01/14/83

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CONSUMERS POWER COMPANY	)	Docket Nos. 50-329 OM & OL
	)	50-330 OM & OL
(Midland Plant, Units 1 and 2)	)	

NRC STAFF FURTHER RESPONSES TO INTERROGATORIES SUBMITTED BY BARBARA STAMIRIS ON AUGUST 30, 1982

I. INTRODUCTION

On August 30, 1982, Intervenor Barbara Stamiris filed "Stamiris Interrogatories and Document Requests to Nuclear Regulatory Commission." By letter dated September 13, 1982, the Staff informed the Board that it would voluntarily respond to those interrogatories. On November 3, 1982, the Staff submitted NRC Staff Partial Response to Intervenor Barbara Stamiris' Interrogatories and Document Request to the Staff, dated August 30, 1982. This is a response to "Interrogatory I" of Ms. Stamiris' August 30, 1982 submittal. Responses to the remainder of the interrogatories contained in that submittal will be filed at a later date.

II. RESPONSES TO INTERROGATORIES

Interrogatory I: COST/BENEFIT: CONTENTION 1b and 1c

1. Is the section 6 DES, FES analysis performed by the NRC intended to represent the cost/benefit analysis to the public of operating the Midland plant?

DESIGNATED ORIGINAL

Certified By Ken Saffer

DS07

Response

Yes. By the phrase "to the public," this question is understood to ask if the customers of Consumer Power Company are included as beneficiaries of electricity produced by operating of the Midland plant.

2. What was the overall life expectancy used in the DES calculations of decommissioning as representing 5% of lifetime cost savings? Do you consider that decommissioning will validly represent only 5% of the life time cost savings of the plant"? Explain.

Response

The overall life expectancy used in the DES can be determined to be 40 years (see below). The savings referred to in this question are those which are estimated to accrue from operating the Midland facility in lieu of more costly, existing generating facilities on the applicant's system. The statement which relates these savings (or, more appropriately, "avoided costs") to the cost of decommissioning, was made in the DES (§ 2.2).

This statement was deleted from the FES in light of the Commission ongoing rulemaking proceeding regarding decommissioning. (See FES, § 5.11). However, its original intent in the DES was to place the applicant's estimate of costs of decommissioning in some perspective. The approximate 5% figure can be determined using the following formula:

$$\text{Percent of Total Lifetime Saving Represented by Decommissioning Cost} = \frac{\text{Cost of Decommissioning (1984 dollars)}}{\text{(1984 Savings from Table 2.2) x 40 years)}} \times 100$$

or

$$4.45 = \frac{\$235 \times 10^6}{(\$132 \times 10^6) \times (40)} \times 100.0$$

3. What is the NRC's projected life expectancy for Units 1 & 2 respectively and at what respective capacities?

Response

The NRC relies on the applicant's estimate of the life expectancies for the units. The NRC has issued operating licenses for commercial nuclear power facilities for forty year periods. Licenses are conditioned, pursuant to Commission's Regulations, to provide for revocation, modification, or suspension should the facility be unable to operate safely.

Staff assumes the question to refer to "capacity factor" rather than capacity. Staff assumes that capacity factors of a nuclear generating facility would typically average between 55 and 65 percent during its operating life with factors near the lower end of the range more likely to be experienced than factors toward the upper end. The estimate of a 66 percent capacity factor for the Midland facility as suggested in the applicant's production cost analysis (FES, § 6.4.1), appears to be somewhat optimistic. However, Staff does not view this estimate as unreasonable.

4. In estimating decommissioning costs--why doesn't the NRC consider the total costs to be collected over the operating lifetime of the plant?

Response

At the present time, there is no requirement in effect which requires utilities to accumulate funds for decommissioning. In its Final Rule on decommissioning the Staff will recommend that utilities be

required to accumulate funds on an annual basis to cover decommissioning costs.

5. Do you consider that decommissioning will validly represent only 5% of the lifetime cost savings of the plant? Explain.

Response

See response to Question 2.

6. How have you estimated the lifetime cost of permanent plant dewatering in your cost/benefit analysis for plant operation and maintenance?

Response

No estimate for the lifetime cost of permanent plant dewatering has been included in the cost/benefit analysis for plant operation and maintenance. The Staff believes, however, that the operation and maintenance expense of dewatering will be less than the cost of its installation - $1.2 million. Hence, operation and maintenance expenses of the dewatering system would be too small to have an effect on the cost/benefit analysis.

7. Give the years, and amounts of initial decommissioning cost estimates provided the NRC at the time of beginning and completion for Big Rock, Palisades, and Midland nuclear plants.

Response

Cost estimates for the decommissioning of the Midland Plant are contained in Sections 5.8 and 8.2 of the Environmental Report (OL Stage), submitted by Consumers Power Company to the NRC. The total cost to decommission the 2 units at Midland is $235 million in 1984 dollars as

indicated in Section 8.2 of the Environmental Report. This estimate is based on Section 5.8 which indicates in Table 5.8-1 that the cost of decommissioning for each unit at the Midland Plant is $82.4 million in 1981 dollars. In addition, Table 5.8-1 indicates that the cost of restoration of the site by dismantling the process evaporator, removing the cooling pond and structures, and relandscaping is $39.5 million in 1981 dollars.

For Palisades and Big Rock Point, operating licensing activities took place more than 10 years ago. More recent estimates of costs of decommissioning for those two plants are contained in a report entitled, "Analysis of Nuclear Power Reactor Decommissioning Costs," AIF/NESP-021, May 1981, prepared by Stone and Webster Engineering Corp. under sponsorship of the Atomic Industrial Forum's National Environmental Study Project, (the NESP report). This report contains a comparison of decommissioning cost estimates made by different sources for 8 PWRs and 5 BWRs, and for 5 generic studies. The source of the information in the NESP report for Palisades and Big Rock Point is a report entitled, "A Nuclear Power Plant Decommissioning Study, Report on Big Rock Point and Palisades Nuclear Plant," Consumers Power Company Nuclear Plant Decommissioning Task Force, November 1978. As presented in the NESP report, the cost estimates for decommissioning of Palisades by prompt dismantlement is $60.7 million in 1978 dollars including $13.4 million for demolition of structures and a 25% contingency factor. The cost estimates for Big Rock Point by prompt dismantlement is $31.4 million in 1978 dollars which includes $5 million for demolition of structures and a

25% contingency factor. (Copies of applicable pages of the NESP report are attached). (Attachment 1)

In reviewing these cost estimates, it should be noted that it is difficult and perhaps misleading to make simple comparisons between different cost estimates for different plants. This is because there are many site-specific and facility-specific considerations in cost estimates of specific facilities. These considerations include such items as plant design and operating conditions, costs specific to a particular location such as taxes and labor costs, and other differing assumptions. For example, the Midland Environmental Report (OL Stage) indicates that the Midland Plant decommissioning costs include $39.5 million of costs which include restoration of the cooling pond area to its approximate condition prior to site preparation, which is a local requirement, and removal of process steam evaporators which are unique to Midland.

