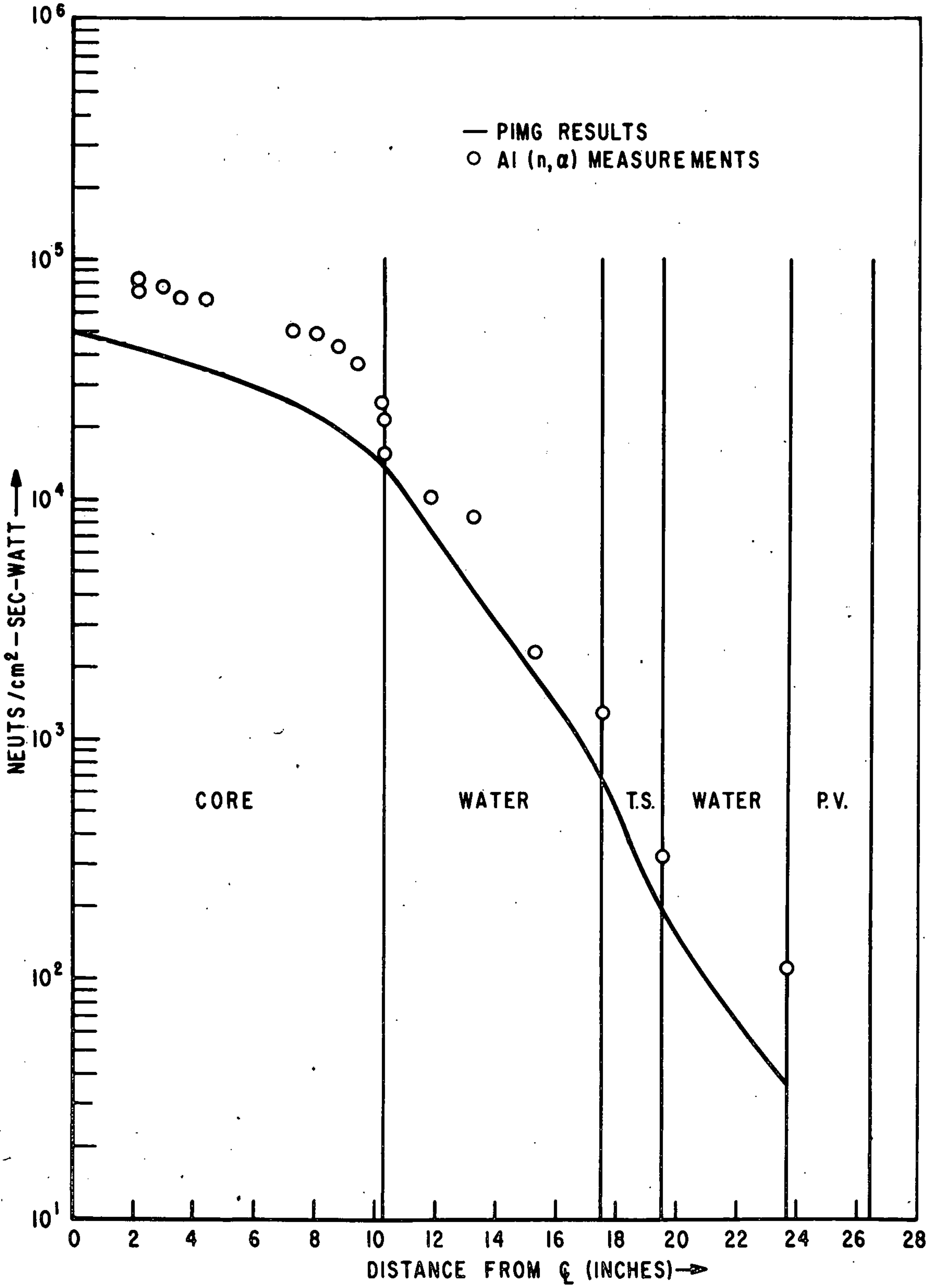


FIGURE 4
COMPARISON OF PIMG-2 CALCULATIONS WITH
THE Al (n,α) THRESHOLD MEASUREMENTS (En>8.6 MEV)
FOR THE SM-1 MOCK-UP (68°F, 0 MWYRS)



at positions of interest in the SM-1, SM-1A and PM-2A vessels are also presented. The >1 Mev and the 0.1 to 1 Mev values are listed separately. It is noted that the ratio of neutrons in the 0.1 to 1 Mev range to neutrons with energies greater than 1 Mev is higher at the SM-1A pressure vessel than at the SM-1 vessel. This is because the SM-1A vessel contains two thermal shields whereas the SM-1 contains only one. While water and steel are nearly equal in degrading neutrons of high energies to below 1 Mev, water is much more effective than steel in slowing down neutrons in the 0.1 to 1 Mev range. Hence, the slowing down of neutrons in the lower energy range is not as effective in SM-1A, and more neutrons in this energy group reach the pressure vessel.

3.2.3.1 SM-1 Mockup

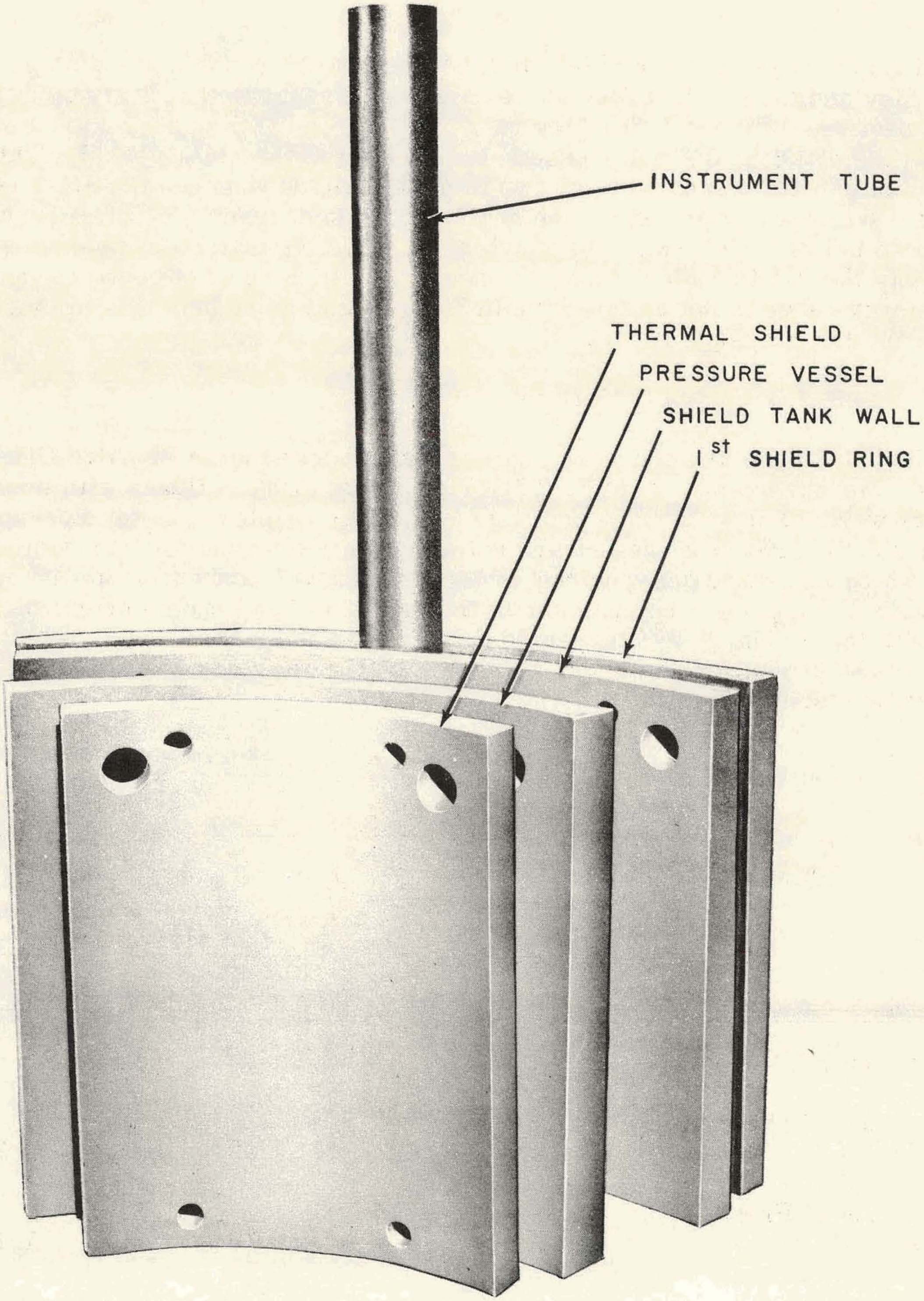
The SM-1 core and non-core regions were mocked up at the Alco Critical Facility. Measurements at 68°F, 0 MWYR, of fast neutron fluxes with threshold detectors were taken at various positions within the reactor. In the mockup, the thermal shield, pressure vessel, and shield rings comprised an abbreviated assembly which encircled one quadrant of the actual SM-1 concentric arrangement as shown in Figure 5. A comparison of the calculated and experimental measurements with the S^{32} (n, p) P^{32} threshold detector (En > 2.9 Mev), is shown as a function of radial position in the SM-1 reactor in Fig.6; values at certain specific locations are tabulated below.

TABLE 2
SM-1 MOCKUP FLUX - En > 2.9 Mev

Position	Measured Flux (neuts/cm^2-sec-watt)	Calculated Flux (neuts/cm^2-sec-watt)	Ratio Measured/ Calculated
Core Centerline	-	1.8×10^6	-
Core-Reflector Interface	4.0×10^5	4.8×10^5	0.83
Inner Surface of Thermal Shield	1.5×10^4	1.2×10^4	1.25
Inner Surface of Pressure Vessel	4.8×10^2	4.2×10^2	1.14

3.2.3.2 SM-1 Reactor

The fast neutron fluxes at various positions of interest in the actual SM-1 (440°F, 0 MWYR) are given below in units of neutrons/cm^2. They are calculated values modified by normalizing to the measured S^{32} (n, p) P^{32} value at the pressure vessel beltline at 68°F, start-of-life, and increased by 30% to account for probable error.



SM-1 SHIELD MOCKUP
FIGURE 5

TABLE 3
SM-1 VESSEL WALL FLUX

Location	Nvt Per MWYR (En >1 Mev)	Nvt to Date/ (En >1 Mev)	Nvt (En >1 Mev) (20 yr. Period, 46% l. f.)
Pressure Vessel*	1.18×10^{17}	2.29×10^{18}	1.09×10^{19}
Support Ring	1.89×10^{16}	3.65×10^{17}	1.74×10^{18}
Nozzle Weld	3.94×10^{15}	7.60×10^{16}	3.62×10^{17}

/ Based upon 19.3 MWYR total energy release, as of September 25, 1961.
* Value in axial plane of maximum exposure.

Location	Nvt Per MWYR En=(0.1-1.0) Mev	Nvt to Date/ En=(0.1-1.0) Mev	Nvt, En=(0.1-1.0) Mev (20 yr. Period, 46% l. f.)
Pressure Vessel*	1.10×10^{17}	2.11×10^{18}	1.01×10^{19}
Support Ring	1.75×10^{16}	3.37×10^{17}	1.61×10^{18}
Nozzle Weld	3.65×10^{15}	7.04×10^{16}	3.36×10^{17}

/ Based upon 19.3 MWYR total energy release, as of September 25, 1961.
* Value in axial plane of maximum exposure.

3.3.2.3 SM-1A Reactor

Similar calculations of integrated fast neutron fluxes were performed at points of interest in the SM-1A (440°F, 0 MWYR) as were performed for the SM-1. The results tabulated below are calculated values modified by the measured-to-calculated ratio for the SM-1 cold, clean mockup. In addition, they were increased by 30% to account for probable error.

TABLE 4
SM-1A VESSEL WALL FLUX

Location	Nvt Per MWYR (En >1 Mev)	Nvt to Date/ (En >1 Mev)	Nvt (En >1 Mev) (20 yrs., 60% l. f.)
Pressure Vessel*	3.35×10^{17}	None	8.05×10^{19}
Support Ring	5.38×10^{16}	None	1.29×10^{19}
Nozzle Weld	3.35×10^{16}	None	8.05×10^{18}

/ Not operational as of September, 1961.
* Value in axial plane of maximum exposure.

Location	Nvt Per MWYR En=(0.1-1.0) Mev	Nvt to Date ⁄ En=(0.1-1.0) Mev	Nvt En=(0.1-1.0) Mev (20 yrs., 60% l.f.)
Pressure Vessel*	8.29×10^{17}	None	1.99×10^{20}
Support Ring	1.33×10^{17}	None	3.18×10^{19}
Nozzle Weld	8.29×10^{16}	None	1.99×10^{19}

⁄ Not operational as of September, 1961.
* Value in axial plane of maximum exposure.

3.2.3.4 PM-2A Reactor

Integrated fast neutron fluxes in neutrons/cm^2 at crucial positions in the PM-2A (510°F, 0 MWYR) are tabulated below. The calculations, at all points, were modified by the ratio of the measured-to-calculated fast flux measurements of the SM-1 cold, clean mockup. No correction for probable error could be made because of the differences in equivalent core radius, pressure vessel diameter, and operating temperature between SM-1 and PM-2A.

TABLE 5
PM-2A VESSEL WALL FLUX

Location	Nvt Per MWYR (En > 1 Mev)	Nvt to Date ⁄ (En > 1 Mev)	Nvt, En > 1 Mev (20 yrs., 40% l.f.)
Pressure Vessel*	2.67×10^{18}	5.33×10^{18}	2.13×10^{20}
Decay Cooling Nozzle	8.92×10^{17}	1.78×10^{18}	7.14×10^{19}
Outlet Nozzle Weld	7.47×10^{17}	1.49×10^{18}	5.98×10^{19}
Vessel Flange Weld	4.67×10^{17}	9.34×10^{17}	3.74×10^{19}
Support Brackets	Negligible	Negligible	Negligible

⁄ Based upon 2 MWYR total energy release as of September, 1961.
* Value in axial plane of maximum exposure.

Location	Nvt Per MWYR En=(0.1-1.0) Mev	Nvt, En=(0.1-1.0) Mev (To Date)/	Nvt, En=(0.1-1.0) Mev(20yrs. 40% l.f.)
Pressure Vessel*	3.25×10^{18}	6.50×10^{18}	2.60×10^{20}
Decay Cooling Nozzle	1.09×10^{18}	2.18×10^{18}	8.74×10^{19}
Outlet Nozzle Weld	9.08×10^{17}	1.82×10^{18}	7.28×10^{19}
Vessel Flange Weld	5.70×10^{17}	1.14×10^{18}	4.56×10^{19}
Support Brackets	Negligible	Negligible	Negligible

/ Based upon 2 MWYR total energy release as of September, 1961.
* Value in axial plane of maximum exposure.

3.2.4 Estimate of Error

It is seen from Fig. 1 through 4 that the agreement between as-calculated and measured flux distributions improves considerably as the value of the threshold energy decreases. In addition, the agreement is considerably better in the radial reflector regions than in the core region for any given threshold energy.

In particular, for the sulfur data ($E_n > 2.9$ Mev), measured values are 14% higher than the analytical results in the reflector regions. For thresholds of 1 Mev and 0.1 Mev, the energies of interest to this report, the agreement is expected to be even better. However, in lieu of actual measurements, the correction factor obtained from the sulfur data is applied to the 1 Mev and 0.1 Mev calculations, for conservatism.

Even though the calculations for the cold, clean SM-1 mockup core have been normalized to experimental measurements, several possible sources of error remain. These include:

a. Experimental Error: The largest source of error inherent in the measurements is introduced by the relatively low power at which the mockup core is operated; the degree of activation of the foils is dependent upon the power level, which effects the statistical errors involved in measuring the count rates. The error involved has been conservatively estimated, by the experimentalists involved, as ±25%.

b. Temperature Effects: The measurements were performed upon a cold, clean core. At the present, no experimental data is available for hot mockup cores. Even though there is no direct evidence at this time that the agreement between analytical and measured results should be strongly dependent upon system temperature, a nuclear temperature uncertainty factor of ±10% is assumed, for conservatism.

c. Core Burnup Effects: The calculations and measurements were performed for rod bank positions corresponding to start of core life. This does not affect the wall flux results even though the movement of the bank during core life causes the axial flux distribution to vary in the core. The effect is important only within the core and in the immediately adjacent radial reflector regions. Since axial fast flux distributions flatten considerably as the distance into the radial reflector increases,[8] the vessel beltline will receive a fairly constant irradiation level during core life. In the regions above and below the active core, the fast flux also varies quite slowly. Even though there is no evidence at this time that the agreement between calculated and measured data at the pressure vessel surface is a function of core life, an additional nuclear uncertainty factor of ± 10%, to allow for possible core burnup effects, was assumed.

Finally, the probable error associated with the calculated fast neutron flux is obtained from the square root of the sum of the squares of the individual errors of the items discussed above. Thus the probable error is,

$$e = \sqrt{(0.25)^2 + (0.10)^2 + (0.10)^2}$$

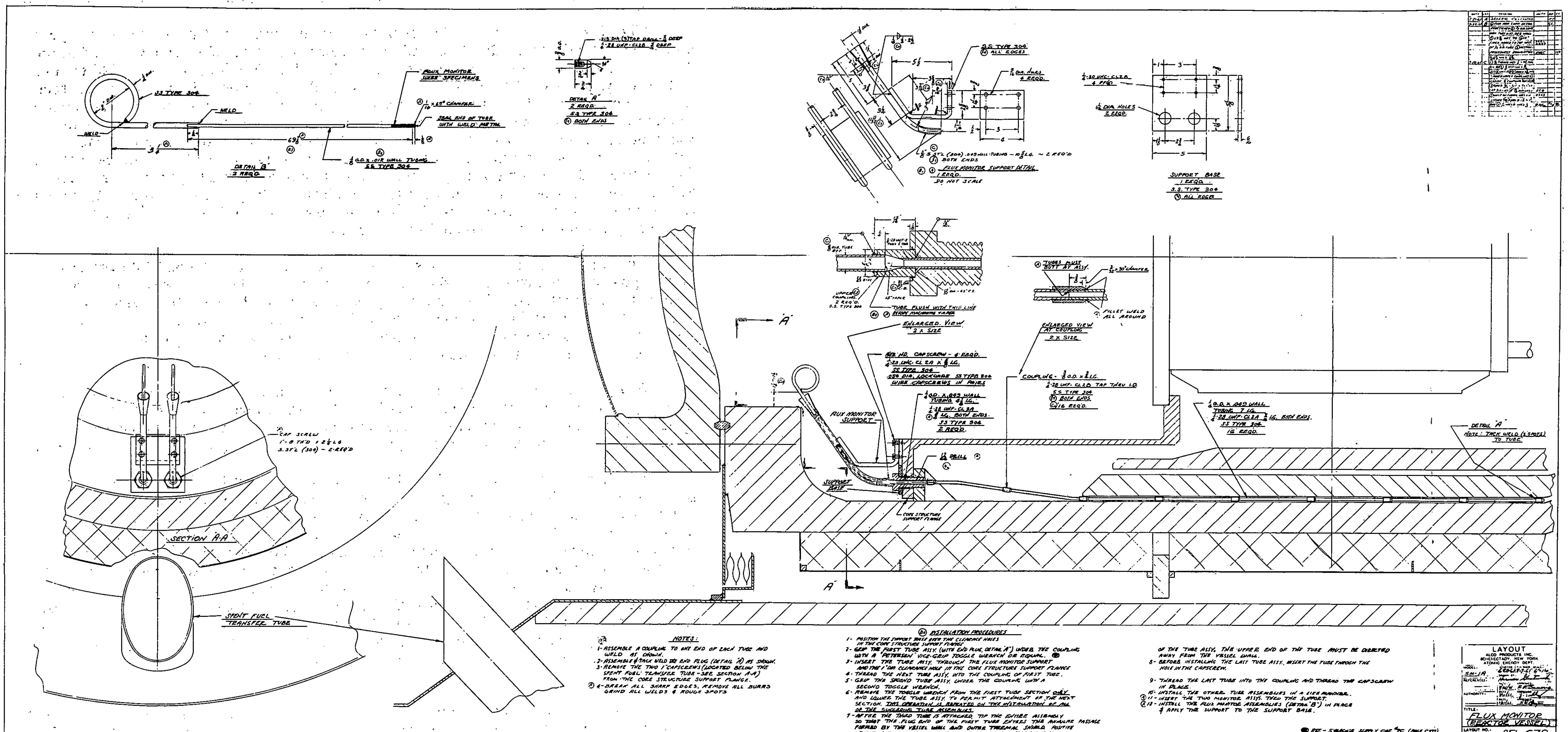
$$\text{or } e = 0.287 \approx 30\%.$$

For radiation damage studies on the various reactors, it was desirable to use the most conservative estimates of nvt. Therefore, the maximum values of nvt for the SM-1 and SM-1A, as given in the preceding section, were based upon the hot calculated values, multiplied by the ratio of the measured cold to calculated cold mockup values, and finally increased by 30%. For the PM-2A, the hot calculated values were multiplied by the ratio of the measured cold to calculated cold SM-1 mockup values. No correction for probable error was made because the PM-2A differs in some significant respects from the SM-1 and SM-1A, e.g. equivalent core radius, operating temperature, and pressure vessel diameter.