It should be noted that, although the power level of Big Rock Point is one-tenth that of the other facilities, its decommissioning cost is not equally small. This is reasonable based on a decommissioning study done by Battelle Pacific Northwest Laboratory (PNL) for the NRC, "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Station," NUREG/CR-0130, (June 1978, and Addendum, August 1979). (A copy has been placed in local public document room in Midland, Michigan, and is available in the public document room in Washington, D.C.) This PNL study developed a method for evaluating the effect of plant size on decommissioning. This method is general and cannot be used alone in estimating decommissioning costs (as indicated above, various plant-specific factors must be considered in arriving at

cost estimates). Nevertheless, the method is useful in understanding how power level can affect costs. The PNL study found that the relationship between power level and cost is more complicated than direct extrapolation of power levels, and that costs are more dependent on plant factors such as dimensional or physical characteristics of components to be dismantled such as mass, volume, and type of component. For example, as indicated in the Addendum to NUREG/CR-0130, although the power level of one of the plants studied by PNL (the 600 Mwt Yankee Rowe plant) is one-sixth that of the reference plant (the 3500 Mwt Trojan plant), its cost of decommissioning is estimated in the PNL study to be approximately one-third that of the reference plant.

8. Provide the most recent NRC generic decommissioning document, SRP, Reg Guides or other decommissioning criteria.

Response

(A) Decommissioning Criteria

Regulations specifically applicable to the decommissioning of power reactors operated by electric utilities, such as is the situation for the Midland Plant, are contained in the Code of Federal Regulations, 10 C.F.R. § 50.82, "Application for Termination of Licenses." Section 50.82 specifies the requirements that must be satisfied to terminate an operating license, including the requirement that the dismantlement of the facility and disposal of the component parts not be inimical to the health and safety of the public.

Regulatory Guide 1.86, "Termination of Operating License for Nuclear Reactors," June 1974, (Attachment 2) describes methods and procedures considered

acceptable by the NRC Staff for termination of operating licenses. Regulatory Guide 1.86 describes alternatives for reactor retirement, contents of applications for a possession-only license, and criteria for decontamination of a reactor for unrestricted use.

Other requirements are applicable to specific activities carried out during the decommissioning, such as, for example, packaging and shipping of radioactive wastes. A discussion of these requirements is contained in Section 5 of the NRC document, NUREG/CR-0130, discussed in the Answer to Stamiris Interrogatory #7.

(B) Generic Decommissioning Documents

Recent NRC generic decommissioning reports pertinent to the Midland Plant are enclosed. These include the NRC sponsored study entitled, "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor," NUREG/CR-0130, prepared by Battelle Pacific Northwest Laboratory, (June 1978, and Addendum, August 1979) (P&L) which is a detailed engineering study of the conceptual decommissioning of a large reference PWR; and the "Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0586, USNRC, January 1981. (Copy has been placed in local public document room in Midland, Michigan, and is also available in the public document room in Washington, D.C.)

9. What is the largest nuclear facility decommissioning thus far by the prompt removal/dismentlement method? Explain in detail its decommissioning and how projections are made from this experience to large commercial reactors.

Response

The largest nuclear power reactor decommissioned thus far is the Elk River Reactor located in Minnesota. This was a 58 Mwt boiling water reactor operated by United Power Association under contract to the Atomic Energy Commission. Elk River was dismantled between 1971 and 1974. A discussion of the Elk River decommissioning is contained in Section 3.1.9 of NUREG/CR-0130 and a more detailed discussion is contained in the paper, "Elk River Reactor Decommissioning," B. J. Davis, Proceedings of the First Conference on Decontamination and Decommissioning of ERDA Facilities, CONF-75-827, p.83, August 1975 (Attachment 3). The following paragraphs discuss the question of how projections are made from this experience to large commercial reactors.

As indicated in the response to Interrogatories #8 and 9, NUREG/CR-0130 is a detailed engineering study of the technology, safety and costs of the decommissioning of a large reference PWR (the 1175 Mw(e) Trojan Nuclear Plant is used as the reference plant). Section 3 of NUREG/CR-0130 presents a review of decommissioning experience accumulated to date, which includes Elk River, as well as other smaller facilities. It is concluded in Section 3 that because of differences in reactor size, type and design, operating time, licensing requirements, and location, that, for items such as cost and radioactive waste volume, extrapolation from previous experiences at small facilities to large commercial reactors is considered to be generally unreasonable.

Section 3 of NUREG/CR-0130 does also conclude that "the primary value of past decommissioning experience is that it identifies the individual measures required for decommissioning." Thus, NUREG/CR-0130,

in developing estimates of cost for large reactors, used the previous experience in its analyses by including methods and techniques that had been demonstrated by experience as being available and successful. These include decontamination methods, dismantling techniques, radiation safety measures, and other procedures. In making the cost estimates for large reactors which are contained in NUREG/CR-0130, P&L utilizes this information, where applicable. P&L also considers the design and plant layout of the large reactors, and estimated conditions in the reactor at the time of shutdown, including estimates of radionuclide inventory and radiation dose rates. In addition, decontamination techniques and radiation protection measures are revised so as to be appropriate for large reactors.

Based on these considerations, NUREG/CR-0130 develops detailed work plans and time schedules to accomplish decommissioning, including those for planning and preparation, decontamination, and component disassembly and transport. In making cost estimates of decommissioning NUREG/CR-0130 includes such matters as work scheduling estimates, staffing requirements, specialty contractors, essential systems, radioactive materials disposal, and supplies. The details of how this analysis is done are described in more detail in Sections 4.2, and 9, and Appendices F and G of NUREG/CR-0130.

10. Explain the apparent discrepancy between EFPY estimates for Unit I given on p. 5-19 and C-10 of the SER.

Response

The effective full power years (EFPY) estimates on pp. 5-19 and C-10 of the SER were calculated by two different staff reviewers using two different sets of data.

The 9 EFPY prediction on p. 5-19 was predicted using surveillance weld data from Babcock & Wilcox Report BAW-1699, "Analysis of Capsule OC II-A from Duke Power Company's Oconee Nuclear Station, Unit 2". The weld material tested was WF-209 which the reviewer considered representative of the Midland Unit 1 WF-70 weld material. WF-70 and WF-209 were both fabricated by Babcock & Wilcox using the same heat number of filler wire, the same type of flux and having similiar copper, nickel and phosphorus chemical composition. The BAW-1699 report indicates that at 3.37×10^{18} n/cm^2 ($E_7$1MeV) the charpy V-notch upper shelf for the WF-209 material will be less than 50 ft-lbs energy absorption. Hence, the reviewer used 3.0×10^{18} n/cm^2 ($E_7$1MeV) as an estimate of the neutron fluence in which the Charpy V-Notch upper shelf energy absorption for NF-70 material would be less than 50 ft-lbs. At the 1/4 vessel wall thickness, the Midland Unit 1 WF-70 weld material will accumulate 3.0×10^{18} n/cm^2 ($E_7$1MeV) at approximately 9 EFPY.

The 15.1 EFPY prediction on SER p. C-10 used the prediction method identified in proprietary Babcock & Wilcox Report BAW-1511 P, "Irradiation - Induced Reduction in Charpy Upper-Self Energy of Reactor Vessel Welds". BAW 1511, April 1980 is a non-proprietary version of this B&W report. The prediction method for determining the drop in upper shelf energy in this report is based upon surveillance data from 37 sets of operating plants, but does not include the data from Babcock & Wilcox

Report BAW-1699. Using the prediction method in BAW-1511 P, the WF-70 weld material's upper shelf energy will drop to 50 ft-lbs when the accumulated neutron fluence reaches approximately 5×10^{18} n/cm^2 ($E_7$1Mev). At 1/4 of the vessel wall thickness, the Midland Unit 1 WF-70 weld material will accumulate approximately 5.0×10^{18} n/cm^2 (E 1MeV) in 15.1 EFPY.

The B&W Owner's Group Surveillance Program includes WF-70 weld metal samples for actually determining accumulated neutron irradiation damage. These samples will be periodically removed and tested to determine the Charpy V-Notch upper shelf energy. The B&W Owner's Group Surveillance Program contains a sufficient number of samples to determine the actual neutron fluence at which the Charpy V-Notch upper shelf energy of WF-70 falls below 50 ft-lbs. These test data will be available before Midland reaches 9 EFPY; hence, the applicant and the NRC will have sufficient time to take appropriate corrective actions, if needed. Therefore, and as further discussed in reponse to Question 21 of this set of interrogatories, the apparent discrepency between the two estimates is not significant in terms of the basis which will be followed to establish the true life of the reactor vessel of Midland Unit 1.

11. What is the EFPY estimate for Unit I and how is this taken into account in the estimated life expectancy for the cost-benefit analysis?

Response

EFPY estimates for Unit 1 are discussed in Question 10 of this set of interrogatories. No specific EFPY is taken into account for the FES cost-benefit analysis. Rather, as summarized in FES Section 6.4, average

annual savings were derived from the period 1984 through 1988. Furthermore, as discussed in the response to Question 21 of this set of interrogatories, the estimated life expectancy for both units is the same.

12. Explain in laymans terms the significance of the I.D. surfaces of RT-NDT temperatures (p. 5-25 SER) and of the large difference (180° after 4 calendar years for Unit I, 46° for Unit II) in limiting welds.

Response

SER Section 5.3.5 (pp. 5-24 and 5-25) provides the staff's assessment of the significance of pressurized thermal shock on the Midland Units 1 and 2 reactor vessels. Reactor vessels become susceptible to pressurized thermal shock as the reactor vessel material's fracture toughness decreases. Fracture toughness will decrease as a result of neutron irradiation. The inside diameter (I.D.) surface of a reactor vessel beltline is the reactor vessel location which will accumulate the greatest amount of neutron fluence and, therefore, will have the greatest decrease in fracture toughness.

A measurement of a material's fracture toughness is the materials reference temperature, RT_{NDT}, which is defined in Section III of the ASME Code. The RT_{NDT} at the ID surface is a measure of the maximum amount of reduction of fracture toughness and the vessel's susceptibility to pressurized thermal shock. The staff estimates that after 4 calendar years the Midland 1 limiting weld's ID surface RT_{NDT} will increase to 180°F and that the Midland 2 limiting weld's I.D. surface RT_{NDT} will increase to 46°F. These estimates indicate that the limiting Midland 1

reactor vessel materials will have less fracture toughness than the limiting Midland 2 reactor vessel material after 4 calendar years of neutron irradiation. Hence, the Midland 1 reactor vessel will be more susceptible to pressurized thermal shock than the Midland 2 reactor vessel.

13. When and how did the NRC first become aware of the defective material in reactor welds manufactured by B&W such as Midland's?

14. When was the NRC first aware of the safety significance of this defective weld material?

Response to Questions 13 and 14

The Staff learned of the Midland vessel weld by 50.55(e) report 73-01 on January 16, 1973. The AEC (the NRC's predecessor) was aware that high copper content in vessels could be of safety significance before the end of 1973 based on a research program by the Naval Research Laboratory. The AEC's understanding of the significance of high copper materials is documented in an internal AEC report prepared by F. B. Litton, "The Effects of Residual Elements on the Sensitivity of Pressure Vessel Steels to Irradation Embrittlement," dated October 1973. (Attachment 4). Further appreciation of the problem with respect to commercial nuclear power plants, indicating that high copper welds in reactor vessels would cause a substantial increase in RT_{NDT}, was gained when Charpy V-Notch impact test data became available from the R. E. Ginna Unit 1 reactor vessel material surveillance program. The Ginna data was documented in Westinghouse Report WCAP-8421, dated November 1974. The staff discussed the chemical composition

of the Midland Unit 1 vessel beltline weld during a November 3, 1976 meeting with the NRC, B&W, and several utilities with B&W plants.

15. When and how was CPC first informed of the NRC's safety concerns about this defective weld material?

Response

On June 19, 1973, a letter to CPC from the AEC Directorate of Regulatory Operations addressed this subject. Also, in July 1975 the NRC issued for comment Regulatory Guide 1.99 (Attachment 5) which notes the adjustment of reference temperatures as a function of neutron fluence and copper and phosphorus chemical composition. (Revision 1 to this Regulatory Guide, dated April 1977 is also attached.) Close out of related Midland inspection concerns is addressed by NRC Inspection Report 79-15, page 4.

16. Did CPC seek NRC concurrence prior to installation of the reactor units? Was the NRC aware of CPC's installation plans for the reactors?

Response

a) Installation of reactor vessels is one of numerous construction activities authorized by issuance of a construction permit. In the absence of any action by the Commission to modify, suspend or revoke a valid construction permit, specific concurrence for such activities is not needed. Thus, no concurrence apart from the granting of construction permits was needed for installation of the Midland vessels and none was requested.

b) Yes.

17. Were any tests of studies performed to better understand the reactor weld material and its potential safety implications prior to installation? If yes, explain and provide study results.

Response

Early industrial concern for the effects of residual elements and neutron irradiation on a material's fracture toughness is documented in the report identified in the response to questions 13 and 14 of this set of interrogatories.

18. Why were preventative measures for defective reactor welds not required prior to installation?

Response

As noted in the responses to Questions 15 and 16 of this set of interrogatories, the initial issuance of Regulatory Guide 1.99 preceded installation of the Midland reactor vessels. This Regulatory Guide, in conjunction with 10 C.F.R. 50 Appendices G and H, identified "preventative measures" (i.e., general procedures acceptable to the NRC staff) to avoid rapidly propagating fracture of reactor vessels.

19. Would thermal annealing be less costly and safer prior to plant operation? Explain.

Response

Although thermal annealing will be less costly and safer prior to plant operation, it only becomes useful after a reactor vessel has accumulated neutron irradiation damage (See responses to questions 20 and 23).

20. Are there other operating plants with defective welds which could benefit from such an annealing experiment at Midland prior to operation? Explain.

Response

Thermal annealing has been shown to improve a materials fracture toughness after it has been reduced as a result of neutron irradiation. Annealing a vessel prior to operation (or while the vessel is not operating) would be of no value with respect to resisting neutron radiation damage. Thermal annealing prior to operation would, therefore, not benefit any plant.

21. What would the estimated life expectancy of Unit I be if thermal annealing were not possible?

Response

Even if thermal annealing were not possible, the estimated life expectancy of Unit 1 would still be 40 calendar years. In lieu of thermal annealing, the applicant could prolong the vessel life by instituting various alternative remedial actions. Examples of these alternatives include a core flux reduction program, improved non-destructive examination techniques, and plant modifications to reduce the susceptibility to pressurized thermal shock on the vessel.

22. How is the NRC judgment made that the public is served by not imposing certain Appendix G & H requirements for reactor weld quality--because of impracticality, hardship, or unusual difficulties to the applicant "without a compensating increase in the level of quality and safety." (p. 5-21 SER)? In answering this question:

    a. Explain in detail how this weighing type judgment is made,

b. What criteria are used to balance safety against factors such as impracticality, hardship, difficulty or others.

c. Name the key individuals responsible for this judgment.

d. Explain whether CPC knowledge of reactor weld deficiencies led to deliberate avoidance of proper weld tests and samples discussed in SER section 5.3.1.2. and 3.