3.3 FISCAL 1962 PROGRAM

During fiscal 1962, additional measurements and analyses will be performed to further reduce the possible sources of error described in the preceding section, and increase the confidence in the predicted integrated fast flux distribution for the various reactors. A summary of the program for FY-62 is described in the following paragraphs.

Information in regard to integrated fast flux magnitudes and distributions in the SM-1A reactor will be obtained by the flux monitors installed at the inside of the pressure vessel wall, prior to reactor startup. Two flux monitors were installed in the SM-1A reactor vessel in order to measure the nvt axially along the vessel wall. The monitors were placed side by side and against the inside wall


DETAIL 'B'
2 REQ'D.
DETAIL 'A'
2 REQ'D.
SUPPORT BASE
1 REQ'D.
S.S. TYPE 304
FLUX MONITOR SUPPORT DETAIL
1 REQ'D.
DO NOT SCALE
ENLARGED VIEW
2 X SIZE
ENLARGED VIEW AT COUPLING
2 X SIZE
FLUX MONITOR SUPPORT
SUPPORT BASE
SECTION 'A-A'
SPENT FUEL TRANSFER TUBE
NOTES:
1- ASSEMBLE A COUPLING TO ONE END OF EACH TUBE AND WELD AS SHOWN.
2- ASSEMBLE & TACK WELD THE END PLUG (DETAIL 'A') AS SHOWN.
3- REMOVE THE TWO 1" CAPSCREWS (LOCATED BELOW THE SPENT FUEL TRANSFER TUBE - SEE SECTION A-A) FROM THE CORE STRUCTURE SUPPORT FLANGE.
4- BREAK ALL SHARP EDGES, REMOVE ALL BURRS GRIND ALL WELDS & ROUGH SPOTS
INSTALLATION PROCEDURES
1- POSITION THE SUPPORT BASE OVER THE CLEARANCE HOLES IN THE CORE STRUCTURE SUPPORT FLANGE
2- GRIP THE FIRST TUBE ASSY. (WITH END PLUG, DETAIL 'A') UNDER THE COUPLING WITH A 'PETERSEN' VICE-GRIP TOGGLE WRENCH OR EQUAL.
3- INSERT THE TUBE ASSY. THROUGH THE FLUX MONITOR SUPPORT AND THE 1" DIA. CLEARANCE HOLE IN THE CORE STRUCTURE SUPPORT FLANGE
4- THREAD THE NEXT TUBE ASSY. INTO THE COUPLING OF FIRST TUBE.
5- GRIP THE SECOND TUBE ASSY. UNDER THE COUPLING WITH A SECOND TOGGLE WRENCH.
6- REMOVE THE TOGGLE WRENCH FROM THE FIRST TUBE SECTION ONLY AND LOWER THE TUBE ASSY. TO PERMIT ATTACHMENT OF THE NEXT SECTION. THIS OPERATION IS REPEATED ON THE INSTALLATION OF ALL OF THE SUCCEEDING TUBE ASSEMBLIES.
7- AFTER THE THIRD TUBE IS ATTACHED, TIP THE ENTIRE ASSEMBLY SO THAT THE PLUG END OF THE FIRST TUBE ENTERS THE ANNULAR PASSAGE FORMED BY THE VESSEL WALL AND OUTER THERMAL SHIELD. POSITIVE INDICATION FOR THE ABOVE IS DETERMINED BY THE ENTRANCE ANGLE OF THE TUBE ASSY. THE UPPER END OF THE TUBE MUST BE DIRECTED AWAY FROM THE VESSEL WALL.
8- BEFORE INSTALLING THE LAST TUBE ASSY., INSERT THE TUBE THROUGH THE HOLE IN THE CAPSCREW.
9- THREAD THE LAST TUBE INTO THE COUPLING AND THREAD THE CAPSCREW IN PLACE.
10- INSTALL THE OTHER TUBE ASSEMBLIES IN A LIKE MANNER.
11- INSERT THE TWO MONITOR ASSYS. THRU THE SUPPORT.
12- INSTALL THE FLUX MONITOR ASSEMBLIES (DETAIL 'B') IN PLACE & APPLY THE SUPPORT TO THE SUPPORT BASE.
REF. - SYRACUSE SUPPLY CAT. #7C (PAGE C377)
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.
MODEL - SM-1A
TITLE - FLUX MONITOR (REACTOR VESSEL)
LAYOUT NO. - AEL-679

of the reactor vessel, parallel to the axis of the vessel, as shown in Dwg. AEL-679.

The flux monitor assembly consists of 70-in. lengths of pure iron wire, pure nickel wire, and cobalt aluminum alloy wire intermittently shielded with cadmium. These detectors will provide nvt data above 4.1 Mev, 5.0 Mev and at 0.025 ev and 0.4 ev respectively. The detectors were sealed in stainless steel tubes to protect them from contact with the primary water. The iron and nickel wires were placed in one tube and the remaining detector in another. A handle in the form of a loop was welded to the top of each monitor to facilitate remote insertion and removal from the permanently installed stainless steel tubes which locate and support them. The permanent monitor housings have a flared, funnel-shaped opening to further aid remote insertion of replacement flux monitors. They were permanently fastened to the core support structure as shown in the drawing and do not interfere with reactor operation or core reloading in any way.

On startup, the flux monitors are to be irradiated for at least 30 days at a steady power level after which they will be removed, and the wires cut into short segments and analyzed. The data will present a complete axial flux map of the vessel wall from the lower vessel girth weld to the upper portion of the coolant nozzles. It is of interest that this monitor design is the first known attempt to map the flux at the wall of a power reactor in this way.

Analysis of the SM-1 flux monitor capsules installed as part of the FY-61 program will be completed during FY-62. The results will be correlated with mockup measurements and subsize Izod test data to provide further confirmation of vessel wall flux values and the degree of radiation damage in the vessel material.

Further experimental work will be done on mockups in FY-62 to reduce the probable error inherent in the present nvt estimates ans to provide data for correlation with the results from flux monitor tests. The new mockups of the SM-1, SM-1A and PM-2A to be erected at the Alco Critical Facility will simulate the operating conditions of temperature, pressure, and control rod bank position of the respective plants rather than the cold, clean conditions of the present mockups. Neutron activation experiments utilizing threshold and resonance data detector foils, to define both the high energy and 1/E energy range, will be performed to establish the neutron intensity as a function of both energy and location in the core and vessel. Threshold detectors being considered include sulfur, aluminum, and U-238; resonance detectors being considered include gold, indium, manganese, sodium, and rhodium.

Analytical work in the PWR Program for FY-62 includes the calculation of the integrated fast neutron flux on the PM-2A vessel. The calculations will utilize the one-dimensional, multigroup P1MG, IBM-7090 code, or P3MG, Philco 2000 code, together with the two-dimensional, few group PDQ (r, z), IBM-7090 code. Calculations for the cold and hot, clean PM-2A reactor will be compared to measurements performed under the FY-62 experimental program. Correlation with experimental data will provide an accurate estimate of the nvt distribution for the PM-2A operating plant and the probable error involved.

4.0 REACTOR VESSEL MATERIALS

The SM-1, SM-1A and PM-2A reactor vessel materials were ordered to Alco and ASTM specifications and all were found to conform with the mechanical and chemical requirements set forth in these specifications. High quality inspection techniques and standards were utilized by the steel producer and by the fabricator of each vessel. The non-destructive inspection methods used are discussed in subsequent sections of this report.

Initial (unirradiated) NDT values and tensile property data were also obtained on each of the reactor vessel steels. The results of these tests are shown under Section 4.0 entitled "Data Summary." In the case of the SM-1, A-212B material trepanned from the reactor vessel at the nozzle opening was used for determining the nil ductility transition temperature (NDT). Conventional Charpy "V" inpact specimens were machined from this trepanned material and tested at various temperatures. The results were analyzed jointly by Alco and NRL. It was concluded that the 20 ft-lb energy level was a conservative value for the NDT fix of this material. The nil ductility temperature corresponding to this energy level on the Charpy "V" transition curve is +40°F.

Mill test tensile property data for the SM-1 pressure vessel material was taken from the mill test report supplied by Lukens Steel Company.

For the PM-2A pressure vessel, NDT and tensile specimens of A-350 LF-3 material were taken from a duplicate vessel forging made from the same ingot, rough machined and heat treated by the same procedures used in manufacturing the PM-2A vessel. For the SM-1A pressure vessel, NDT and tensile specimens of A-350 LF-1 modified were taken from a test slab forged and heat treated from the same heat of steel and at the same time the SM-1A vessel forgings were produced. NRL drop weight specimens were used for NDT determinations on both of these materials at the U.S. Naval Research Laboratory. The nil ductility temperatures as reported by NRL from these tests were -40°F for the SM-1A and -80°F for the PM-2A vessel material. Low initial NDT vaues of this order of magnitude are highly desirable, particularly since irradiation of structural materials at high flux levels can result in a decided increase in NDT, thus restricting its use for long term reactor vessel applications.

Impact energy-temperature transition curves for the above materials were also obtained by NRL using standard Charpy "V" specimens. It was determined that for the unirradiated vessel materials, the NDT corresponded to impact energies of 32 and 35 ft-lbs for the PM-2A and SM-1A materials, respectively. These impact energy values were used for subsequent determination of increases in NDT for irradiated Charpy "V" impact specimens of pressure vessel materials.

The information contained in this report on the SM-1, SM-1A, and PM-2A reactor vessels was compiled from data reported by the following:

(1) Lukens Steel Company - Supplier of the SM-1 vessel material - A-212 Grade "B" roll clad with stainless steel type A-240 Grade S.

(2) Alco Products, Inc. - Latrobe Plant - Supplier of the SM-1A and PM-2A vessel materials. (A-350 LF-1 and A-350 LF-3 respectively).

(3) Alco Products, Inc. - Dunkirk Plant - Fabricator of the SM-1, SM-1A and PM-2A reactor vessels.

(4) U. S. Naval Research Laboratory - Source of initial NDT data on PM-2A and SM-1A reactor vessel materials.

4.1 DATA SUMMARY - SM-1 REACTOR VESSEL

1. Type of Material - A-212 Grade "B" clad with stainless steel Type A-240 Grade S and purchased to ASTM specification A-264 Grade 3.

2. Supplier - Lukens Steel Company.

3. Mill Test Data - The SM-1 vessel material was checked to ASTM A-212 Grade B for the plate, to ASTM A-240 Grade "S" for cladding chemical composition, and to ASTM A-264 Grade 3 for the shear strength of the plate cladding interface.

   (a) Chemical Composition -

Heat No.	Material	C	Mn	P	S	Si	Cr	Ni
21866-12	A-212B	0.26	0.85	0.016	0.024	0.020	-	-
21866-12	A-240 Grade S	0.027	1.04	0.023	0.010	0.45	18.57	9.40
24045-15	A-212B	0.25	0.78	0.018	0.029	0.17	-	-
24045-15	A-240 Grade S	0.044	0.90	0.024	0.016	0.61	18.54	9.32

   (b) Tensile Properties -

Heat No.	Material	Tensile Strength	Yield Point	% Elong. in 2 in.	Shear Strength
21866-12	A-212B	77,000 psi	46,500 psi	33%	-
21866-12	A-264 Grade 3	-	-	-	29,500 psi
24045-15	A-212B	75,500 psi	44,000 psi	33%	-
24045-15	A-264 Grade 3	-	-	-	32,500 "

   (c) Other Tests -

   Intergranular Corrosion Tests performed on the stainless cladding according to ASTM specification A-240-54 Section 12 (1955 Edition) were satisfactory.

Bend Tests on the stainless steel clad plate performed in accordance with ASTM specification A-264-44T, Section 7, Item 2 and Section 8, Item 2 (1955 Edition) were satisfactory.

4. Elevated Temperature Mechanical Property Data*

(a) A-240 Grade S - Type 304 Cladding

Temp.	Tensile Strength	0.2% Yield Strength	Poisson's Ratio	Modulus of Elasticity
R.T.	79,000 psi	32,000 psi	0.23	29×10^6
200°F	72,000 psi	25,000 psi	0.26	28×10^6
400°F	66,000 psi	20,000 psi	0.29	27×10^6
600°F	62,500 psi	17,500 psi	0.31	25.5×10^6
800°F	59,500 psi	16,000 psi	0.32	24×10^6
1000°F	54,000 psi	15,000 psi	0.31	23×10^6
1200°F	44,000 psi	12,000 psi	0.29	22×10^6

(b) A-212 Grade "B" (Base Metal)

R.T.	76,000 psi	45,000 psi		29.5×10^6
200°F	78,000 psi	44,000 psi		28.5×10^6
400°F	77,500 psi	40,000 psi		27×10^6
600°F	73,000 psi	36,000 psi		26×10^6
800°F	60,000 psi	30,000 psi		24×10^6
1000°F	42,000 psi	24,000 psi		23×10^6
1200°F	24,000 psi	14,000 psi		21.5×10^6

* This data collected from many sources for design engineering information. It was not obtained from tests on actual SM-1 reactor vessel material.

5. Heat Treatment -

(a) A-212 Grade B clad with A-240 Grade S stainless steel was heated to 1950^{O}F and air-cooled. (By Lukens Steel Company.)

(b) Clad plate was heated to 1750^{O}F at Alco Products, Inc. (Dunkirk Plant) for roll forming. Working was completed above the upper critical temperature of ~ 1530^{O}F. (A_{C3})

(c) Rolled clad plate was welded, then stress relieved at 1150^{O}F. (Alco Products, Inc. - Dunkirk Plant)

6. Initial NDT -

+40^{O}F Determined jointly by Alco Products, Inc. and NRL on A-212B material trepanned from the reactor vessel at the nozzle opening. (See Fig. 6. Impact Energy Transition Curve.)

7. Method of Cladding the Stainless Steel to the Base Metal -

The stainless steel clad, A-240 Grade S, was roll bonded to the A-212 Grade B carbon steel base metal by Lukens Steel Company.

8. Welding of Vessel -

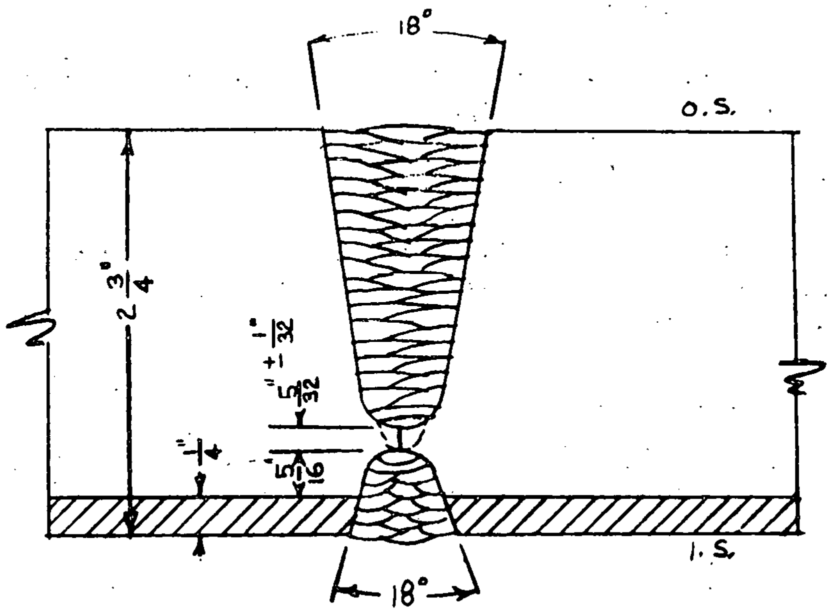
The shell barrel longitudinal and girth seam welds were made in accordance with Alco Dwg. B-150552.

9. Non-Destructive Inspection Tests -

(a) No non-destructive tests were performed on the stainless steel clad carbon steel plate prior to vessel fabrication.

(b) All reactor vessel weld seams were inspected radiographically to the 2% sensitivity level.

ALCO PRODUCTS, INCORPORATED DUNKIRK, N.Y.		
ORDER NO. 43060 REF. DWG. NO. D-145912	WELDING SPECIFICATION UPPER SHELL LONG & GIRTH SEAMS	DWG. NO. B-150552



BEAD NO. (Approx.)	PROCESS	ELECTRODE SIZE	ELECTRODE TYPE	CURRENT VOLTS	CURRENT AMPERES	FLUX	SPEED IPM
I.S.							
1	Metal Arc	1/8"	E-7016	22-24	110-130	—	—
2-4	Metal Arc	3/16"	E-7020	24-26	175-240	—	—
	Back Gouge or Chip to Sound Weld Metal						
O.S.							
1-46	Sub. Arc	1/8"	Oxweld 40A	34-38	450-600	#80	16-20
I.S.							
5-7	Metal Arc	5/32"	Type 310	22-24	110-135	—	—
8-16	Metal Arc	3/16"	Type 308L	22-24	125-160	—	—

4.2 DATA SUMMARY - SM-1A REACTOR VESSEL

1. Type of Material - A-350 Grade LF-1 Forging modified by the addition of up to 2% Nickel.

2. Supplier - Alco Products, Inc. (Latrobe Plant).

3. Heat No. - 76840 (acid open hearth, fully killed).

4. Forging Serial No. - 373150.

5. Mill Test Data -(On shell barrel).

   (a) Chemical Composition -

C	Mn	P	S	Si	Ni	Cr	Mo	V	AL
0.13	0.78	0.025	0.030	0.27	1.65	0.05	0.04	0.04	0.067

   (b) Tensile Properties -

Tensile Strength	0.2% Offset Yield Strength	% Elong.	% R. A.	Charpy Keyhole
72,750 psi	50,250 psi	33.5	63.6	22.5 ft-lbs @ -50°F

6. Data on SM-1A Test Slab -

   (a) Chemical Composition - A-350 LF-1 Modified (Heat #76840)

Element	Analysis
Carbon	0.15
Manganese	0.79
Phosphorus	0.027
Sulfur	0.033
Silicon	0.25
Nickel	1.71
Chromium	0.05
Molybdenum	0.04
Copper	0.088
Vanadium	0.040
Aluminum	0.075
Cobalt	0.003
Tin	0.007
Oxygen	19 ppm
Hydrogen	0.7 ppm
Nitrogen	50 ppm

(b) Tensile Properties -

Test Temp.	Tensile Strength	0.2% Offset Yield Strength	% Elong.	% R. A.
*R. T.	71,750 psi	57,500 psi	33%	63.3%
200°F	71,000 psi	54,000 psi	-	-
400°F	70,000 psi	45,000 psi	-	-
*600°F	64,500 psi	38,000 psi	30%	72.5%
800°F	50,000 psi	31,000 psi	-	-
*1100°F	33,250 psi	19,250 psi	41.5%	89.9%

* Test data obtained on SM-1A test slab. All others are interpolated values.