Response

a&b. The NRC evaluates the alternative test program proposed by the applicant. If the applicant's program provides a level of safety and quality equivalent to that required by Appendices G and H, 10 C.F.R. Part 50, then an exemption from the specific requirements of Appendices G and H, 10 C.F.R. Part 50 can be granted pursuant to 10 C.F.R. 50.12 NRC decisions regarding exemptions are made with the benefit of technical specialists (in this case, experienced materials engineers) who use their engineering judgment to determine whether the proposed alternative test program provides a level of quality and safety equivalent to that required by Appendices G and H, 10 C.F.R. Part 50.

c. The CPC request for exemption from the requirements of Appendices G and H, 10 C.F.R. Part 50 was reviewed by Mr. Peter Nagata of EG&G, Idaho National Engineering Laboratory.

The NRC technical monitor who was responsible for evaluating the recommendations of Mr. Nagata was Mr. Barry J. Elliot of the Materials Engineering Branch, Division of Engineering, NRR, NRC. The recommendation to grant exemption to the licensee was reviewed and concurred upon by Mr. Warren S. Hazelton, Section Leader and Acting Branch Chief of the Materials Engineering Branch and by Dr. William V. Johnston, Assistant Director for Materials and Qualification

Engineering, Division of Engineering, NRR, NRC. Other management pursuant in the DE, DL and ONRR concurred in the decision.

d. The Staff has no reason to believe that CPC's knowledge of reactor weld deficiencies led to deliberate avoidance of proper weld tests and samples.

23. Why isn't preventative annealing of Unit I required prior to operation?

Response

Reactor vessels may be thermally annealed to reverse the reduction in fracture toughness resulting from neutron irradiation damage which has already occured. Annealing a vessel prior to its receiving neutron irradiation, therefore, would be of no value.

24. How does the B&W Surveillance program (SER p. 5-21) protect against the possibility of sudden rupture in PTS raised by Basedekas?

Response

The B&W Owner Group Surveillance Program will provide fracture toughness data for the limiting weld material in the Midland 1 reactor vessel. This fracture toughness data will be utilized to determine the ability of the Midland 1 vessel to resist a pressurized thermal shock.

Interrogatory II: QA QUESTIONS

To be answered at a later date.

Interrogatory III: EFFECTS DEWATERING: CONTENTION 3

Answers filed November 3, 1982.

Interrogatory IV: INDEPENDENT AUDIT: CONTENTION 4

To be answered at a later date.

Respectfully submitted,

Michael N. Wilcove
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 14th day of January 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CONSUMERS POWER COMPANY	)	Docket Nos. 50-329 OM & OL
	)	50-330 OM & OL
(Midland Plant, Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF FURTHER RESPONSES TO INTERROGATORIES SUBMITTED BY BARBARA STAMIRIS ON AUGUST 30, 1982"in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk through deposit in the Nuclear Regulatory Commission's internal mail system, this 14th day January 1983:

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Attachment I

TABLE C-20

DECOMMISSIONING COST ESTIMATE

Study Name: Big Rock Point
Reference No.: 6
Size: 63 MWe Type: BWR
Utility/Agency Name: Consumers Power Company
Decommissioning Alternative: Prompt Removal/Dismantle

		Cost in Millions of 1978 Dollars
Mobilize, Demobilize, Temporary Facilities		1.12
Supplies, Power, Contractors, Nuclear Insurance		3.21
Equipment		2.46
Staff Labor		7.78
Demolition Services		5.04
Disposal		2.20
Overheads		3.28
	Subtotal	25.09
	Contingency	6.27
	Total	31.36

TABLE B-30

DECOMMISSIONING COST ESTIMATE

Study Name: Palisades
Reference No.: 10
Size: 740 MWe Type: PWR
Utility/Agency Name: Consumers Power Company
Decommissioning Alternative: Immediate Dismantling

		Cost in Millions of 1978 Dollars
Mobilize/Facilities		1.304
Power/Supplies/Contractor Services/Insurance		6.400
Equipment		1.464
Staff Labor		8.984
Demolition Services		13.408
Disposal		12.616
Overheads		4.416
	Subtotal	48.592
	Contingency	12.148
	Total	60.740

Attachment 2

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

June 1974

REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

A. INTRODUCTION

Section 50.51, "Duration of license, renewal," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each license to operate a production and utilization facility be issued for a specified duration. Upon expiration of the specified period, the license may be either renewed or terminated by the Commission. Section 50.82, "Applications for termination of licenses," specifies the requirements that must be satisfied to terminate an operating license, including the requirement that the dismantlement of the facility and disposal of the component parts not be inimical to the common defense and security or to the health and safety of the public. This guide describes methods and procedures considered acceptable by the Regulatory staff for the termination of operating licenses for nuclear reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

When a licensee decides to terminate his nuclear reactor operating license, he may, as a first step in the process, request that his operating license be amended to restrict him to possess but not operate the facility. The advantage to the licensee of converting to such a possession-only license is reduced surveillance requirements in that periodic surveillance of equipment important to the safety of reactor operation is no longer required. Once this possession-only license is issued, reactor operation is not permitted. Other activities related to cessation of operations such as unloading fuel from the reactor and placing it in storage (either onsite of offsite) may be continued.

A licensee having a possession-only license must retain, with the Part 50 license, authorization for special nuclear material (10 CFR Part 70, "Special Nuclear Material"), byproduct material (10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material"), and source material (10 CFR Part 40, "Licensing of Source Material"). until the fuel, radioactive components, and sources are removed from the facility. Appropriate administrative controls and facility requirements are imposed by the Part 50 license and the technical specifications to assure that proper surveillance is performed and that the reactor facility is maintained in a safe condition and not operated.

A possession-only license permits various options and procedures for decommissioning, such as mothballing, entombment, or dismantling. The requirements imposed depend on the option selected.

Section 50.82 provides that the licensee may dismantle and dispose of the component parts of a nuclear reactor in accordance with existing regulations. For research reactors and critical facilities, this has usually meant the disassembly of a reactor and its shipment offsite, sometimes to another appropriately licensed organization for further use. The site from which a reactor has been removed must be decontaminated, as necessary, and inspected by the Commission to determine whether unrestricted access can be approved. In the case of nuclear power reactors, dismantling has usually been accomplished by shipping fuel offsite, making the reactor inoperable, and disposing of some of the radioactive components.

Radioactive components may be either shipped offsite for burial at an authorized burial ground or secured

on the site. Those radioactive materials remaining on the site must be isolated from the public by physical barriers or other means to prevent public access to hazardous levels of radiation. Surveillance is necessary to assure the long term integrity of the barriers. The amount of surveillance required depends upon (1) the potential hazard to the health and safety of the public from radioactive material remaining on the site and (2) the integrity of the physical barriers. Before areas may be released for unrestricted use, they must have been decontaminated or the radioactivity must have decayed to less than prescribed limits (Table I).

The hazard associated with the retired facility is evaluated by considering the amount and type of remaining contamination, the degree of confinement of the remaining radioactive materials, the physical security provided by the confinement, the susceptibility to release of radiation as a result of natural phenomena, and the duration of required surveillance.

C. REGULATORY POSITION

1. APPLICATION FOR A LICENSE TO POSSESS BUT NOT OPERATE (POSSESSION-ONLY LICENSE)

A request to amend an operating license to a possession-only license should be made to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545. The request should include the following information:

a. A description of the current status of the facility.

b. A description of measures that will be taken to prevent criticality or reactivity changes and to minimize releases of radioactivity from the facility.

c. Any proposed changes to the technical specifications that reflect the possession-only facility status and the necessary disassembly/retirement activities to be performed.

d. A safety analysis of both the activities to be accomplished and the proposed changes to the technical specifications.

e. An inventory of activated materials and their location in the facility.

2. ALTERNATIVES FOR REACTOR RETIREMENT

Four alternatives for retirement of nuclear reactor facilities are considered acceptable by the Regulatory staff. These are:

a. Mothballing. Mothballing of a nuclear reactor facility consists of putting the facility in a state of protective storage. In general, the facility may be left intact except that all fuel assemblies and the radioactive fluids and waste should be removed from the site. Adequate radiation monitoring, environmental surveillance, and appropriate security procedures should be established under a possession-only license to ensure that the health and safety of the public is not endangered.

b. In-Place Entombment. In-place entombment consists of sealing all the remaining highly radioactive or contaminated components (e.g., the pressure vessel and reactor internals) within a structure integral with the biological shield after having all fuel assemblies, radioactive fluids and wastes, and certain selected components shipped offsite. The structure should provide integrity over the period of time in which significant quantities (greater than Table I levels) of radioactivity remain with the material in the entombment. An appropriate and continuing surveillance program should be established under a possession-only license.

c. Removal of Radioactive Components and Dismantling. All fuel assemblies, radioactive fluids and waste, and other materials having activities above accepted unrestricted activity levels (Table I) should be removed from the site. The facility owner may then have unrestricted use of the site with no requirement for a license. If the facility owner so desires, the remainder of the reactor facility may be dismantled and all vestiges removed and disposed of.

d. Conversion to a New Nuclear System or a Fossil Fuel System. This alternative, which applies only to nuclear power plants, utilizes the existing turbine system with a new steam supply system. The original nuclear steam supply system should be separated from the electric generating system and disposed of in accordance with one of the previous three retirement alternatives.

3. SURVEILLANCE AND SECURITY FOR THE RETIREMENT ALTERNATIVES WHOSE FINAL STATUS REQUIRES A POSSESSION-ONLY LICENSE

A facility which has been licensed under a possession-only license may contain a significant amount of radioactivity in the form of activated and contaminated hardware and structural materials. Surveillance and commensurate security should be provided to assure that the public health and safety are not endangered.

a. Physical security to prevent inadvertent exposure of personnel should be provided by multiple locked barriers. The presence of these barriers should make it extremely difficult for an unauthorized person to gain access to areas where radiation or contamination levels exceed those specified in Regulatory Position C.4. To prevent inadvertent exposure, radiation areas above 5 mR/hr, such as near the activated primary system of a power plant, should be appropriately marked and should not be accessible except by cutting of welded closures or the disassembly and removal of substantial structures

and/or shielding material. Means such as a remote-readout intrusion alarm system should be provided to indicate to designated personnel when a physical barrier is penetrated. Security personnel that provide access control to the facility may be used instead of the physical barriers and the intrusion alarm systems.

b. The physical barriers to unauthorized entrance into the facility, e.g., fences, buildings, welded doors, and access openings, should be inspected at least quarterly to assure that these barriers have not deteriorated and that locks and locking apparatus are intact.

c. A facility radiation survey should be performed at least quarterly to verify that no radioactive material is escaping or being transported through the containment barriers in the facility. Sampling should be done along the most probable path by which radioactive material such as that stored in the inner containment regions could be transported to the outer regions of the facility and ultimately to the environs.

d. An environmental radiation survey should be performed at least semiannually to verify that no signficant amounts of radiation have been released to the environment from the facility. Samples such as soil, vegetation, and water should be taken at locations for which statistical data has been established during reactor operations.

e. A site representative should be designated to be responsible for controlling authorized access into and movement within the facility.

f. Administrative procedures should be established for the notification and reporting of abnormal occurrences such as (1) the entrance of an unauthorized person or persons into the facility and (2) a significant change in the radiation or contamination levels in the facility or the offsite environment.

g. The following reports should be made:

(1) An annual report to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, describing the results of the environmental and facility radiation surveys, the status of the facility, and an evaluation of the performance of security and surveillance measures.

(2) An abnormal occurrence report to the Regulatory Operations Regional Office by telephone within 24 hours of discovery of an abnormal occurrence. The abnormal occurrence will also be reported in the annual report described in the preceding item.

h. Records or logs relative to the following items should be kept and retained until the license is terminated, after which they may be stored with other plant records:

(1) Environmental surveys,

(2) Facility radiation surveys,

(3) Inspections of the physical barriers, and

(4) Abnormal occurrences.

4. DECONTAMINATION FOR RELEASE FOR UNRESTRICTED USE

If it is desired to terminate a license and to eliminate any further surveillance requirements, the facility should be sufficiently decontaminated to prevent risk to the public health and safety. After the decontamination is satisfactorily accomplished and the site inspected by the Commission, the Commission may authorize the license to be terminated and the facility abandoned or released for unrestricted use. The licensee should perform the decontamination using the following guidelines:

a. The licensee should make a reasonable effort to eliminate residual contamination.

b. No covering should be applied to radioactive surfaces of equipment or structures by paint, plating, or other covering material until it is known that contamination levels (determined by a survey and documented) are below the limits specified in Table I. In addition, a reasonable effort should be made (and documented) to further minimize contamination prior to any such covering.

c. The radioactivity of the interior surfaces of pipes, drain lines, or ductwork should be determined by making measurements at all traps and other appropriate access points, provided contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement should be assumed to be contaminated in excess of the permissable radiation limits.

d. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated in excess of the limits specified. This may include, but is not limited to, special circumstances such as the transfer of premises to another licensed organization that will continue to work with radioactive materials. Requests for such authorization should provide:

(1) Detailed, specific information describing the premises, equipment, scrap, and radioactive contaminants and the nature, extent, and degree of residual surface contamination.

(2) A detailed health and safety analysis indicating that the residual amounts of materials on surface areas, together with other considerations such as the prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

e. Prior to release of the premises for unrestricted use, the licensee should make a comprehensive radiation survey establishing that contamination is within the limits specified in Table I. A survey report should be filed with the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, with a copy to the Director of the Regulatory Operations Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report should:

(1) Identify the premises;

(2) Show that reasonable effort has been made to reduce residual contamination to as low as practicable levels;

(3) Describe the scope of the survey and the general procedures followed; and

(4) State the finding of the survey in units specified in Table 1.

After review of the report, the Commission may inspect the facilities to confirm the survey prior to granting approval for abandonment.

5. REACTOR RETIREMENT PROCEDURES

As indicated in Regulatory Position C.2 several alternatives are acceptable for reactor facility retirement. If minor disassembly or "mothballing" is planned, this could be done by the existing operating and maintenance procedures under the license in effect. Any planned actions involving an unreviewed safety question or a change in the technical specifications should be reviewed and approved in accordance with the requirements of 10 CFR §50.59.

If major structural changes to radioactive components of the facility are planned, such as removal of the pressure vessel or major components of the primary system, a dismantlement plan including the information required by §50.82 should be submitted to the Commission. A dismantlement plan should be submitted for all the alternatives of Regulatory Position C.2 except mothballing. However, minor disassembly activities may still be performed in the absence of such a plan, provided they are permitted by existing operating and maintenance procedures. A dismantlement plan should include the following:

a. A description of the ultimate status of the facility

b. A description of the dismantling activities and the precautions to be taken.

c. A safety analysis of the dismantling activities including any effluents which may be released.

d. A safety analysis of the facility in its ultimate status.