(c) ASTM End Quench Test for Hardenability of Steel (ASTM A-255-48T)

Data tabulated below are average readings from hardness traverses taken 180° apart on 1.0 in. diameter Jominy test specimens: -

Hardness Rockwell "C"	Distance from Quenched End of Specimen in Sixteenths of Inch
40	1
35	2
24	4
19	6
17	8
15	10
13	12
11	14
10	16
9	18
8	20
7	22
7	24
7	26
6	28
6	30
6	32

(d) ASTM Grain Size (McQuaid - Ehn Test)

Grain size No. 8 as determined from Grain Size Chart of ASTM specification E-19-46 (In 1955 edition of ASTM standards).

(e) Thermal Critical Temperatures (Determined according to ASTM specification E-80-49T in 1958 edition of ASTM standards).

$A_{c1} = 1325^{o}F$

$A_{c3} = 1550^{o}F$

$A_{r1} = 1200^{o}F$

$A_{r3} = 1480^{o}F$

7. Heat Treatment -

(a) Quenched from 1600°F) at Alco - Latrobe
(b) Tempered at 1250°F)

(c) Welded, then stress relieved at 1150°F (Alco - Dunkirk)

NOTE: All test specimens were also stress relieved at 1150°F.

8. Initial NDT -

(a) -40°F Determined by NRL on unirradiated samples of the A350 LF-1 test slab forging. (See Fig. 6. Impact Energy Transition Curve.)

9. Method of Cladding the Base Metal with Stainless Steel

(a) Type 304 stainless steel overlay was used for cladding the A-350 LF-1 base metal.

(b) The overlay was applied to the shell barrel as per Alco welding specification shown on Dwg. B-41201-1-200.

(c) The upper and lower girth seam welds on this vessel were made in accordance with the welding specifications shown on Alco Dwg. B-41201-1-203A and B-41201-1-204A respectively.

(d) Seam welds on shell barrel were made in accordance with Alco Dwg. B-41201-1-203.

(e) All welds on inlet and outlet nozzles were made in accordance with Alco Dwg. B-41201-1-201.

10. Non-Destructive Inspection Tests -

A. The SM-1A Reactor Vessel Forgings (A-350 Grade LF-1 modified) were ultrasonically inspected at Latrobe as follows: -

(1) Test Specifications - MIL-STD-271 "Ships" dated 7/27/56.

(2) Equipment - Sperry Ultrasonic Reflectoscope - contact method.

(3) Coverage - 100 percent with adequate crystal overlap.

(4) Test Frequencies - 1.0 and 2.25 megacycles. Longitudinal and 45 degree shear wave.

(5) Crystal Size - 1-1/8 in. diameter longitudinal beam and 1 in. shear wave.

(6) Scanning Speed - 6 inches per second maximum.

(7) Material Finish - 250 rms or better.

(8) Couplant - Oil "SAE No. 20".

(9) Test Standards -

(a) Longitudinal Beam - First end reflection was set to at least 3/4 the screen height.

(b) Shear Wave - Where applicable, a 3 percent test notch was cut axially along the outer surface of the forgings.

(c) Rejection Levels -

Longitudinal Test - Any indication which exceeds 5 percent of the first end reflection at a test frequency of 1.0 megacycles.

Shear Wave Test - Any indication which exceeds in magnitude the indication from the 3 percent test notch.

(10) Test Procedure -

(a) Shell Forging - Drawing C-41201-1-104-C.
Ring Forgings - All

General - All ring forgings and the shell barrel were examined using both the longitudinal and shear wave methods.

Longitudinal Beam - The outer surface and both ends of each ring forging were examined using the test standard for the longitudinal method.

Shear Wave - A test notch was cut axially along the outer surface of each ring forging. The depth of the notch did not exceed 3 percent of the wall thickness. The sound beam was directed into the forging from the outer surface in a circumferential direction.

(b) Disc Forgings - Forgings of this type were examined using the longitudinal beam crystal. Testing was done from the outer surface and both faces of each disc type forging. The test standard for the longitudinal method was used in all cases.

(11) Test Results

(a) Ring Forgings - All.

No defects were found approaching or exceeding the test standards as set forth in MIL-STD-271 "Ships".

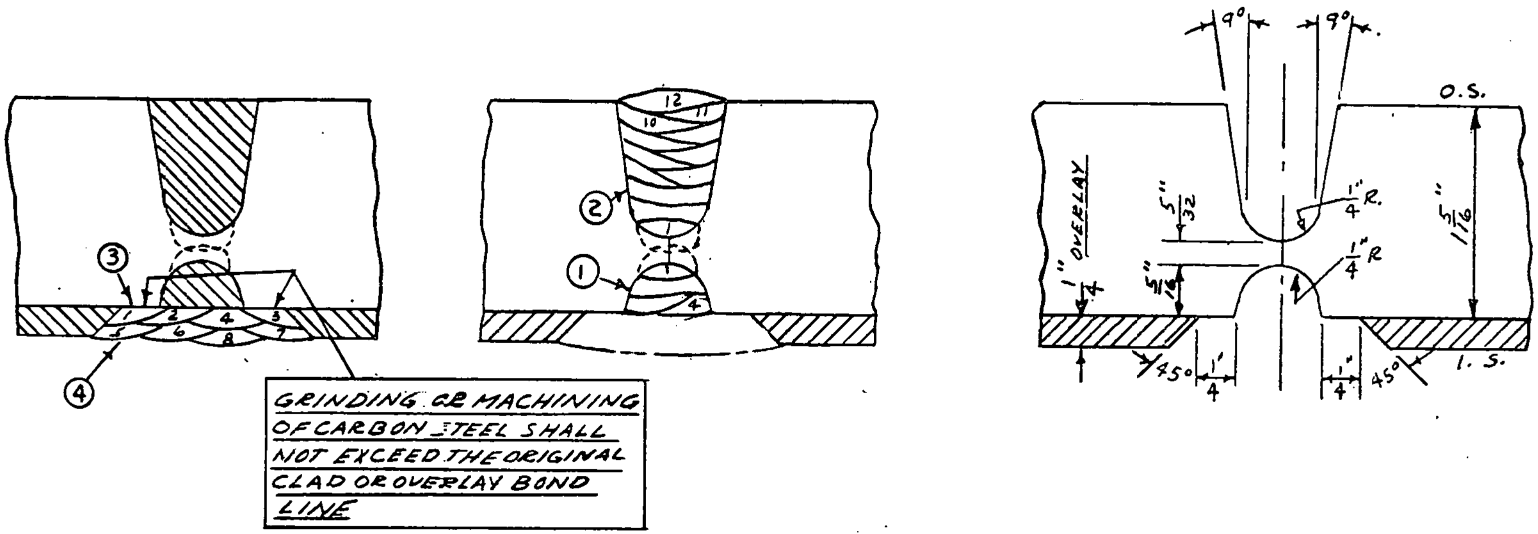
No ultrasonic indications were observed exceeding 5 percent of the first end reflection on the "Longitudinal Test", or 3 percent of the material thickness on the "Shear Wave" test.

(b) Disc Forgings - No indications exceeding 5 percent of the first end reflection were observed on disc type forgings.

B. The SM-1A Reactor Vessel Weld Seams and Overlay - Inspected at Dunkirk.

(1) Overlay - Liquid penetrant inspection to MIL-STD 271-A
- Ultrasonic inspection (100% loss of back reflection)

(2) Weld Seams - Radiographic inspection (2% Sensitivity)



INSPECTION: 1) RADIOGRAPH COMPLETED WELD EXCLUDING OVERLAY

2) LIQUID PENETRANT EXAMINATION OF THE COMPLETED OVERLAY & AREA 1" EACH SIDE OF THE COMPLETED WELD

BASE MATL. FGD. CARB. STL. SA-350 GR."LF-1")

(ADDED NICKEL UP TO 2% MAX.)

PREHEAT 150°F. PRIOR TO WELDING EXCLUDING SEQ. ④

SEQUENCE	BEAD NO	PROCESS *	FILLER MATERIAL		CURRENT		FLUX OR GAS SHIELD	TRAVEL SPEED I.P.M.
			SIZE	TYPE	VOLTS	AMPS		
①	1 / 2-4	M.A.	3/16"	E-7016 / E-7020	22-24 / 30-36	250-300 / 175-250		
②	1-12	S.A.	1/8"	OXWELD 40A	32-34	450-475	UNIONMELT GR80 OR 78	18-25 I.P.M.
③	1-4	M.A.	3/16"	E-309-15	23-25	125-165		
④	5-8	M.A	3/16"	E-308ELC-15	23-25	125-165		

*M.A. = METAL ARC

*S.A. = SUBMERGED ARC

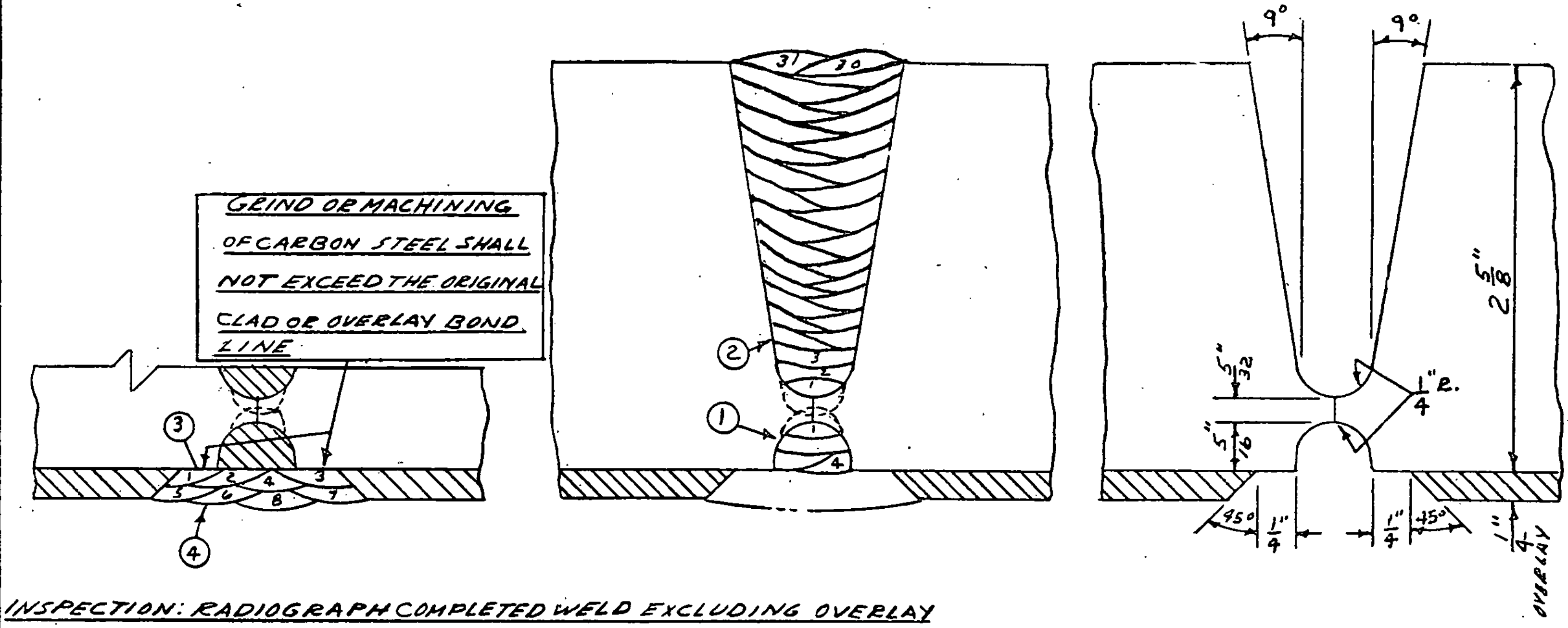
ISSUE	DATE	BY	CKD.		CERT.
A	1/19/57	DFB		INCREASED OVERLAY FROM 3/16" TO 1/4"	
B	1/23/55	DFB		ADDED PREHEAT	

REVISIONS

DUNKIRK, N.Y. **ALCO PRODUCTS, INC.** PLANT No. 3

WELDING SPECIFICATION FOR LOWER SHELL GIRTH SEAMS

DWG. NO. REF.	MADE	CKD.	CERTIFIED	DRAWING NO.			
					ORDER NO.	ITEM	IDENT. NO.
F-41201-1-2	DFB			B	41201	1	204A
SPEC. NO.	SCALE	DATE					
41201-W-1	FULL						



INSPECTION: RADIOGRAPH COMPLETED WELD EXCLUDING OVERLAY
LIQUID PENETRANT EXAMINATION OF THE COMPLETED OVERLAY & AREA 1" EACH SIDE OF THE COMPLETED WELD

BASE MATERIAL: FGD. CARBON STL. (SA-350 GR. "LF-1") ADDED NICKEL UP TO 2% MAX.

△ PREHEAT 150°F PRIOR TO WELDING EXCLUDING SEQ. ④

SEQUENCE	BEAD NO	PROCESS	FILLER MATERIAL SIZE	FILLER MATERIAL TYPE	CURRENT VOLTS	CURRENT AMPS	FLUX OR GAS SHIELD	TRAVEL SPEED I.P.M.
①	1 2-4	M.A.	3/16"	E-7016 E-7020	22-24 30-36	250-300 175-250		
②	1-31 APPROX	S.A.	1/8"	OXWELD 40 A	32-34	450-475	UNIONMELT GR80 OR EQ.	18-25 I.P.M.
③	1-5	M.A.	3/16"	E-309-15	23-25	125-165		
④	6-10	M.A.	3/16"	E-308ELC-15	23-25	125-165		

* M.A. = METAL ARC
* S.A. = SUBMERGED ARC

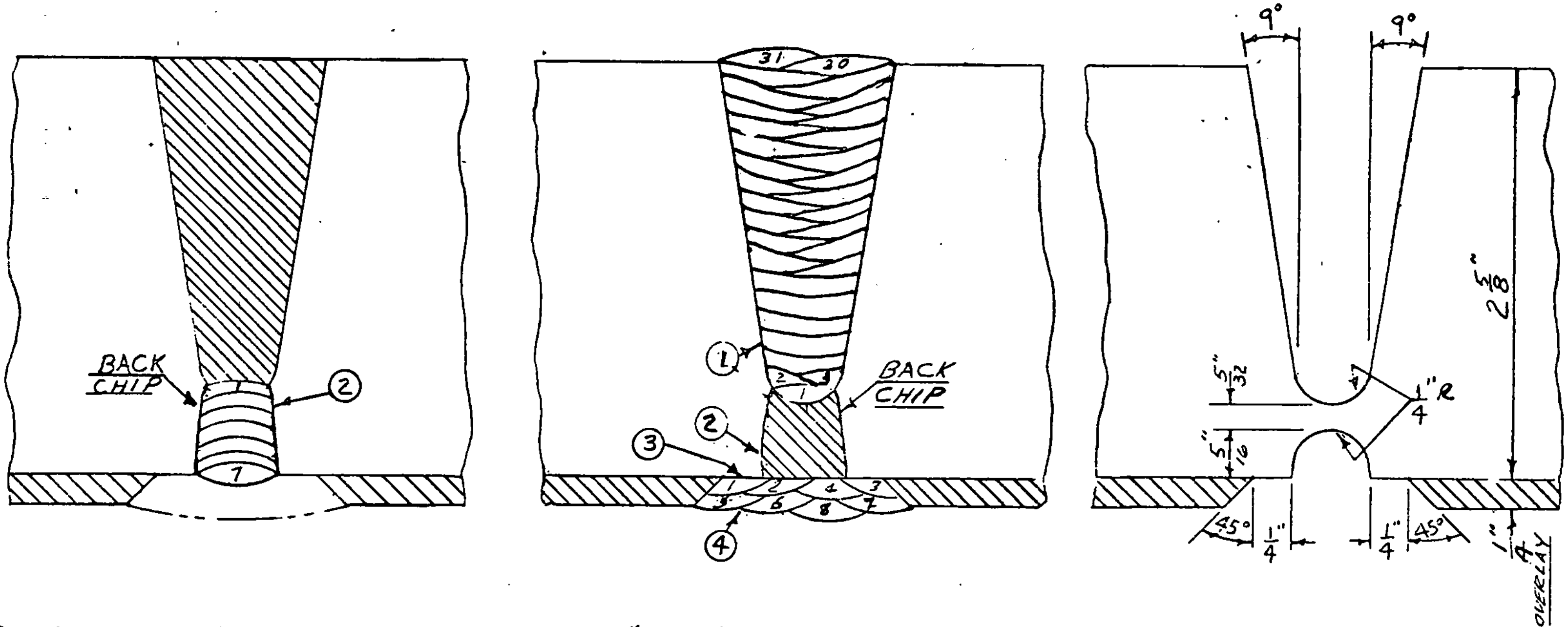
ISSUE	DATE	BY	CKD.		CERT.
A	1/19/57	D B		INCREASED OVERLAY FROM 3/16" TO 1/4"	
B △	1/23	DPB		ADDED PREHEAT	

REVISIONS

DUNKIRK, N.Y. **ALCO PRODUCTS, INC.** PLANT No. 3

WELD SPECIFICATION FOR UPPER SHELL GIRTH SEAMS

DWG. NO. REF.	MADE	CKD.	CERTIFIED	DRAWING NO.			
F-41201-1-2	DFB				ORDER NO.	ITEM	IDENT. NO.
SPEC. NO.	SCALE	DATE		B	41201	1	203 A
41201-W-1	FULL						



BASE MATLS. FGD. CARB STL. (SA-350 GR. "LF-1") ADDED NICKEL UP TO 2% MAX

INSPECTION: 1) RADIOGRAPH COMPLETED WELD EXCLUDING OVERLAY
2.) LIQUID PENETRANT EXAMINATION OF THE COMPLETED OVERLAY & AREA 1" EACH SIDE OF THE COMPLETED WELD

PREHEAT 150°F PRIOR TO WELDING EXCLUDING SEQ. 4

SEQUENCE	BEAD NO.	PROCESS *	FILLER MATERIAL		CURRENT	
			SIZE	TYPE	VOLTS	AMPS
1	1-3 APPROX (O.S.)	M.A.	3/16"	E-7016	22-24	250-300
1	4-31 APPROX.	M.A.	1/4"	E-7020	30-36	250-350
2	1-7	M.A	3/16"	E-7020	30-36	175-250
3	1-4	M.A.	5/32"	E-309-15	24-26	120-160
4	5-8	M.A.	3/16"	E-308ELC	25-27	150-190

* METAL ARC

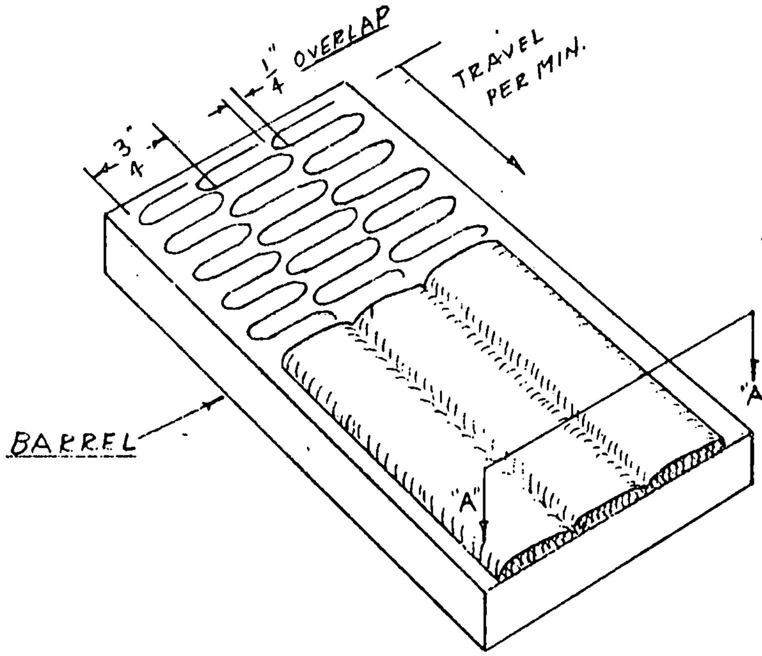
ISSUE	DATE	BY	CKD.		CERT.
A	1/18	D.F.B		INCREASED OVERLAY FROM 3/16" TO 1/4"	
B	1/23	D.F.B		ADDED PREHEAT	

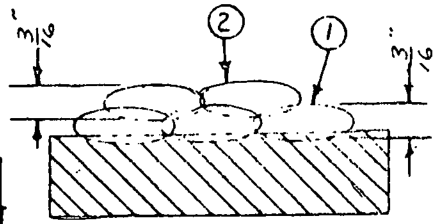
REVISIONS

DUNKIRK, N.Y. **ALCO PRODUCTS, INC.** PLANT No. 3

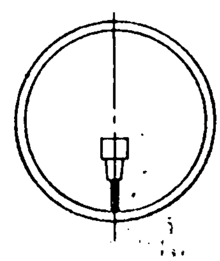
WELDING SPECIFICATION FOR UPPER SHELL GIRTH SEAMS

DWG. NO. REF.	MADE	CKD.	CERTIFIED	DRAWING NO.			
F-41201-1-2	D.F.B.				ORDER NO.	ITEM	IDENT. NO.
SPEC. NO.	SCALE	DATE		B	41201	1	203
4-1201-W-1	FULL						





"A"-"A"



POSITION OF ELECTRODE

INSPECTION: LIQUID PENETRANT OR FLOURESCENT POWDER EXAMINATION & ULTRASONIC TEST THE COMPLETED WELD

BASE MATLS: FGD. CARB. STL. (SA-350 GR."LF-1" ADDED NICKEL UP TO 2% MAX.