Upon satisfactory review and approval of the dismantling plan, a dismantling order is issued by the Commission in accordance with §50.82. When dismantling is completed and the Commission has been notified by letter, the appropriate Regulatory Operations Regional Office inspects the facility and verifies completion in accordance with the dismantlement plan. If residual radiation levels do not exceed the values in Table I, the Commission may terminate the license. If these levels are exceeded, the licensee retains the possession-only license under which the dismantling activities have been conducted or, as an alternative, may make application to the State (if an Agreement State) for a byproduct materials license.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

ELK RIVER REACTOR DISMANTLING

Attachment 3

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ABSTRACT

The Elk River Reactor (ERR) was constructed by the AEC on land leased from the United Power Association (UPA). Upon expiration of the operating term of the contract, UPA waived its option to purchase the reactor. Pursuant to the terms of the contract and lease, the AEC had the obligation to make the site usable without danger to the public health and safety. Final agreement specified that the ERR would be dismantled and removed with the exception of the underground portion of the Reactor Building. The underground portion was to be left in place and any resulting cavities filled with earth and rubble. The dismantling program was carried out in three overlapping phases as follows:

1. The Planning Phase which included the preliminary planning and selection of the dismantling approach.
2. The Dismantling Phase which included all work performed to remove the reactor facility and restore the site to its pre-reactor condition, and
3. The Closeout Phase which included the final site survey and efforts necessary to terminate the AEC license and contract.

Of particular interest was the use of a remotely operated plasma cutting torch to section the pressure vessel internals, the pressure vessel and the outer thermal shield, the use of explosives in removal of the biological shield and the method of establishment of the criteria for material disposal.

I. INTRODUCTION

The Elk River Reactor (ERR), a 58 MW(th), indirect-cycle, natural circulation, boiling water reactor was constructed by Allis-Chalmers under an Atomic Energy Commission (AEC) contract and was operated by the United Power Association (UPA) of Elk River, Minnesota, under contract to the Commission. The reactor was constructed adjacent to an existing conventional steam electrical plant on land owned by UPA and leased to the AEC (Figure 1).

Initial reactor criticality was achieved on November 19, 1962, with power operation commencing on July 13, 1964. The plant was commercially operated by UPA until final shutdown on January 31, 1968, resulting in a total of 53,000 MWD_{th} of power.

Upon expiration of the operating term of the contract, UPA waived its option to purchase the plant. Pursuant to the terms of the contract and lease, the AEC had the obligation to make the site usable without danger to the public health and safety. The final agreement between UPA and the AEC specified that the ERR would be dismantled and removed approximately to grade with the exception of the underground portion of the reactor building which would be completely removed and that the resulting cavities would be filled with earth and rubble.

Physical dismantling began in June 1972 and was completed in July 1974. Upon completion of the dismantling operation, the site was returned to

approximately the condition which existed prior to the installation of the reactor with all vestiges of the reactor plant having been removed (except for portions of subgrade foundations). The final radiation survey of the site produced no evidence of radiation levels in excess of normal background. No license or other authorization is required and no continuation of radiological monitoring is necessary.

II. GENERAL SUMMARY

The ERR dismantling program was carried out in three overlapping phases which can be identified as follows:

A. Planning Phase - This included the preliminary planning and the selection of the dismantling approach with analyses of feasibility and cost. Scoping documents describing the general dismantling sequence were prepared including, the Elk River Reactor Dismantling Environmental Statement, the Elk River Reactor Dismantling Plan, eleven Activity Specifications, and twenty-one Detailed Working Procedures.

B. Dismantling Phase - This phase included all work performed to remove the ERR facility and restore the site. Minor revisions were made to the methods developed in the planning phase to accommodate existing field conditions.

C. Facility Closeout Phase - This phase included those activities necessary to terminate the "Possession Only" authorization and the AEC contract. These activities included preparation of a Final Radiation Survey Report and Final Program Report.

The responsibilities of the major participants in the dismantling program are outlined below:

A. United Power Association Responsibility

UPA performed the authorized dismantling activities retaining full responsibility for all of the work including safety. UPA identified and retained approval controls over those portions of the AEC Dismantling Plan and Activity Specifications which might have endangered the integrity of UPA's steam electric generating facilities.

To provide assurance that UPA management was carrying out its responsibilities for the protection of the public, plant personnel and property from any attendant hazards during these operations, a Safety Review Committee was appointed by UPA and acted as an independent advisory group to UPA management, and as such, provided independent safety review of activities during the ERR dismantling program.

B. Gulf United Nuclear Fuels Corporation (GU) Responsibility

GU prepared and recommended an AEC Dismantling Plan and Activity Specifications for review and comment by all participants. It also provided UPA and the AEC, as requested, engineering services, safety committee participation and other services and support for the dismantling both on and off-site.

C. Atomic Energy Commission Responsibility

1. Chicago Operations Office (CH)

    a. CH was responsible for all contract activities including safety and the management of the dismantling including the administration of the UPA and GU contracts. CH provided official notice to UPA and GU of AEC approvals for the Dismantling Plan, Activity Specifications, safety, con-

struction and subcontracts and other actions requiring AEC approval.

b. The CH-ERR Site Representative (AEC Representative) provided UPA and GU with official AEC approval of Detailed Working Procedures and such other actions under a) above which were specifically delegated to him, monitored the work, and assisted participants in the dismantling effort.*

2. Division of Waste Management and Transportation (WMT)*

WMT provided programmatic and technical responsibility for the work.

III. PROGRAM PLANNING

For planning purposes the dismantling was divided into eleven major tasks as follows:

A. Facility and site preparation.

B. Removal of equipment and systems exterior to the biological shield not required to support other dismantling activities.

C. Removal of the coal-fired superheater and associated systems.

D. Removal of the superheater and a portion of the service building connecting the reactor building with the UPA Steam Plant facility.

E. Removal of the pressure vessel internals.

F. Removal of the pressure vessel and segmentation of the outer thermal shield.

G. Removal of the outer thermal shield and all radioactive or contaminated structures.

* Originally, these activities were the responsibility of the Division of Reactor Development and Technology.

H. Removal of the reactor building and all remaining non-contaminated equipment contained therein.

I. Removal of miscellaneous equipment.

J. Disposal of all government-owned property.

K. Program closeout including final radiation survey and program reports.

The documentation required for each major task consisted of an Activity Specification which provided a general scope including sequencing, health and safety considerations, and a conceptual design for special tooling. Each Activity Specification was supported by engineering studies to scope or justify the activity and contained technical guidelines for the preparation of Detailed Working Procedures. The Detailed Working Procedures were prepared immediately prior to the implementation of a particular task and included a step-by-step removal approach, tooling requirements, and a discussion of radiological and industrial safety requirements.

The general dismantling sequence evolving from the planning may be divided into three distinct but overlapping stages: 1) Removal of the most highly radioactive components including the reactor internals, pressure vessel, etc; 2) Removal of plant systems and equipment outside the biological shield containing low levels of radioactive contamination; and 3) Removal of non-contaminated structures.

The second and third stages were in general accomplished using conventional demolition techniques and will not be addressed further. Only those activities concerned with the removal, packaging, and disposal of the most

highly radioactive components (Tasks E and F noted above) and other contaminated structures (Task G above) will be discussed.

IV. Removal and Disposal of Pressure Vessel Internals, Pressure Vessel and Segmentation of the Outer Thermal Shield

The removal of reactor internals (shown schematically in Figure 2) involved handling highly radioactive components. The calculated inventory of radioactivity and the contact radiation levels of these components are given in Figure 3.

A. Equipment

With the exception of the equipment used to segment the inner thermal shield, pressure vessel and outer thermal shield, the equipment used was typical of long-handled special purpose equipment used at most reactor installations.

The equipment used to remotely segment the inner thermal shield, pressure vessel and outer thermal shield consisted of a system of inter-related parts which can be described as follows:

1. A cutting torch system which consisted of all the usual equipment required for operating a commercial plasma or oxy-acetylene torch.
2. A manipulator which carried the torch through the required range of travel, provided structural strength for torch rigidity, and provided accurate control of torch speed and torch-to-work distance.
3. A system which provided remote control of the manipulator and the cutting process.

4. Segment handling tools.

5. Other miscellaneous equipment including a contamination control system, manipulator storage stand, support platform, and lower radial bearing.

The testing and development work required to bring the plasma and oxy-acetylene cutting process to the desired level of readiness involved three different testing programs: a) bench scale tests, b) full scale mockup tests of the complete manipulator and plasma torch system, and c) separate tests of the in-air cutting procedures.

The data obtained from this testing program established all of the parameters for both underwater and in-air plasma arc cutting including voltage, amperage, torch-to-work distance, nozzle size, gas flow, travel speed, and starting conditions.

A preliminary testing program for segmentation of the outer thermal shield indicated that the plasma torch technique was not feasible for this operation since cutting was required in areas containing lead. Therefore, an additional development effort was conducted which resulted in an oxy-acetylene torch system which could be mounted on the original manipulator.

B. Removal

The upper stainless steel control rod guide and shroud assembly was the first internal component to be removed. It was mechanically attached to the lower shroud assembly with tackwelded machine screws.

The upper shroud was separated from the lower shroud by use of a hydraulic bolt shearing tool. After detachment, the shroud was transferred in-air from the reactor vessel to the fuel element storage well (FESW) and placed on an underwater cutting platform where a commercially available sheet metal nibbler, modified for both underwater and remote operation was used to segment the shroud into suitable sizes for loading into two shielded cask liners (Figure 4).

The lower zircoloy control rod guide and shroud assembly was separated from the shroud support plate by shearing 36 stainless steel cap screws. After detachment, the lower shroud was transferred in-air to the FESW where it was segmented and packaged in the same manner as the upper shroud.

Detaching the shroud and core support plate from the support stand was relatively simple as it only required the removal of four captive bolts using a long-handled socket tool. After detachment, the plate was lifted, moved through the fuel transfer canal to the FESW and loaded directly into a shielded cask liner.

The core support stand was attached to the pressure vessel with four captive bolts which were removed using a long-handled socket tool. The stand was then transferred in-air to the FESW for underwater loading into a shielded cask liner.

The feedwater distribution ring and four shadow shields were fastened to the inner thermal shield with bolts which were removed using an air-operated impact wrench. The shadow shields were loaded underwater directly into a shielded cask liner located in the bottom of the pressure vessel. The feedwater distribution ring was removed in four segments and transferred in-air to a cask liner located in the FESW.

A commercially available hand-held plasma arc cutting torch was used to segment the stainless steel steam baffle while a remotely operated plasma torch was used to segment the inner thermal shield (Figure 5). This latter component was a 1-inch thick, type 304 stainless steel cylinder approximately 81-inches I.D. and 12-feet high. All cutting was done underwater within a containment enclosure formed by the side walls of the reactor cavity. All penetrations into the cavity were sealed and a metal cover called a contamination control envelope was placed over the cavity. The containment enclosure had a separate exhaust fan and filtering system containing a prefilter and a high efficiency particulate filter capable of removing most of the airborne contamination generated during cutting operations. After segmentation of the inner thermal shield, the contamination control envelope was removed and the segments were transferred to the FESW for underwater loading into a cask liner.

The next major task was removal of the pressure vessel. The vessel was an 84-inch I.D. cylinder approximately 25-feet in height with

3-inch carbon steel walls and a 0.1-inch, type 304 stainless steel inner cladding. The pressure vessel was segmented using the plasma torch equipment described previously with modifications to the torch and torch arm. Cutting was performed in-air within the containment enclosure with the vessel water level maintained slightly below the cut elevation. The upper six-foot section of the vessel including the vessel flange was removed as one piece, (Figure 6), sealed with metal plates and shipped as its own container. The remairder of the vessel was cut into 19 segments. The segments were transferred in-air to cask liners either located on the main operating floor or in the FESW (Figure 7).

The outer thermal shield consisted of a 3-inch annulus of lead sandwiched between two 3/4-inch carbon steel plates and was approximately 15-1/2 feet in height and 8-1/2 feet in diameter. It had been fabricated in three 120° sections and assembled onsite utilizing 3-inch thick welded carbon steel joints.

Since the temperatures generated by the plasma torch would cause significant lead vaporization, it was necessary that an oxy-acetylene torch system which operated at a lower temperature be used for segmentation of the outer thermal shield. The torch system was mounted on the plasma torch manipulator and operated from a remote location outside the containment enclosure. The geometry of the cutting in this case was more complex than for either the inner thermal shield or the pressure vessel. Approximately 300 cuts were required of six

different types; namely, (1) 3/4-inch plate with lead backing, (2) lead melting, (3) stainless steel cooling tubes, (4) 3/4-inch plate with concrete backing, (5) 5-1/2-inch carbon steel plate with concrete backing and (6) severing of stainless steel cooling coil penetrations. The cutting pattern resulted in six large segments which were transferred in-air to shipping containers (Figure 8) located on the basement floor of the reactor building.

V. Removal and Disposal of Radioactive and Contaminated Materials from the Biological Shield and Reactor Building

A. Description of Structures

An isometric view of the ERR biological shield (B/S) and fuel element storage well is shown in Figure 9. This structure was approximately 45-feet high with a maximum thickness of 9-feet and was composed of 3,500 psi (minimum) concrete reinforced with #9 rebar located at a depth of approximately 4-inches from all exposed surfaces on a 12-inch by 12-inch vertical and horizontal grid with additional reinforcement at various elevations.

The operating floors in the reactor building consisted of the sub-basement floor which contained the building sump and liquid waste processing equipment, the basement, the intermediate floor, and the main floor, each of which was reinforced with #6 and #9 rebars on 12-inch centers.

B. Removal Methods

The following criteria were considered in selecting a removal method for the concrete and associated steel structures:

1. Confinement of the reactor building could not be breached until all radioactive material had been removed or adequately contained.
2. The removal method had to lend itself to control of airborne and surface contamination.
3. The spread of contamination outside of the reactor cavity during removal of the interior portion of the cavity had to be minimized.
4. The removal method could not affect operation of the UPA fossil-fired electrical generation plant.
5. Radiation exposure to dismantling personnel had to be minimized.
6. The size and weight of removed materials had to be controlled to meet shipping and burial criteria.

Experience during the dismantling indicated that both flame cutting and thermal lance techniques were not feasible due to the generation of large quantities of toxic gases and smoke.