PREHEAT	SEQUENCE	BEAD NO.	PROCESS *	FILLER MATERIAL		CURRENT		FLUX OR GAS SHIELD	OSCILLATION	TRAVEL
				SIZE	TYPE	VOLTS	AMPS		PER MIN.	I.P.M.
150°F	1	1	S.A.	1/8"	ER309L	38-40	320-340	ARCOSITE S-4	44/MIN.	6 1/2"
70°F	2	2	S.A.	1/8"	ER-308L	38-40	320-340	ARCOSITE S-4	44/MIN	6 1/2"

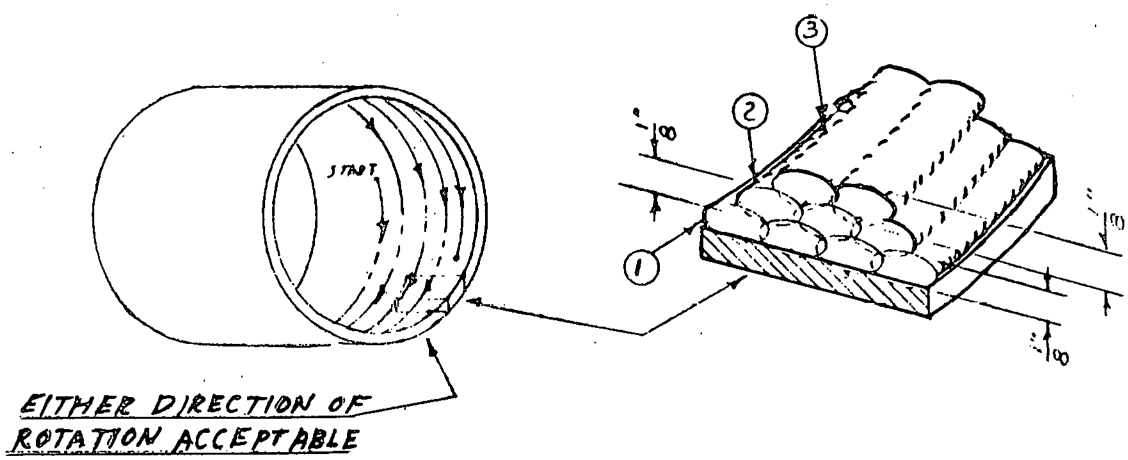
* SUBMERGED ARC

ISSUE	DATE	BY	CKD.	REVISIONS	CERT.
A	1/23/59	DFB	RR	ADDED PREHEAT	RR

DUNKIRK, N.Y. ALCO PRODUCTS, INC. PLANT No. 3

WELDING SPECIFICATION FOR STAINLESS STEEL OVERLAYS

DWG. NO. REF.	MADE	CKD.	CERTIFIED	DRAWING NO.			
					ORDER NO.	ITEM	IDENT. NO.
H-41201-1-11	DFB			B	41201	1	200
SPEC. NO.	SCALE	DATE					
41201-W-1	NONE						



INSPECTION: LIQUID PENETRANT OR FLOURESCENT POWDER EXAMINATION & ULTRASONIC TEST THE COMPLETED WELD. IF ULTRASONIC TEST IS IMPRACTICLE, THEN THE PENETRANT TEST WILL SUFFICE

BASE MAT'LS: FGD. CARB. STL. (SA-350 GR. "LF-1") ADDED NICKEL UP TO 2% MAX.

SEQUENCE	BEAD NO	PROCESS	FILLER MATERIAL SIZE	FILLER MATERIAL TYPE	CURRENT VOLTS	CURRENT AMPS
1	1	M.A.*	5/32"	E-309-15	21-23	135-200
2	2	M.A.*	3/16"	E-308L	22-24	160-240
3	3	M.A.	3/16"	E308L	22-24	160-240
DIA. SMALLER THAN 3" USE 5/32" DIA. ELECTRODE						

Δ PREHEAT TO 150°F PRIOR TO WELDING FOR SEQ (1)

* METAL ARC

A	1/18	DFB		ADDED SEQ (3)	
B Δ	1/23	DFB		ADDED PREHEAT	RR

REVISIONS

DUNKIRK, N.Y. ALCO PRODUCTS, INC. PLANT No. 3

WELDING SPECIFICATION FOR STAINLESS STEEL OVERLAYS

H-41201-1-11	DFB				
41201-W-1	NONE		41201	1	201

4.3 DATA SUMMARY - PM-2A REACTOR VESSEL

1. Type of Material - ASTM A-350 Grade LF-3 (Forging)

2. Supplier - Alco Products, Inc. (Latrobe Plant)

3. Heat No. 76809 (Acid open hearth, fully killed.)

4. Forging Serial Nos. - 378839 and 378840

5. Mill Test Data (On shell barrel)

   (a) Chemical Composition -

Serial No.	C	Mn	P	S	Si	Ni	Cr	Mo	V
378839	0.15	0.53	0.029	0.029	0.29	3.33	0.04	0.04	0.04
378840	0.15	"	"	"	"	"	"	"	"

   (b) Tensile Properties -

Serial No.	Tensile Strength	Yield Point	% Elong.	% R. A.	Charpy Keyhole
378839	80,500 psi	62,250 psi	28.5	66.8	~ 41 ft-lbs @ -50°F
378840	80,500 psi	57,000 psi	30.5	67.4	~ 39 ft-lbs @ -50°F

6. Test Data on PM-2A Duplicate Vessel Forging. (Same heat and ingot as operating vessel forging.)

   (a) Chemical Composition - (A-350 Grade LF-3)

Element	Analysis
Carbon	0.145
Manganese	0.52
Phosphorus	0.031
Sulfur	0.032
Silicon	0.25
Nickel	3.28
Chromium	0.04
Molybdenum	0.05
Copper	0.134
Vanadium	0.04

Aluminum	0.057
Cobalt	0.010
Tin	0.014
Oxygen	30 ppm
Hydrogen	0.8 ppm
Nitrogen	44 ppm

(b) Elevated Temperature Tensile Properties -

Test Temp.	Tensile Strength	0.2% Offset Yield Strength	% Elong.	% R. A.
*R. T.	83,750 psi	62,500 psi	30.0	69.9
200°F	82,000 psi	57,000 psi	-	-
400°F	82,000 psi	54,000 psi	-	-
*600°F	77,500 psi	47,750 psi	33.5	72.3
800°F	62,000 psi	40,000 psi	-	-
*1100°F	30,000 psi	-	41.5	89.4

* Test data obtained on PM-2A duplicate forging. All others are extrapolated values.

(c) ASTM End Quench Test for Hardenability of Steel. (ASTM A-255-48T)

The data tabulated below are average readings from hardness traverses taken 180°F apart on 1.0 in. diameter Jominy test specimens: -

Hardness Rockwell "C"	Distance from Quenched End of Specimen in Sixteenths of Inch
40	1
38	2
29	4
23	6
22	8
19	10
18	12
16	14
16	16
15	18
14	20
13	22
13	24
13	26
12	28
12	30
12	32

(d) ASTM Grain Sizes (McQuaid - Ehn Test)

Grain size No. 8 as determined from grain size chart of ASTM specification E-19-46 in 1955 edition of ASTM standards.

(e) Thermal Critical Temperatures (Determined according to ASTM specification E-80-49T in 1958 edition of ASTM Standards)

$A_{c_1} = 1280^{o}F$

$A_{c_3} = 1475^{o}F$

$A_{r_1} = 1140^{o}F$

$A_{r_3} = 1315^{o}F$

7. Heat Treatment

(a) Normalized at $1550^{o}F$)
(b) Quenched at $1500^{o}F$) At Alco - Latrobe
(c) Tempered at $1200^{o}F$)

(d) Welded, then stress relieved at 1150^{o} - Alco, - Dunkirk

NOTE: All test specimens were also stress relieved at $1150^{o}F$.

8. Initial NDT -

$-80^{o}F$ (Determined by NRL on unirradiated samples of the A-350 Grade LF-3 duplicate vessel forging. (See Fig. 6. Impact energy transition curve.)

9. Method of Cladding the Base Metal with Stainless Steel -

(a) Type 304 stainless steel overlay was used for cladding the A-350 LF-3 base metal.

(b) The overlay was applied to the shell barrel as per Alco welding specification shown on Dwg. B-41202-1-201.

(c) The girth seam welds on this vessel were made in accordance with the welding specifications shown on Alco Dwg. B-41202-1-203.

10. Non-Destructive Inspection Tests -

(a) PM-2A Reactor Vessel Forgings (A-350 Grade LF-3) - were ultrasonically inspected to the same testing procedures and standards as that used for the SM-1A reactor vessel forgings.

(1) Test Results -

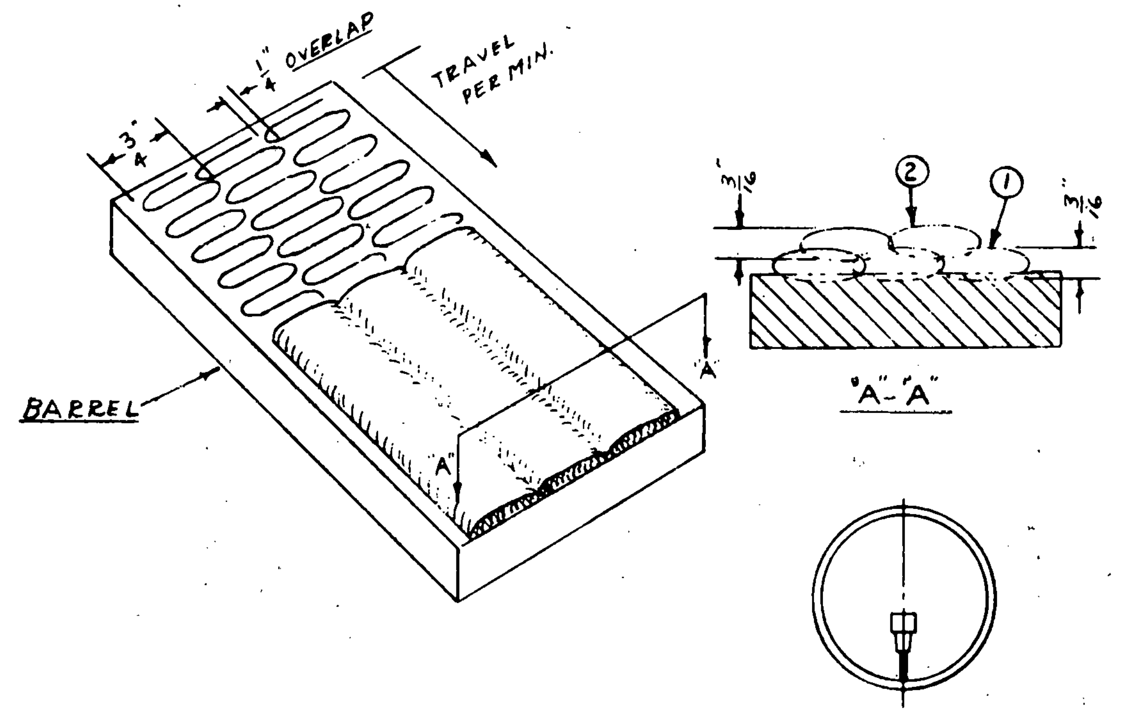
Shell Forgings - No defects were found approaching or exceeding the test standards as set forth in MIL-STD-271 "Ships". No ultrasonic indications were observed exceeding 5 percent of the first end reflection on the "Longitudinal" test, or 3 percent of the material thickness on the "Shear Wave" test.

Disc Forgings - No indications exceeding 5 percent of the first end reflection were observed on the disc type forgings.

(b) PM-2A Reactor Vessel Weld Seams and Overlay:-

(1) Overlay - Liquid Penetrant inspection to MIL SDT 271-A

(2) Weld Seams - Radiographic Inspection - (2% Sensitivity)
- Magnaflux - (Back Chip)



POSITION OF ELECTRODE

INSPECTION: PENETRANT TEST AFTER MACHINING OF OVERLAY WELD

BASE MAT'L'S SA-350 GR "LF-3"

PREHEAT	[illegible]	LAYER NO.	PROCESS *	FILLER MATERIAL SIZE	FILLER MATERIAL TYPE	CURRENT VOLTS	CURRENT AMPS	FLUX OR GAS SHIELD	OSCILLATION PER MIN.	TRAVEL I. P. M.
300°F	1	1	S.A.	1/8"	ER-309L	38-40	320-340	ARCOSITE S-4	44/MIN	6 1/2
70°F	2	1	S.A.	1/8"	ER-308L	38-40	320-340	ARCOSITE S-4	44/MIN	6 1/2

*SUBMERGED ARC.

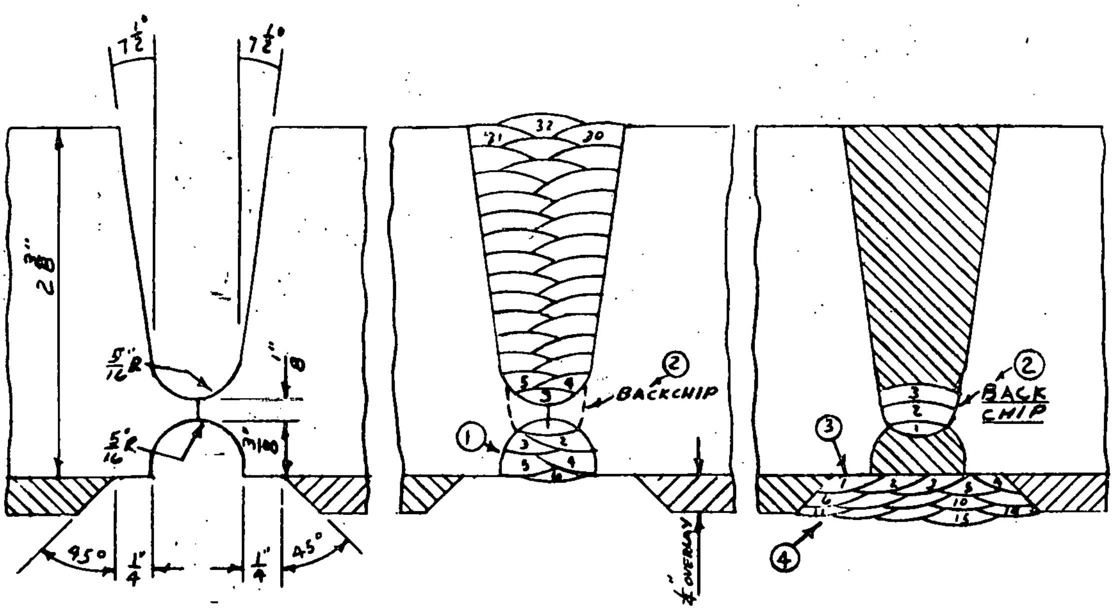
ISSUE	DATE	BY	CKD.		CERT.

REVISIONS

DUNKIRK, N.Y. **ALCO PRODUCTS, INC.** PLANT No. 3

WELDING SPECIFICATION FOR STAINLESS STEEL OVERLAYS

DWG. NO. REF.	MADE	CKD.	CERTIFIED		ORDER NO.	ITEM	IDENT. NO.
F-41202-1-2	DFB			B	41202	1	201
SPEC. NO.	SCALE	DATE					
41202-W-1	NONE						



BASE MATERIAL: FGD. CARB. STL. (SA-350 GR. "LF-3" (EXCEPT CHARPY IMPACT TESTS @ MINUS 50°F.

INSPECTION: MAGNETIC PARTICLE TEST BACK CHIP.
PENETRANT TEST COMPLETED STAINL. OVERLAY
RADIOGRAPH TEST COMPLETED JOINT

PREHEAT	FLUX	TRAVEL I.P.M.	SEQUENCE	BEAD NO.	PROCESS	FILLER MATERIAL		CURRENT	
						SIZE	TYPE	VOLTS	AMPS
300°F			①	1-6 APPROX	M.A.	5/16"	E-8016-C1 PFH 75LP	22-24	175-240
300°F			②	1-32 APPROX	M.A.	3/16"	E-8016-C1 PFH 75LP	22-24	175-240
300°F			③	1-5	MA	3/16"	E-309-15	25-27	150-190
70°F			④	6-15	M.A.	3/16"	E-308ELC-15	25-27	150-190

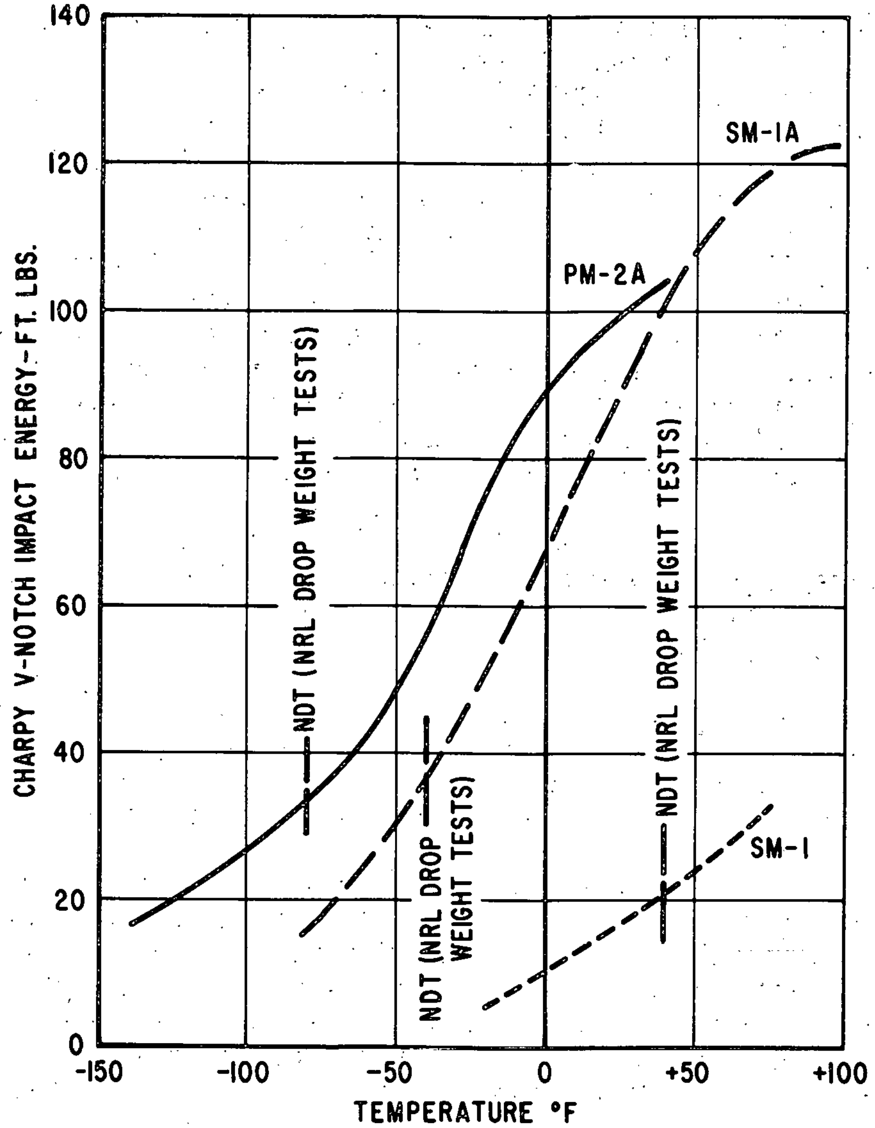
DUNKIRK, N.Y. **ALCO PRODUCTS, INC.** PLANT No. 3

WELDING SPECIFICATION FOR SHELL GIRTH SEAMS

ISSUE	DATE	BY	CKD.	REVISIONS	CERT.
A	11/[illegible]	DFB		CHANGED TO HAND WELD	

DWG. NO. REF.	MADE	CKD.	CERTIFIED	DRAWING NO. ORDER NO.	ITEM	IDENT. NO.
F-41202-1-2	D.F.B.			B 41202	1	203
SPEC. NO.	SCALE	DATE				
41202-W-1	FULL					

FIGURE 6
IMPACT ENERGY TRANSITION CURVES
FOR THE SM-1, SM-1A AND PM-2A
REACTOR VESSELS



4.4 IRRADIATED PROPERTIES

4.4.1 Fiscal 1961 Program

The increase in nil-ductility transition temperature of a material exposed to irradiation is one of the most sensitive indices to material damage. To define the degree of damage or shift in transition temperature that might be expected in the Army vessels, NRL included samples of Army materials in their existing research program on radiation damage.

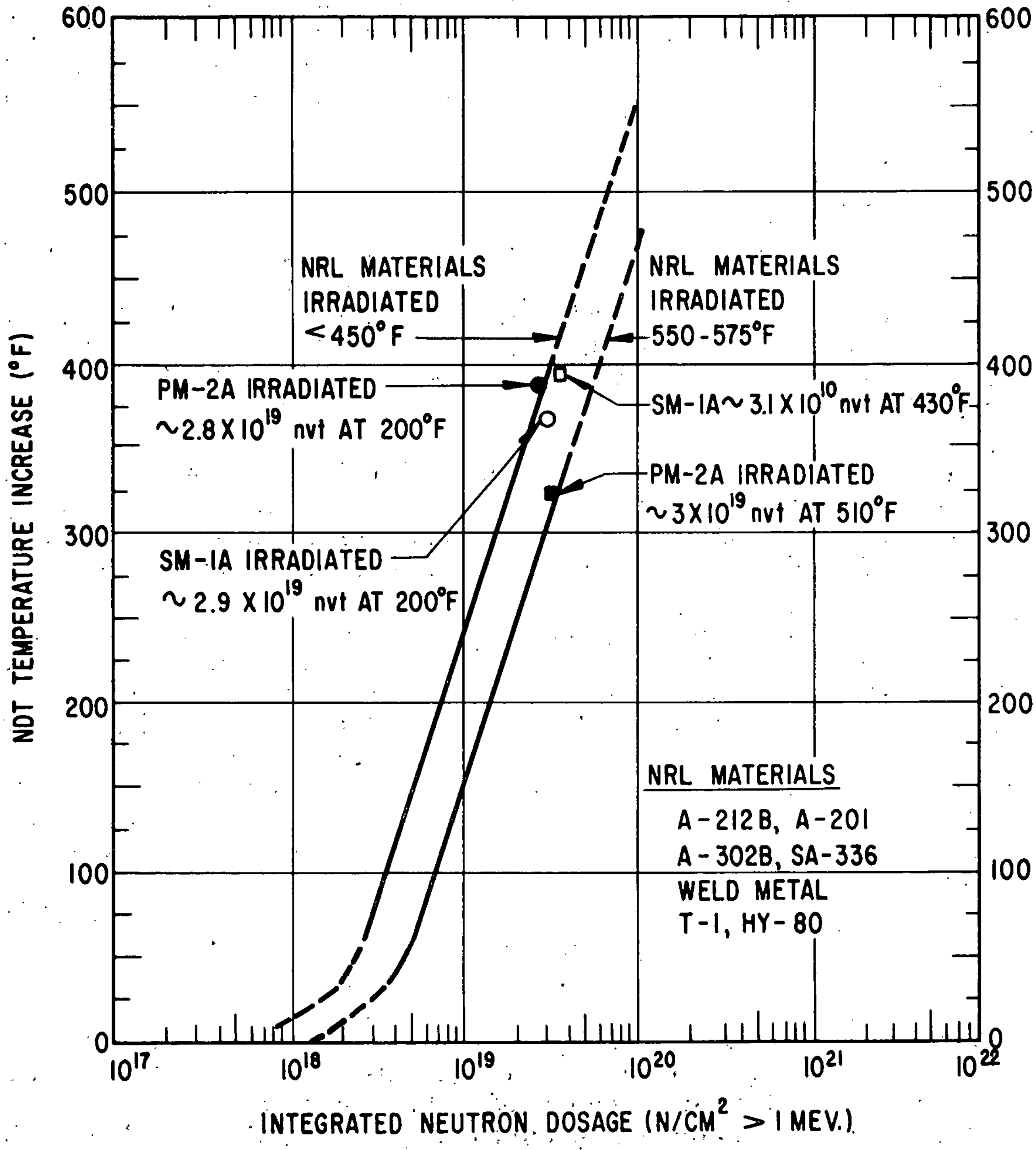
To date, metallurgical testing of these materials by NRL has consisted of NDT determinations on samples of A-212B, A-350 LF-1 modified, and A-350 LF-3 irradiated both "cold" ($\sim 200^oF$) and "hot", i. e. at reactor operating temperatures of 430^oF for the SM-1 and SM-1A vessel materials and 510^oF for the PM-2A vessel material. The results of these tests are shown in Fig. 7. This graph also shows the increase in transition temperature vs. integrated neutron flux ≥ 1 Mev for a number of NRL materials irradiated at temperatures below 450^oF, and at temperatures ranging from 550^oF-575^oF at various flux levels. The solid portion of each curve in this figure represents the pattern followed by the data points obtained on the irradiated NRL materials. The dotted lines are an extrapolation of the solid portions of these curves. The transition temperature shifts obtained for the PM-2A and SM-1A reactor vessel materials irradiated at 200^oF and at reactor operating temperatures are superimposed on the NRL curves. The increase in transition temperature for the SM-1 and SM-1A reactor vessel materials irradiated at 430^oF are shown to fall on the upper NRL curve in Fig. 7. The PM-2A vessel material,which was irradiated at 510^oF,experienced a somewhat smaller increase in transition temperature corresponding to the lower NRL curve. However, for calculation purposes, it was conservatively assumed to have the same increase as SM-1 and SM-1A materials.

Because of the excellent correlation between the NRL data and that obtained on the PM-2A and SM-1A vessel materials, the curves shown in Fig. 7 are considered to be an accurate representation of the NDT shifts which can be expected for the Army reactor vessel steels as a function of integrated neutron flux.

In addition to the testing performed by NRL on these materials, ORNL determined transition temperature shifts for sub-size Izod specimens of A-212B reactor vessel material which had been irradiated for the duration of Core I (16. 4 MWYR) in holders inserted in the upper core support ring of the SM-1. The resulting NDT shifts obtained from these specimens were converted to Charpy V - transition temperature shifts by the addition of a conversion factor previously determined by ORNL in their work on sub-size Izod-Charpy V-notch energy temperature correlations. It was also experimentally determined with an SM-1 mockup in the Alco Critical Facility using sulfur foils, that a specific surveillance specimen position received a fast flux dosage 2. 9 times as great as the pressure vessel wall. The ORNL observed transition temperature shift at this specific specimen location indicated that the total integrated flux was 4.8×10^{18} nvt. Dividing this value by 2. 9 indicated that the SM-1 pressure vessel had accrued

FIGURE 7

TRANSITION TEMPERATURE SHIFT Vs.
INTEGRATED NEUTRON FLUX (>1 MEV)



1.6 x 10^{18} nvt at the end of Core I. This value was in good agreement with a calculated value of 1.32 x 10^{18} nvt previously determined using the IBM 650 computer codes PDQ-2 and P1MG-2.

Besides NDT determinations on the as-irradiated SM-1, SM-1A and PM-2A materials, annealing studies are also being conducted on the latter two vessel steels to provide data for extending the life of these vessels. It has been shown[9] that a major fraction of the original mechanical properties can be restored after irradiation by annealing at elevated temperatures. Primary emphasis is being given the PM-2A material because of the relatively short duration of its current operating limits.

4.4.2 Fiscal 1962 Program

Other studies scheduled for the radiation damage program include irradiation of stressed specimens in the research reactors (MTR and LITR), development of a strain cycling test specimen and the development of an allowable stress test to determine the degree of conservatism inherent in operating a vessel below the NDT at a stress equal to a fraction of the yield strengh as specified in the Alco design criteria in Section 6.0.

Irradiated tensile properties of SM-1A and PM-2A vessel materials will also be determined. The increase in yield strength due to irradiation is of primary interest because it may permit relaxation of reactor vessel operating limits and can lead to extended life of the PM-2A and SM-1A vessels without annealing.

Surveillance type capsules containing specimens of Army reactor vessel materials have been placed in the SM-1 and SM-1A. Two dummy fuel elements of the type shown in Fig. 8, each loaded with twelve capsules containing specimens of A-212B, A-350 LF-1 (modified), A-350 LF-3 and five other reference materials, are now under irradiation in SM-1 Core II. Anticipated flux dosages for one core life range from 1 x 10^{20} nvt to 1 x 10^{21} nvt. This total integrated neutron flux is at or above the maximum 20 year nvt expected for any of the Army reactor vessels. Data obtained from specimens exposed to such exceptionally high neutron fluxes will aid considerably in predicting the nvt which can be expected for reactor vessels with an extended design life or higher flux intensities. One dummy element (12 capsules) will remain in the reactor for the life of the core, while the other element will be removed for testing after partial core life.

A total of eight capsules have also been prepared for irradiation in positions above the SM-1A core. Fig. 9 shows the hanger assembly and capsules along with two of the four units which have been placed on the core support ring. These capsules contain Charpy-V and tensile specimens of A-212B (SM-1), A-350 Grade LF-1 modified (SM-1A), A-350 Grade LF-3 (PM-2A) and two other reference materials.

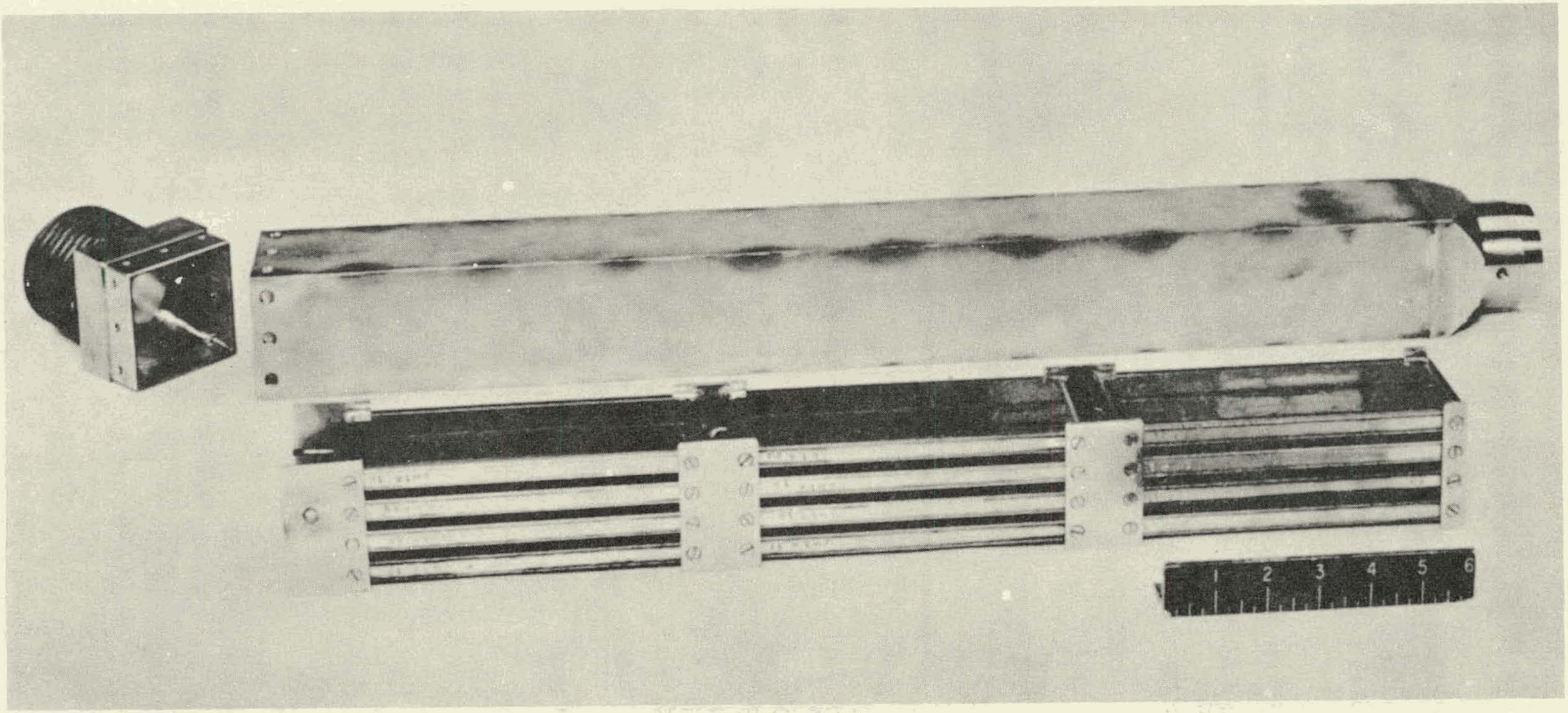


FIG. 8 CHARPY-V AND TENSILE SPECIMAN ASSEMBLY FOR SM-1 REACTOR

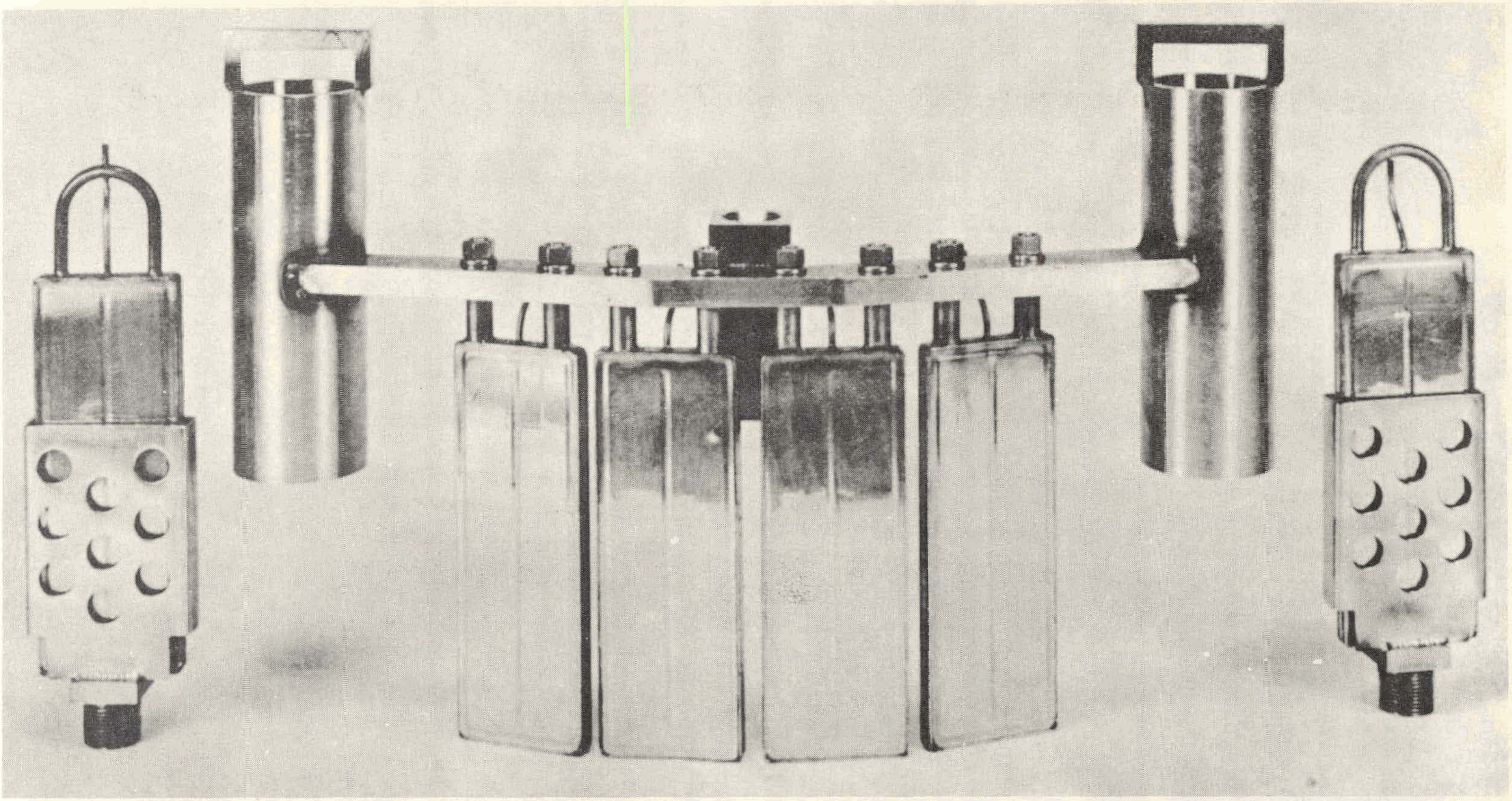


FIG. 9 EXPERIMENTAL SPECIMAN ASSEMBLIES FOR PLACEMENT IN PERIPHERAL POSITIONS OF SM-1A REACTOR

5.0 VESSEL STRESSES

The concept that was followed to avoid brittle fracture of the Army vessels was to control the primary system pressure and core power level at temperatures below the full power operating level, in order to limit stresses to the values prescribed by the Alco design criteria for irradiated vessels. These criteria, presented in Section 6.0, state that the maximum principal stress must be limited to 18 percent of the yield strength of the vessel material at temperatures below the FTE. This rule is the foundation of the Army Nuclear Power Program approach to the control of the radiation damage problem.

As a result of the rapid decline in nvt (and transition temperature increase) with axial distance from the beltline, a vessel has not one but a whole range of FTE temperatures with the highest value at the beltline. In constructing the operating boundaries of a given vessel, the maximum permissible FTE was determined as described in detail in Section 7.0, from a consideration of the operating characteristics of the primary system. This temperature marked the level below which stresses were to be controlled and, through the NRL damage curve and Alco flux estimates, established the period during which the operating limits were valid. The pressure and power level were then reduced to satisfy the design criteria at the beltline. These conditions could be maintained until the temperature had dropped to the FTE of the point nearest the beltline experiencing higher stresses. At that temperaure, a further reduction in pressure was made to bring the stress at that location within the criteria. This process was repeated for other points of stress concentration until a pressure boundary had been described for temperatures from the operating range to the lowest temperature at which the primary system might be pressurized.

The results of the detailed stress analyses of the highly irradiated areas of the three reactor vessels are summarized in the following paragraphs. In all cases, transient thermal stresses and nozzle stresses due to pressure and piping loads were calculated using the methods and stress computation factors of the Navy design code, "Tentative Structural Design Basis For Reactor Pressure Vessels and Directly Associated Components", (Revised December 1, 1958)[10]. Discontinuity stresses and statically indeterminate members were analyzed using the methods of classical stress analysis described in standard texts on the subject such as Timoshenko's "Theory of Plates and Shells"[11], and Seeley and Smith's "Advanced Mechanics of Materials"[12].

Stresses were calculated at the beltline both for full power conditions and for the reduced pressure and power levels required to satisfy the design criteria. Stresses at nozzles and supports were calculated at reduced pressure and power if they caused a reduction in the operating limits, and at full power if they did not.

5.1 SM-1 VESSEL, (Dwg. AEL-47)

The beltline of the vessel was the only region of the vessel receiving an integrated neutron flux high enough to cause significant radiation damage. It was also the area of lowest stress, since it contained no nozzle penetrations, structural discontinuities or other types of stress concentrations. At temperatures where stresses were to be controlled to prevent brittle fracture, i.e. during startup and shutdown, the stresses which had to be considered were the membrane stresses due to pressure and the thermal stresses due to temperature transients. Because of the low core power level during such operations, gamma heating stresses were low.

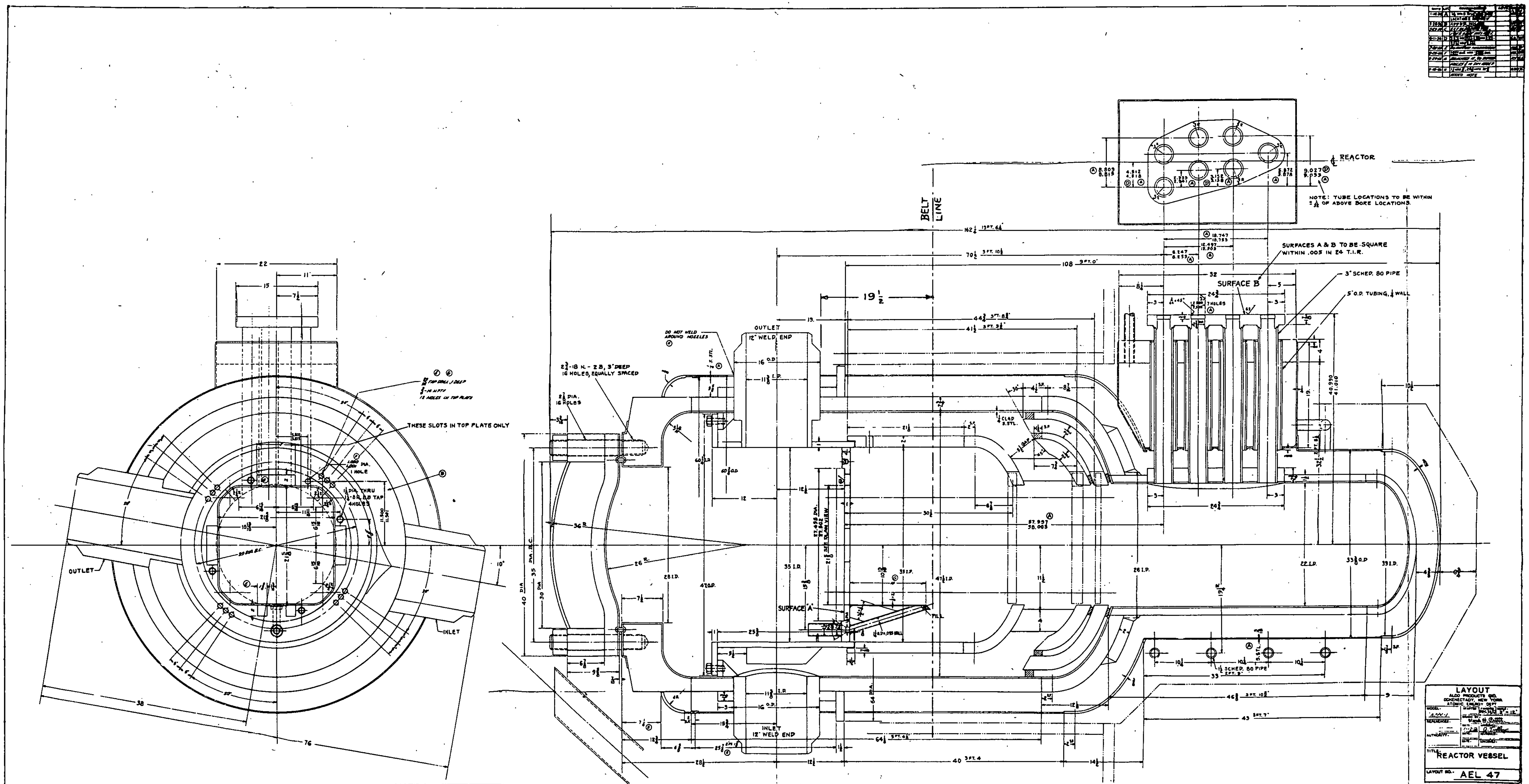
Stresses in the SM-1 vessel at the beltline are summarized in Table 6 for temperatures above and below 305°F, the maximum allowable midplane FTE. Above 305°, stresses are those due to full power operating conditions and allowable stresses are taken from the Navy Code. Below 305°, stresses are limited to 18 percent of the yield strength as specified by the design criteria in Section 6.0 of this report. In order to meet this limit, reactor operating pressure was restricted to 600 psig and core power was limited to 1 Mw. at the lower temperatures. Transient thermal stresses were calculated for a conservative 50°/hr transient even though in actual operation all transient operations are limited to 30°/hr to avoid excessive stresses in the Y-valve in the primary loop.

TABLE 6
SM-1 VESSEL STRESSES

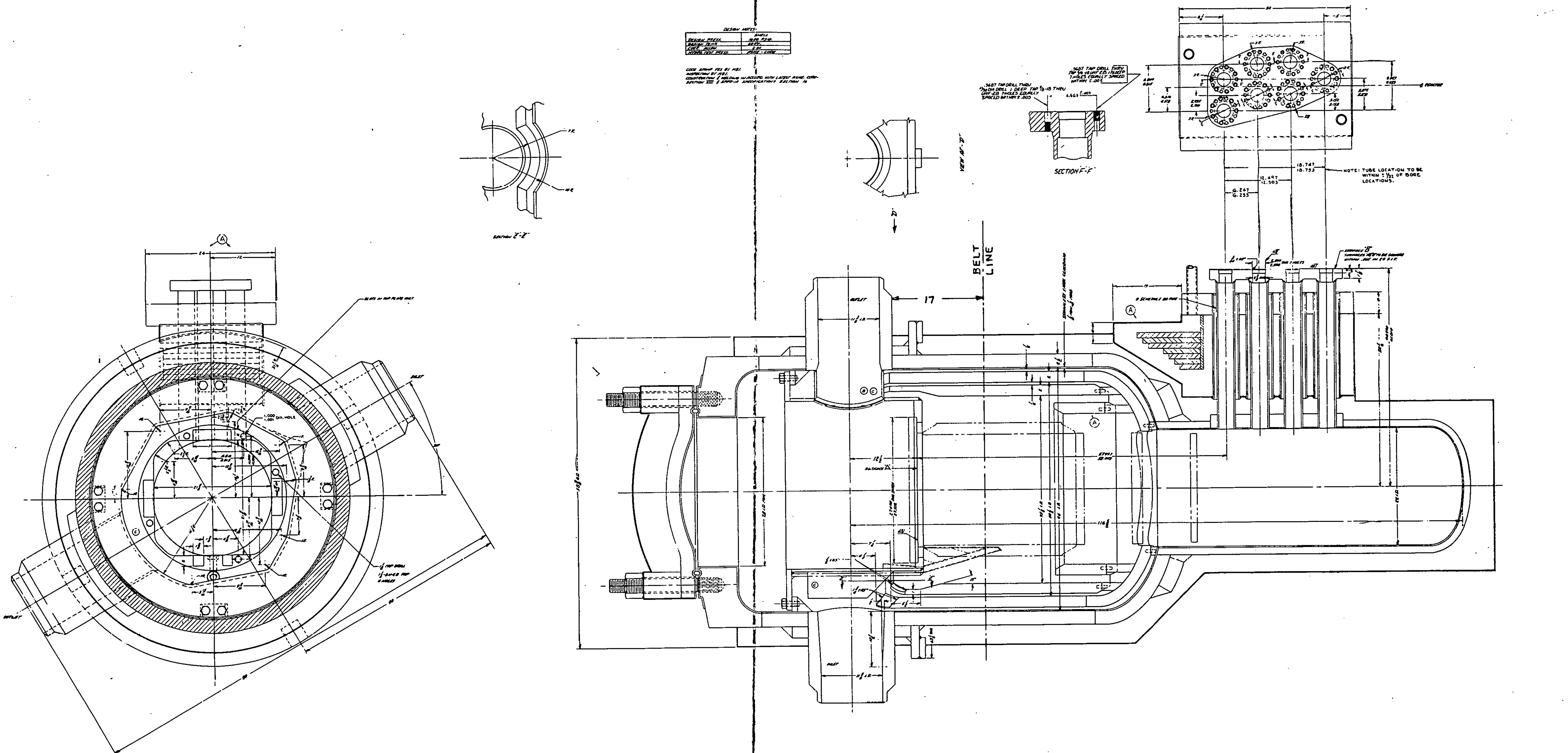
Temperature °F	Pressure psig	Power	Pressure Stress, psi	Thermal Stress, psi γ heating	Thermal Stress, psi 50°/hr	Total Stress psi	Allowable Stress, psi
≥ 305	1200	10 MW.	12,100	2200	530	14,830	30,500
< 305	600	< 1 MW.	6,050	< 220	530	6,800	7,500

5.2 SM-1A VESSEL, (Figure 10)

The regions of the SM-1A vessel which were of interest were the beltline, support ring and coolant nozzles. However, only the beltline region was important in establishing the operating boundaries because the FTE of the nearest high-stress area, the support ring, was below the temperature at which the system must be pressurized above the nominal level of 50 psig. As in the SM-1, the beltline was the area of lowest stress due to the absence of nozzles, penetrations, discontinuities or stress

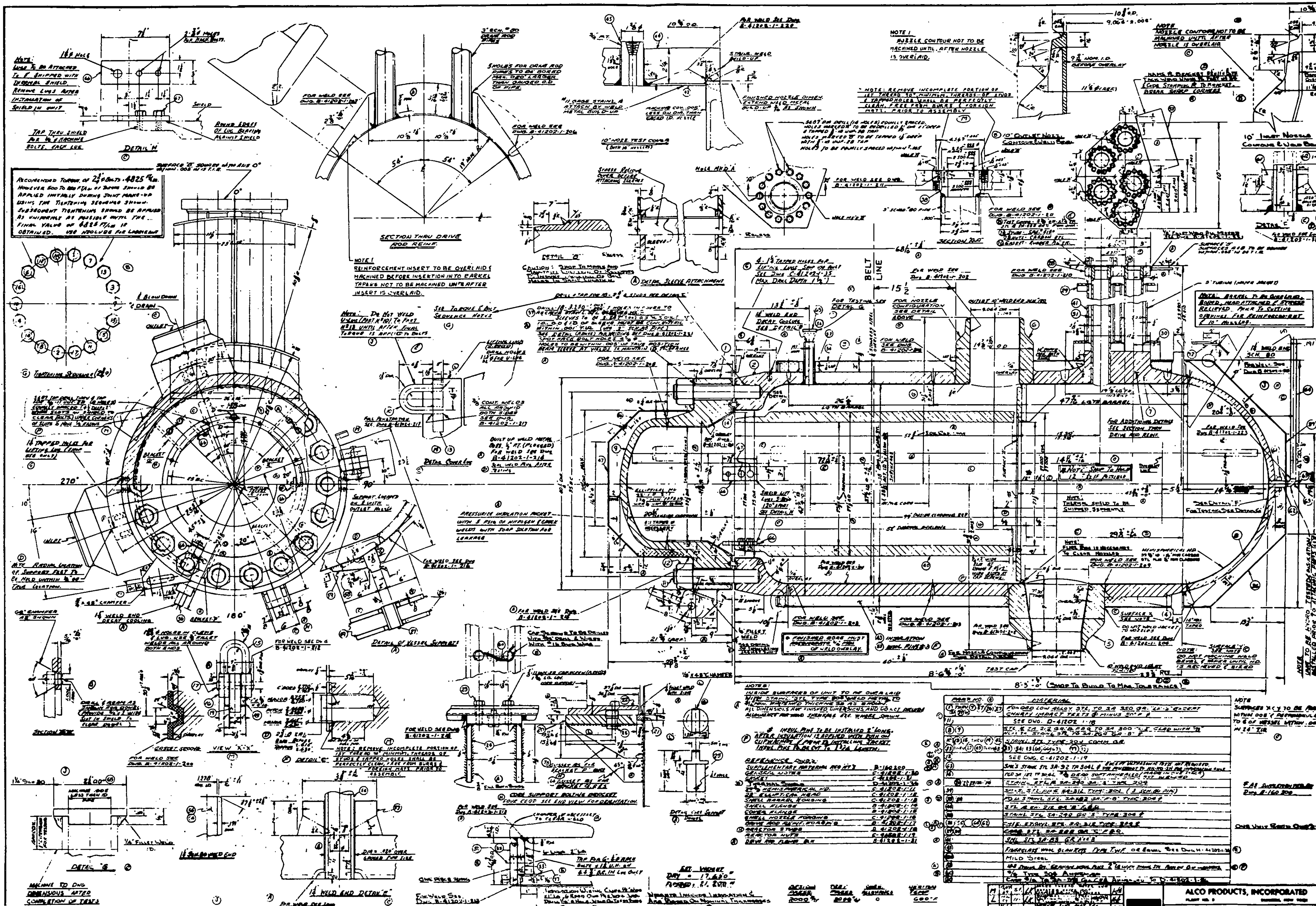


SM-I REACTOR VESSEL



SM-IA REACTOR VESSEL

Figure 10 - Reactor Vessel



PM-2A REACTOR VESSEL

concentrations. Stresses were primarily those of a simple pressurized cylinder with some contribution from transient and steady state temperature distributions. Reduction of the power to 2 Mw and the pressure to 808 psig brought them within the 18 percent of yield strength limit. The nearest region of higher stress was the point at which the vessel support ring was attached, two in. below the bottom of the main coolant nozzles. The neutron flux at this point however, was only 16 percent of the midplane value. After 62.6 MWYR of operation, the period during which the current operating limits are valid, the integrated flux will be 3.36×10^{18} neut/cm^2, so that the FTE will have risen to only 105°F. Thus, no pressure reduction was required in the operating limits to control stresses at the support ring or at the nozzles, which have an even lower FTE.

Stresses in the vessel barrel and nozzles were computed using the Navy Code as a guide. Stresses at the support ring caused by the temperature difference between the ring and the vessel required a somewhat different approach. The ring has a slower response to primary fluid thermal transients than the vessel itself and tends to restrain the free thermal expansion of the vessel. In calculating the stress due to this restraint,the difference in the free-body thermal expansions of vessel and ring was computed first. Then the interaction forces required to eliminate the discontinuity between the two members were determined taking into account the relative stiffnesses of ring and cylinder. The interaction forces were in the form of four equal and uniformly spaced radial loads on the vessel wall. Stresses due to these loads were then computed using the Navy Code analysis for radially applied local loads on a cylinder. The local bending stresses at the support ring caused by the reactions required to support the dead weight of the vessel were also computed using the local load analysis.

The stresses summarized in Table 7 were calculated at full power conditions and, at the beltline, for the reduced pressure (808 psig) required by the operating limits at temperatures below 360°. Stresses at the nozzles and support ring were not involved in setting the current operating limits, so they were listed only for full power operating conditions. Transient thermal stresses were calculated using a conservative 50°/hr transient instead of the 20°/hr transient specified for normal operation.

5.3 PM-2A VESSEL (Dwg. F-41202-1-2)

The compactness of the portable PM-2A vessel leads to generally higher radiation levels than those experienced by the SM-1 and SM-1A vessels. It also is the reason for the three pressure steps in the maximum pressure limit for this vessel, Fig. 13, compared to the single step for the other vessels. Actually a fourth step in pressure was determined for the outlet nozzle weld. However, the FTE temperatures and required pressure levels of the outlet nozzle and decay cooling nozzle were so close together that they were combined conservatively to

TABLE 7 SM-1A VESSEL STRESSES

BELTLINE: Circumferential stress, inner surface

Temperature ^{o}F	Pressure psig	Power MW	Pressure Stress, psi	Thermal Stress, psi γ heating	Thermal Stress, psi $50^{o}/hr$	Total Stress psi	Allowable Stress, psi
$\geq$ 360	1200	20	11,200	1900	630	13,732	30,500
< 360	808	2	7,480	190	630	8,300	8,300

VESSEL WALL AT SUPPORT RING AND NOZZLES

Pressure = 1200 psig, Temperature = $430^{o}F$

Circumferential stress, inner surface

Location	External Load Stress, psi	Pressure Stress, psi	Thermal Stress, psi Steady-State	Thermal Stress, psi $50^{o}/hr$	Total Stress psi	Allowable Stress, psi
Support ring	-1160	11,200	205	1990	12,235	30,500
Nozzle	- 136	25,600	*	*	*	30,500

* These values are to be calculated during FY'62 stress analysis program.

form one step at the higher of the two FTE's and the lower of the two required pressures. Thus, the step at 235° and 385 psig represents the FTE of the decay cooling nozzle and the pressure required to satisfy the design criteria at the outlet nozzle.

Another result of the small vessel size is that areas of stress concentration are close together and closer to the beltline, with the result that differences in integrated flux and FTE temperature are smaller than was the case in SM-1 and SM-1A. In the stress summary, Table 8, the stresses were calculated at each point of interest for both full power conditions and the reduced pressure and power required to meet the design criteria for operation below the FTE of that point. The methods of analysis for the beltline, decay cooling nozzle, and outlet nozzle again were those of the Navy Code. The vessel flange stresses were calculated from a discontinuity analysis of the entire vessel flange, cover and stud region including the effects of initial stud tightening, internal pressure and temperature distributions. Thermal stresses were conservatively calculated for a transient of 50^{o}/hr rather than the 30^{o}/hr limit specified during heatup and cooldown, including emergency (decay heat removal) cooling.

TABLE 8, PM-2A VESSEL STRESSES

BELTLINE - Circumferential Stress, Inner Surface

Temperature °F	Pressure psig	Power Mw	Pressure Stress, psi	Thermal Stress, psi: γ heating	Thermal Stress, psi: 50°/hr	Total Stress psi	Allowable Stress psi
≥ 380°	1750	10	15,200	6,300	580	22,080	30,500
< 380°	975	1	8,480	630	580	9,690	9,700

VESSEL WALL STRESS AT DECAY COOLING NOZZLE, OUTLET NOZZLE AND FLANGE WELD

Circumferential Stress, Inner Surface

Location	Temp. °F	Press. psig	Power Mw	Pressure Stress, psi	External Load Stress, psi	Thermal Stress, psi: γ heating	Thermal Stress, psi: 50°/hr	Total Stress, psi	Allowable Stress, psi
Decay Cooling Nozzle	235	440	1	9,520	0	0	580	10,100	10,100
Outlet Nozzle	210	385	0	7,500	2200	0	580	10,300	10,300
Flange Weld	140	200	0	2,880	-6730	0	0*	-3,850	10,400

* The 50°/hr cooling rate will not be experienced at this temperature.

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6.0 DESIGN CRITERIA FOR IRRADIATED VESSELS

The design criteria formulated specifically to prevent brittle fracture of irradiated pressure vessls are contained in APAE No. 98, which forms an attachment to this section of the report. The vessel operating boundaries shown in Figs. 11, 12 and 13 and discussed in Section 7.0 are the result of applying these criteria to the stress analyses and metallurgical data on the SM-1, SM-1A and PM-2A vessels. It should be noted that none of the operating limits are based upon the higher irradiated yield strength of the vessel steel. Although the mid-sections of the SM-1A and PM-2A vessels satisfy the requirements of the design criteria for basing allowable stress on the irradiated yield strength, it was not used because the necessary metallurgical data on Army vessel steels are not yet available. These data will be obtained during the 1962 Army radiation damage program and will be incorporated in revised operating limits before the time limits on the current curves expire.

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7.0 SURVEILLANCE PROVISIONS AND OPERATING LIMITS

7.1 VESSEL MONITORING PROVISIONS

Two major steps have been taken to monitor radiation damage. The first was to make flux measurements within the operating vessels as well as on mockups of the core and vessel of each plant, as discussed in Section 3.0 of this report. Two flux monitoring capsules, installed in the SM-1 vessel in peripheral positions above the core, will provide flux measurements applicable to the subsize Izod specimens which were installed in similar positions during reactor construction. These measurements will be related to vessel wall flux as well, through correlation with core mockup measurements. Installation of replaceable flux monitors in the SM-1A vessel to measure the intensity, spectrum and axial distribution of flux at the vessel wall was a significant technical achievement. It will provide an experimental check on the accuracy of axial flux distribution calculations as well as a complete flux map of the irradiated area of the vessel.

The second step, discussed in Section 4.0, was to obtain metallurgical data from surveillance specimens irradiated in the reactors at a flux somewhat higher than that received by the vessel wall. A total of eight surveillance specimen capsules have been prepared by NRL for installation in the SM-1A vessel at peripheral positions above the core. They contain Charpy V and tensile specimens of the three Army reactor materials in addition to two NRL reference materials. Similar packages will be placed in the SM-1 to supplement data that will be obtained from the specimens currently being irradiated in two dummy fuel elements in the SM-1 core. Such tests, conducted in the environment of the reactor to be monitored, greatly reduce the uncertainties with regard to the effects of spectrum, irradiation rate and temperature that might exist in data obtained in research reactors.

7.2 VESSEL OPERATING LIMITS

The limitations which have been imposed on the allowable system pressure and core power during transient temperature conditions are shown in Figs. 11, 12 and 13 for the SM-1, SM-1A and PM-2A vessels respectively. The operating limits for each vessel were established as required by the increase in NDT caused by fast neutron irradiation.

The operating boundaries are presented as a pressure band varying with temperature. Although the stress calculations were based on metal temperature, the operating curves are plotted as a function of water temperature. No conversion was required between the two, as calculations indicated a maximum difference between metal and water temperatures of only 3.5^{o}, well within the error of recording instruments at the plants. The upper limit of pressure is obtained by limiting vessel stresses to the design criteria, Section 6.0, for the prevention of brittle fracture. The lower limit is the higher of (1) the saturation curve shifted 20^{o}

FIGURE 11

SM-1 OPERATING LIMITS FOR 92.5 MWYR

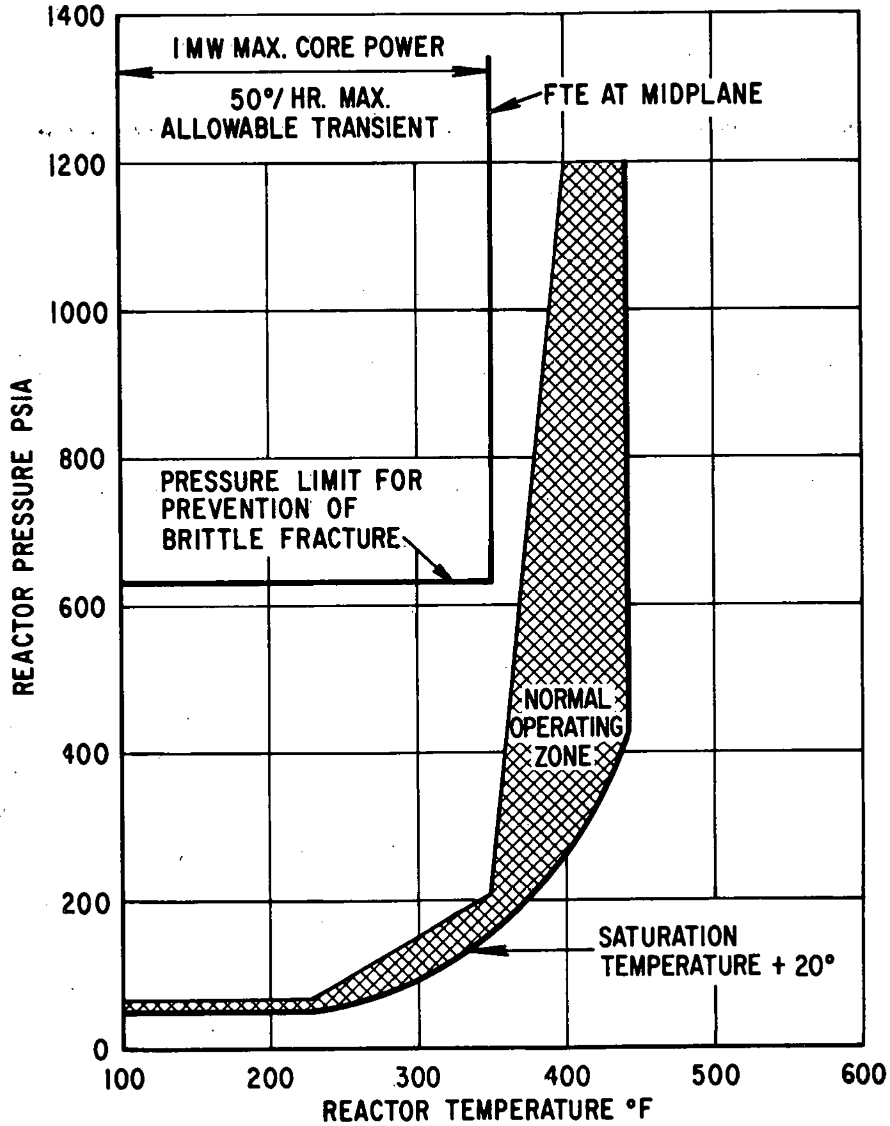


FIGURE 12
SM-1A OPERATING LIMITS FOR 62.6 MWYR

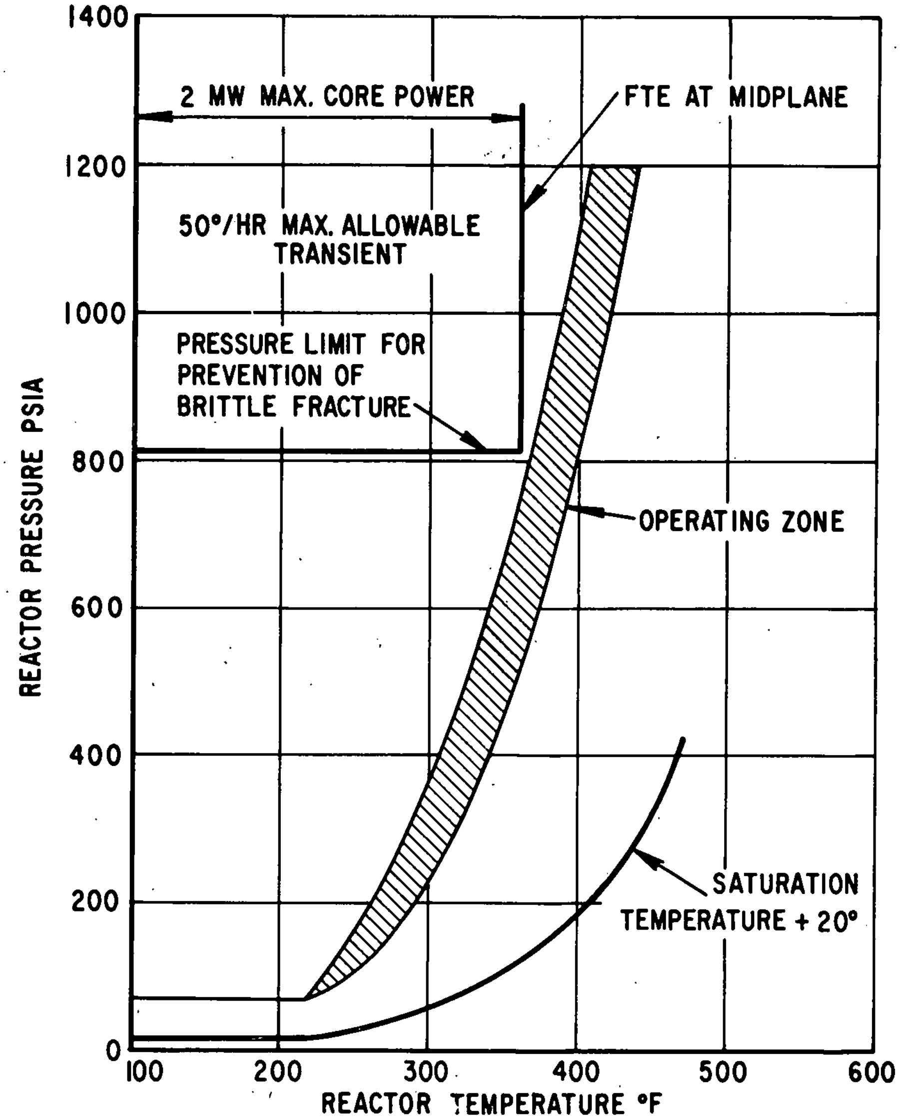
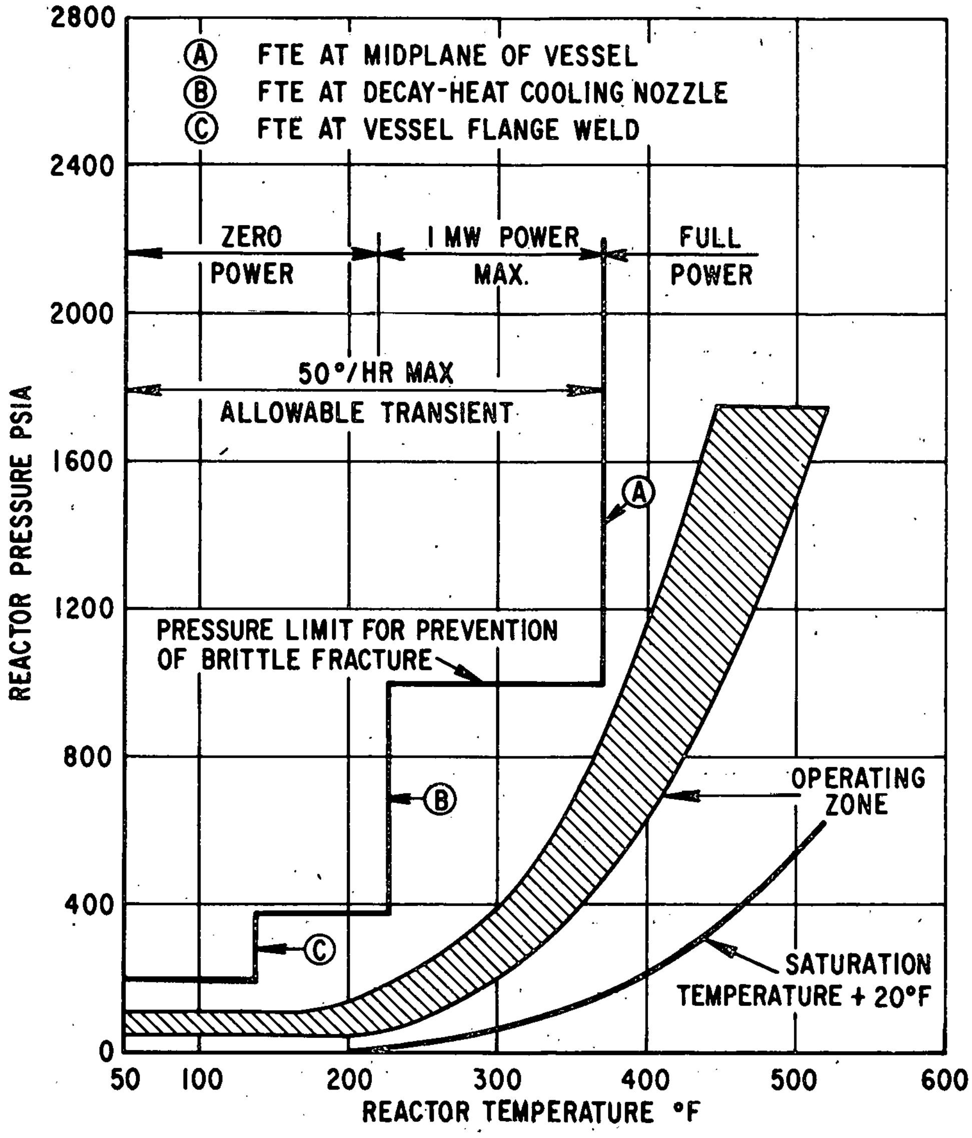


FIGURE 13
PM-2A OPERATING LIMITS FOR 12.3 MWYR



lower in temperature to prevent boiling in the core, or (2) the minimum pressure required for pump operation. The vertical (constant temperature) lines in the figures represent the FTE (NDT +60°) of some critical area of the vessel. Other operating restrictions required to limit stresses below the maximum FTE were that thermal transients could not exceed 50°/hr and that core power was limited as shown on the figures. The actual operating zone which appears on each figure was determined by NPFO and published in each operating manual.

In order to establish the operating limits, the maximum allowable NDT at the region of highest flux, the vessel mid-plane, was first determined. This maximum allowable NDT was determined from the premise that the vessel reached its service limit when the radiation-increased FTE temperature (NDT +60°) of the midplane had reached a level 60° to 70° below the reactor inlet temperature at operating conditions. Experience with SM-1 has shown that depressurization must be started when the system is still 60° to 70° above FTE to reduce pressure to the required level by the time FTE is reached.

The integrated fast neutron flux corresponding to the maximum NDT was obtained from the upper NRL curve, Fig. 7. The ratio of that flux to the calculated flux after a 20-yr lifetime determined the length of vessel life during which the operating limits were valid.

After satisfying the design criteria at the region of highest flux by setting the pressure to limit stresses to 18 percent of the yield strength, the process was repeated for other areas of lower flux and higher stress. In this way, the maximum allowable system pressure was defined for the range from reactor operating temperature to vapor container ambient temperature.

The flux intensity at the vessel wall falls off rapidly with axial distance along the vessel wall away from the midplane. It was found that the flux at the main coolant nozzles of the SM-1 and SM-1A vessels was so low compared to the midplane that the FTE at these points was not increased sufficiently to require a further pressure reduction to limit stresses. Thus, the SM-1 and SM-1A upper pressure limits required only one reduced pressure level to insure safety below the midplane FTE temperature. In contrast, the PM-2A required three steps in the maximum pressure limit at temperatures corresponding to the FTE of the midplane, the decay-cooling nozzle and the vessel flange weld. This was due to the smaller diameter and general compactness of the portable PM-2A vessel and the correspondingly higher flux intensity at the vessel wall.

The establishment of operating zones within these limits placed the operation of the SM-1, SM-1A and PM-2A vessels on a sound basis with regard to radiation damage. Figures 11, 12 and 13 show that the operating zones, specified in the operating manual of each reactor, fall below the maximum allowable pressure in each case. Consequently, at temperatures below the midplane FTE, stresses are always within the limits specified by the design criteria for the prevention of brittle fracture. In this manner, operating limitations have been imposed on each reactor to implement the results of the radiation damage studies. The operating curves will be revised periodically to incorporate new experimental data resulting from the continuing radiation damage program.

For the SM-1 the present operating curve, valid for 92.5 MWYR, covers the full 20 yr life of the reactor at its current operating load factor of about 46 percent. The flux levels on which the operating limits were based have been well established through calculations correlated with experimental measurements made on a mockup of the core and vessel. Thus, no further work is required on the SM-1 at its present level of usage. If greater utilization of the reactor were desired, some corrective action would be necessary. A study which confirmed the feasibility of annealing the vessel in place was completed during FY-61 and provides the necessary basis for final design of an annealing system if it should prove desirable.

The SM-1A limits will be valid for a period of 62.6 MWYR from startup, now planned for late 1961. At the anticipated load factor of 60 percent, or 12 MWYR per year, the operating curves will be valid for 5.2 calendar years. Under the FY-62 joint radiation damage program between the Army, Alco Products, Inc. and NRL, the means of extending this limit are being defined. Preliminary studies made during the FY-61 program indicated that the expected increase in yield strength due to irradiation will permit increasing the service life of the vessel by a factor of ten to 626 MWYR. This corresponds to more than the design service life of 20-years even at a load factor of 100 percent. During FY-62, data on the increase in yield strength will be obtained by NRL from specimens of SM-1A vessel material which have already been irradiated and are in storage pending completion of testing facilities. Further work on the SM-1A will include analysis of flux monitor wires which have been inserted in the vessel adjacent to the inner wall. This will provide a complete axial flux map at the vessel wall and an additional check on analytical flux calculations. These data and the flux measurements to be made on an SM-1A core-vessel mockup in the Alco Critical Facility, correlated with calculations described in Section 3.0, will provide a complete picture of the neutron flux intensity and distribution at the inner vessel surface. This information together with the metallurgical data from NRL should permit extension of the operating curves to cover the entire 20-yr design life of the vessel.

The PM-2A limits have the shortest duration of the three, conservatively determined to be 12.3 MWYR. This corresponds to a calendar year period of about 3 years at a load factor of 40 percent. For this reason, the corrective action required for extending the PM-2A service life has the highest priority in the 1962 program. Extension of the service life to 20.7 MWYR can be realized by use of the increase in irradiated yield strength. Further extension will be obtained when the apparently lower shift in transition temperature for this reactor is taken into account. NRL data (Figure 7) on PM-2A steel irradiated at the reactor temperature of 510^{o} indicate a 50^{o} to 60^{o} lower shift in transition temperature than was experienced by SM-1 and SM-1A material irradiated at 430^{o}. This is due to the higher operating temperature of the PM-2A. The transition temperature shifts used in establishing the present operating limits were taken from the upper edge of the NRL data band so that they are exceptionally conservative. Further data on both transition temperature and yield strength will be obtained by NRL during FY-62.

A program to provide for annealing of the PM-2A vessel to recover some of the transition temperature shift has also been planned. During FY-62, NRL will obtain data on PM-2A material which will have been subjected to two successive irradiation-anneal cycles in an effort to define reirradiation phenomena and a practical annealing temperature and time. At the same time, Alco will complete a basic study to select an annealing system which is scheduled for final design during the first half of FY-63. Under this schedule, annealing equipment can be available on-site by the time the current, conservatively-determined operating limits expire in April, 1964. Extension of the operating limits during the interim period through use of increased yield strength, the reduced transition temperature shift due to the elevated operating temperature, or establishment of an operating load factor of less than 40 percent, will permit some relaxation of the current schedule. More accurate determination of vessel wall flux is also scheduled. The calculated values are to be correlated with experimental data from core and vessel mockups. Data from measurements on a cold mockup are being interpreted at this time and will be available by the end of calendar year 1961. Experiments on a hot mockup will be performed in FY-62 and correlated with calculated fluxes. The latter experiment will reduce the error involved in correlation of cold mockup data.

The FY-61 Army radiation damage program provided the fundamental data and design criteria for the immediate establishment of conservative reactor operating limits. The continuing FY-62 program is aimed at defining the corrective action required at each plant to extend service life, and to do it before such action is actually required. In this way, the present control of the radiation damage problem will be continued for the full life of each plant.

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8.0 HAZARDS ANALYSIS OF REACTOR VESSEL FAILURE

The applicable summary reports for the SM-1, SM-1A, and PM-2A present the most recent available analyses of the hazards of reactor vessel failures for these types of vessels. The hazards summaries have been re-examined in light of the questions raised involving an assumed instantaneous brittle failure of the pressure vessel as a result of radiation damage. Since the interaction between the various items involved with this problem varies from plant to plant, the problem is treated independently for each plant.

8.1 SM-1 REACTOR VESSEL

8.1.1 Pressure Rise in the Vapor Container

If it is assumed that the reactor vessel undergoes a complete, instantaneous, brittle failure following the worst credible sequence of system failures, and the maximum storage of energy in the primary and secondary systems is released, then an accident is developed which has the same characteristics as the maximum credible accident described in the SM-1 hazards report(13). This accident will be contained by the system. The pressure has been calculated to rise to a peak of 82 psia as is shown in Fig. 14.

8.1.2 Vapor Containment Pressure Limitations

The SM-1 vapor containment vessel will have a peak stress of 18,000 psi or 56% of the minimum yield strength, when a pressure of 82 psia is attained within the containment volume. It is believed that stresses up to 80% of the yield strength may be tolerated for a vessel of this type and usage. It is thereby concluded that the vapor container will contain an accident of this type and that no hazard will be established outside of the immediate surroundings of the plant.

8.1.3 Missile Analysis

Previous to this, the failure of the SM-1 reactor vessel due to brittle failure has not been considered sufficiently credible to justify a complete and detailed missile analysis of reactor vessel fragments. Although embrittlement is most severe at the area of the vessel beltline, due to the high nvt at this area, a failure in this region would not be expected to release any significant missiles because of two reasons.

First, the vessel is anchored at a level which is above the beltline and failure of the vessel at the beltline cannot produce a missile which can violate the vapor container.

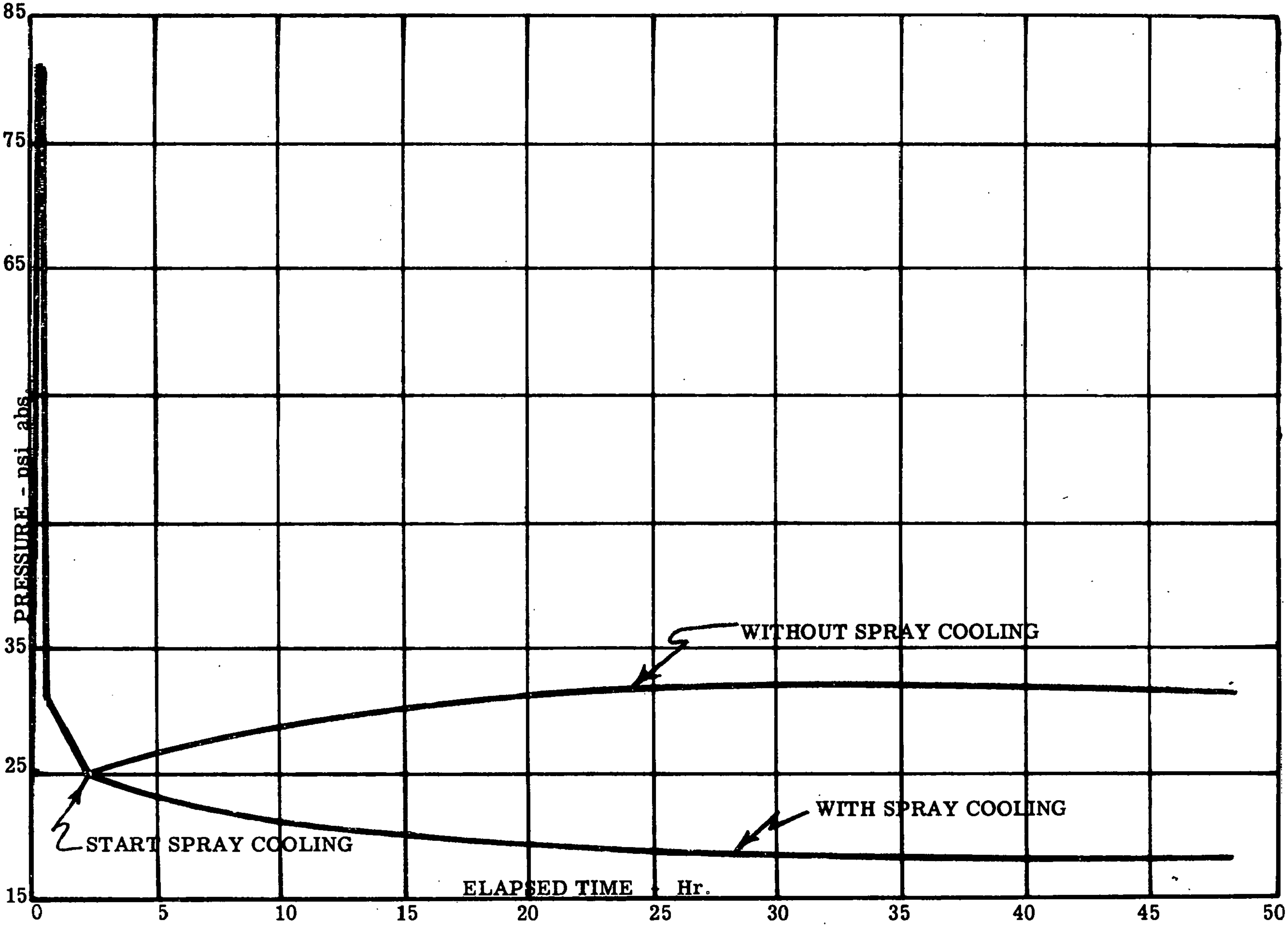

FIG. 14 PRESSURE VS. TIME IN VAPOR CONTAINER AFTER MAXIMUM CREDIBLE ACCIDENT

Secondly, the following paragraph taken from a preliminary report [14] BY G. R. Irwin et al, indicates that the probability of fragmentation of the vessel wall from rapid crack propagation would be unlikely. The report states in part,

> "The expected crack propagation failure would appear to be either a split through the chamber wall at the cylindrical belt which received most radiation damage or a split at the joint of a nozzle to the cylinder. In either case a long crack would result in a jet of water producing sudden lateral thrust as well as loss of water. Fragmentation of the vessel wall from rapid crack propagation would be unlikely; the probabilities favor the ability of the outer shields to enclose and limit the spread of damage due to the fracture."

8.1.4 Fuel Meltdown and Fission Product Release to Vapor Container

Upon a complete loss of coolant, which would occur in the event of brittle fracture of the reactor vessel, the SM-1 Core will begin melting in about 4 minutes. Limited experimental evidence [15] indicates that a maximum of about 17% of the fission product gamma activity may be expected to be released from the fuel if it melted in a steam environment. This release would amount to 5×10^6 equivalent curies after a longtime operation. For an accident occuring in a closed vapor containment, it is not too important exactly how long it takes for the fuel element melting to begin, or exactly how much activity release may be anticipated because the accident is contained. The ensuing accident has consequences approaching the magnitude of the maximum credible accident.

If primary system conditions are maintained such that a brittle failure of the reactor vessel is considered credible while the vapor containment is open, then an entirely different form of accident becomes possible. The environmental consequences approach as a most severe limit the consequences of a catastrophe as analyzed in the hazards report for the SM-1. For the catastrophe, the consequences are summarized in Table 9. The consequences are strongly dependent upon weather conditions. In the present hazards technology, the interpretation of these results is made for widely separate weather conditions. The resulting doses scope the extent of a catastrophe.

At present, the operating procedures [16] require that the power be below 200 kw for 2 hr, and the pressure be below 200 psi after reactor shutdown before the vapor container is opened. At the present time, the waiting period required after an extended full power run to assure that no melting of the fuel will occure upon a loss of coolant is not defined. The definition involves extensive analysis which cannot be completed without fundamental data on air film coefficients of open and blocked fuel element channels. It is suggested in Section 8.4 that such data be obtained by the AEC as part of its basic research program.

TABLE 9
RADII OF HAZARD AREAS FOR INTERNAL DOSAGE (METERS)

Wind Velocity = 2 m/sec

Organ	Type of Dose	Large Lapse n = 1/5		Small or Zero Lapse n = 1/4		Modified Inversion n = 1/3		Large Inversion n = 1/2	
		1 day	60 day	1 day	60 day	1 day	60 day	1 day	60 day
Lung	Lethal	14	0	15	0	15	0	15	0
	Emergency	170	0	220	0	275	0	350	0
	Evacuation	1100	0	1800	0	3400	0	6000	0
Thyroid	Lethal	0	0	0	0	0	0	0	0
	Emergency	245	700	605	1700	1600	4500	6800	19,000
	Evacuation	15,000	5,800	38,000	14,000	110,000	39,000	630,000	185,000
Bone	Lethal	0	290	0	570	0	760	0	2350
	Emergency	1000	2000	2200	4250	5,600	7,800	16,000	30,000
	Evacuation	1,170	2,900	2550	5800	6,400	12,500	19,000	48,000

Wind Velocity = 5 m/sec

Organ	Type of Dose	Large Lapse n = 1/5		Small or Zero Lapse n = 1/4		Modified Inversion n = 1/3		Large Inversion n = 1/2	
		1 day	60 day	1 day	60 day	1 day	60 day	1 day	60 day
Lung	Lethal	18	0	18	0	18	0	18	0
	Emergency	190	0	250	0	320	0	410	0
	Evacuation	1300	0	2100	0	3900	0	7600	0
Thyroid	Lethal	0	0	0	0	0	0	0	0
	Emergency	70	400	220	940	440	1500	2350	12,000
	Evacuation	10,500	3800	27,000	9,500	78,000	27,500	500,000	160,000
Bone	Lethal	0	84	0	225	0	505	0	1650
	Emergency	650	1500	1500	3200	3800	7600	13,000	28,500
	Evacuation	780	2050	1750	4800	4400	11,500	15,500	45,000

8.1.5 Fisson Product Release from Vapor Containment

The initial design and testing of the SM-1 vapor containment was based upon a maximum permissible gross leak rate of 1.8 ft^3/day at 66 psig. Complete retesting to this specification of 66 psig is not feasible. However, retesting is planned in October-November 1961, by a pressure loss technique. Test procedures for this have been written. It is expected that the maximum sensitivity for the retest will be 100 ft^3/day. The resultant hazards of a leak rate of this magnitude are discussed in the following paragraph.

8.1.6 Resultant Hazards to Personnel

The hazards resulting from leakage of fission products from the vapor containment have been analyzed based upon the maximum design leak rate of 1.8 ft^3/day, and an assumed gross leak rate of 100 ft^3/day. Conservative meteorological assumptions were made. The calculated exposure based on the max. design leak rate was 0.002 curie-seconds/$meter^3$ at the nearest dwellings, in Fairfax Village, when they are directly downwind. This is based on the above maximum design leak rate and 0.1 C-sec/m^3, using 100 ft^3/day leakage. These exposures, accumulated in the first 48 hr, should account for most of the total exposure. Further, it is scarcely conceivable that a wind should be flowing in exactly the same direction for the entire 48-hr period. The limiting exposure is considered to be 0.1 C-sec/m^3. Levels beyond this should be avoided by evacuation or othermeasures(17). Therefore, there is no serious problem with the design leak rate assumed. At the higher assumed leak rate, moreover, the exposure barely reaches an evacuation dose.

Evacuation procedures have been formulated for various types of nuclear incidents at the SM-1. If the maximum design leak rate specified above, is maintained, there is no need for evacuation of the SM-1 area as a result of any credible accident. If a leak rate as great as 100 ft^3/day is experienced, then evacuation for varying time periods may be required to distances as great as 75 meters from the leak. Since access control is maintained, emergency procedures are prepared, and warning systems are installed, no uncontrollable hazards can develop.

8.2 SM-1A REACTOR VESSEL

8.2.1 Pressure Rise in the Vapor Container

Brittle failure of the SM-1A reactor vessel will yield an accident having the same characteristics as the maximum credible accident for the SM-1. The pressure in the inner containment may be expected to reach 118 psia and the pressure in the outer containment may reach as much as 28.6 psia. The hazards report for the SM-1A(18), gives details of the calculations.

8.2.2 Vapor Containment Limitations

The inner vapor containment at the SM-1A was designed for a maximum pressure of 135 psia and the outer containment was designed for 29.7 psia. Therefore, it is considered to be able to contain the accident defined by this report.

8.2.3 Missile Analysis

An extensive missile analysis assuming a brittle failure of the reactor vessel is not considered necessary for reasons stated in the discussion of the SM-1

8.2.4 Fuel Meltdown and Fission Product Release to Vapor Container

Upon a complete loss of coolant, the SM-1A core will begin melting in about 2.3 minutes following an extended full power run. Under these conditions, about 80% of the fuel may be expected ultimately to melt. The percentage and type of release will parallel that of the SM-1. The gross gamma activity release will be about 1×10^7 equivalent curies. About 0.05% of the gross fission products or 5×10^3 equivalent curies will escape through the moat into the outer containment.

As with the SM-1, if the SM-1A vapor containment is open and the primary system is in a condition in which brittle vessel failure is considered credible, and accident may ensue which has consequences approaching those of the catastrophe. This catastrophe is defined in the hazards report for the SM-1A.

8.2.5 Fission Product Leakage from Vapor Containment

Helium leak testing of the SM-1A containment vessel at construction showed no leaks when the sensitivity of the instrumentation was set at 1×10^{-8} cm^3/sec. Conservative assumptions were used to conclude that leakage may not be expected to exceed 3.2×10^{-4} ft^3/hr from the entire vapor container under the conditions anticipated following the maximum credible accident. The fission products which escape the moat and become available for leakage to the environment consist principally of the inert gases. The other fission products have been found by experimentation to be substantially washed out in the vapor suppression(19).

8.2.6 Resultant Hazards to Personnel

The radiological hazards of a contained maximum credible accident center around exposures at the auxiliary power plant. Under the conditions postulated here and with the maximum leakage which could have escaped detection, the exposure dose rate at the auxiliary power plant, from radioisotopes contained in the outer containment vessel following the maximum credible accident, will decrease from 1300 mr/hr immediately to 400 mr/hr at one hour after the accident. After one day the dose rate will be reduced to 100 mr/hr. No one in this area will accumulate more than the emergency exposure of 25 rem.

8.3 PM-2A REACTOR VESSEL

8.3.1 Pressure Rise in the Vapor Container

An instantaneous, complete, brittle failure of the PM-2A reactor vessel would release the energy stored by the water in the reactor vessel to the annulus between the vessel wall and the shield tank wall. It wad estimated in the PM-2A hazards report [20] that the sudden release of the energy to this annulus will result in a pressure of 1200 psi in the annulus. This will result in rupture of the bellows, since its design pressure is 15 psi. Therefore, the 1200 psi pressure will be felt in the vapor container dome region. The weakest point in the dome region is at the reinforcement pad where the dome joins the vapor container shell. This point is calculated as having a bursting pressure of 450 psi so that it will probably fail. It has not been established however, whether failure of the dome would occur inside or outside of the vapor container shell.

If failure of the dome occurs outside the vapor container shell, it will occur below the junction of the dome and the upper shield tank and the integrity of the vapor container will be lost. If failure of the dome occurs inside the vapor container proper, the pressure in the vapor container will rise to 140 psi and containment integrity will be maintained since the vapor container was designed to the ASME Unfired Pressure Vessel Code for 150 psi pressure.

8.3.2 Missile Analysis

No analysis of the results of vessel fragmentation was considered necessary for the reasons stated in the discussion of the SM-1 and for the reason that the vapor containment cannot be guaranteed in any case assuming an accident of the type discussed here.

8.3.3 Fuel Meltdown and Fission Product Release to Vapor Container

Upon complete loss of coolant, melting of the PM-2A core will begin within about 3-1/3 minutes. The amount and type of fission products released are essentially the same as in the SM-1. As with the SM-1 and SM-1A, if the primary system is in a condition in which its failure is considered credible, a fuel meltdown may occur. Unlike the SM-1 and SM-1A however, little of the fission product release is expected to reach the atmosphere. This is based on the fact that no auxiliary power is immediately available for the tunnel ventilation system, in the event of reactor failure. The release of primary water would be completed and the resultant pressure buildup in the tunnel would be quenched before fuel meltdown occurred. No driving force would then remain to carry the fission products out of the reactor tunnel.

8.3.4 Fission Product Release from Vapor Containment

Helium leak tests were made on the vapor containment after site assembly. No leak above 3×10^{-6} cc/sec was observed. Any fission products which might leak would be expected to have the same characteristics as those released from the fuel upon meltdown.

8.3.5 Resultant Hazards to Personnel

Even under the very conservative assumption that the fission products released from the molten core are all released to the atmosphere, no severe radiation hazards will result. Under these conditions and assuming a wind blowing toward inhabited areas of the camp (which has never occurred to the best of our knowledge) the maximum exposure to personnel in the tunnels of the camp is less than the maximum permissible emergency exposure dose (25 rem). The nearest off-site habitation is at Thule Air Force Base, over 100 miles away. The emergency dose will be experienced at a distance no greater than 40 miles from the release site.

8.4 CONCLUSIONS AND RECOMMENDATIONS

Based on current knowledge, the SM-1 and the SM-1A vapor containers will contain an accident involving a brittle failure of the reactor vessel. It is not certain whether the PM-2A vapor container can do this; therefore additional work in this area has been recommended.

In order for Alco Products to perform further evaluation of the type of accident indicated in this report, the basic information listed below will be required. This information would properly be obtained by the AEC as part of its basic research program.

1. Experimental verification of the expected mode of failure of an embrittled, irradiated vessel. In the event that fragmentation does occur, data would be required on the size, number and velocity distribution of the missiles formed.

2. Measurement of the amount and type of fission products released from freshly spent, high burnup, stainless steel type fuel elements .

3. Measurement of air film coefficients in both open and blocked fuel element channels.

This basic work would be used to define hazards parameters for the SM-1, SM-1A, and PM-2A. Additional work should also be done to determine pressure buildups in the reactor vessel area of the PM-2A, postulating an accident of the type discussed in this report.

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