Drilling and rock-jacking were feasible for the removal of concrete walls up to a maximum thickness of 2-feet; however, the time required made these methods uneconomical. Rock-jacking is also not an effective method for the removal of thick, heavily reinforced concrete structures such as the B/S. Other methods including the use of a demolition ball and wall sawing were considered. In both cases, tests carried out indicated that these techniques were not feasible from a time or production standpoint.

The use of explosives for removal of the B/S and FESW was only con-

sidered after all other techniques had proved unsuccessful. The main concerns in using explosives were the control of airborne contamination and the assurance that vibrations generated by the blasting would not disrupt the operation of the turbine generators in the UPA Steam Plant. Consequently, a testing program which consisted of loading progressively larger charges in the upper portion of the B/S and measuring the vibrations and dust levels resulting from their detonation was made. The results of these tests indicated that a maximum charge weight per delay of 1-1/4 lbs. could be used without producing vibrations in excess of those normally seen during startup of the turbine generators and that airborne contamination could be controlled using blasting mats and a localized fog spray system.

C. Operations

All blasting operations were under the direct control of a qualified explosives engineer. In addition, a Detailed Working Procedure was written to assure administrative control of the blasting operations. This procedure provided a general removal sequence and specified the radiological and industrial safety criteria to be used during the blasting.

In general, two types of drill patterns were used. The first pattern consisted of 1/2 to 5-foot deep boreholes with burdens of 1-1/3 to 1-1/2 feet and spacings of 1-1/2 to 3 feet depending upon the reinforcing rod pattern and location. This pattern was then usually loaded with conventional cartride type dynamites.

The second type of drill-pattern involved boreholes 12 to 16 feet in depth with burdens of 8 to 18-inches and fairly wide spacings of 2-1/2 to 5-feet. The smaller burden and larger spacings were used to remove reinforcing rod mats located 4-inches from exposed surfaces. Removal of these rod mats provided good relief to the next row of production holes which were drilled using larger burdens and smaller spacings. Both vertical and horizontal holes were used and they were normally loaded with several strands of detonating cord.

The maximum blast fired inside the reactor building was 1-1/4 lbs. per delay on 20 delays.

No air blast pressure damage occurred during the blasting operations conducted within the sealed reactor building. Outside noise levels were minimal although some blasts could be faintly heard approximately 300 feet from the blast site.

All blasts were monitored using accelerometers mounted on the turbine generator pedestals in the UPA Steam Plant. In addition, a portable seismograph was used to monitor vibrations at various locations on the reactor site. During blasting, no vibrations were produced which exceeded the normal startup vibrations of the UPA Steam Plant turbine generators.

D. Removal Sequence

Radiation surveys performed prior to the beginning of major demolition work indicated that the following structures contained sufficient

radioactive contamination to require their removal under existing radiation procedures:

1. B/S and FESW
2. Main, intermediate, and basement floors
3. Subbasement floor (contaminated to a depth of approximately 6-inches)

Contamination also existed at isolated areas including portions of the reactor building inner concrete liner. The general sequence of demolition proceeded from the top of the B/S downward. Attempts were made to identify all radioactive or contaminated material and remove it as the demolition proceeded toward the subbasement.

The first structure to be removed using explosives was the portion of the B/S extending from elevation 947 ft. 6 in. to elevation 938 ft. 0 in. The majority of concrete in this area was noncontaminated; however, surface contamination resulting from reactor operations (fuel transfer, other dismantling activities), did exist at elevation 947 ft. 6 in. and on the inner and outer surfaces of the reactor cavity.

The next structure to be removed was the main floor at elevation 938 ft. 0 in. Various removal methods were considered including wall sawing techniques. However, it was found that the most efficient method for removal of the floor consisted of using a jackhammer to expose the rebar on a given floor area supported by a floor beam.

After the rebar was exposed and a crane attached to the floor slab, the ends of the support beam were cut using explosives. The rebar was then cut and the floor slab removed and prepared for shipment.

The next stage of demolition consisted of removal of the entire B/S and FESW structure from elevation 938 ft. 0 in. to elevation 923 ft. 0 in. Although radioactive contamination existing in this area was higher than in the upper portion of the B/S, the possibility of an airborne contamination problem was still minimal and therefore, the major concern during this phase was efficiently removing the material.

The next phase of B/S removal consisted of the removal of the relatively highly radioactive portion of the B/S extending from elevation 923 ft. 0 in. to 903 ft. 0 in. In this particular phase, contamination control was the prime consideration and therefore, all blasting operations proceeded from inside the reactor cavity outward (Figure 10) with an appropriate seal over the top of the cavity at elevation 923 ft. 0 in. Starting with the inner ring of reinforcing rods, progressively larger rings were removed. Excellent control of debris and dusts was exercised and no significant airborne contamination problems were encountered.

The final phase of work consisted of removal of the remaining B/S and FESW structure, the intermediate and basement floors and removal of approximately 6-inches of the subbasement floor (Figures 11 and 12).

In general, these activities were carried out using methods similar to those used for the previous work.

E. Material Disposal

For obvious economic reasons, it was desirable to dispose of as much demolition debris as possible in local landfills. Because there were no burial facilities for radioactive materials in the State of Minnesota, a lack of disposal standards for activated materials, and because of existing adverse public reaction to the nuclear industry from certain sectors, great pains were taken to insure that no radioactivity above background, remained in the structures that were disposed of in Minnesota. For these reasons, the term "detectable reactor originated radioactivity" or DROR was specified contractually and defined for this project. It should be emphasized that DROR as defined below is unique to the ERR project, it is a one-time requirement, and there is no intent to suggest a guideline for future decommissioning actions or to supercede guidelines issued by the Nuclear Regulatory Commission. The term DROR was applicable only to demolition rubble that was to be left in the State of Minnesota and was defined procedurally by a special sampling and analytical method. Briefly the sampling technique involved removal of at least 50 pounds of surface concrete in situ from a predetermined sampling area of approximately 200 ft^2 or less (Figure 13). If the concrete were in the form of rubble, sufficient 50 pound random samples were taken to represent 1% of the total weight of the rubble sampled. Samples were analyzed by gamma spectrometric analysis with a large volume (4-inch x 5-inch) NaI(Tl) crystal. Normal background radiation levels in concrete were

determined with a number of samples taken at the site but outside of the reactor building. For the two radionuclides most often observed in structural materials ^{137}Cs and ^{60}Co -- the minimum detectable activity at the two-sigma level was estimated to be 0.12 and 0.04 pCi/gm, respectively. If a sample of concrete did not contain these concentrations, the concrete was judged to be free of detectable reactor originated radioactivity and was disposed of at a sanitary landfill near the ERR site.

VI. Radiological Conditions and Personnel Exposure

In spite of the attention given during dismantling to remote or underwater operations, the use of temporary radiation shielding and other safeguards, significant radiation exposure was received by those engaged in dismantling activities. The greatest portion of this exposure was received during almost constant work in low level radiation fields, varying between 5 and 15 mR/hr.

During dismantling, a total of 75 rem of whole body penetrating exposure was received by approximately 100 people connected with the dismantling project. The average whole body exposure was about 0.8 rem while the maximum total exposure received by any workman was about 4.5 rem.

Dismantling operations required the expenditure of approximately 12 rem of personnel exposure for removal of the reactor internals, 45 rem for removal of the pressure vessel and outer thermal shield, 12 rem for removal of the biological shield, and 6 rem for removal of systems external to the biological shield.

Internal deposition of radioactive material in personnel was estimated periodically with a whole body counter. The radionuclides identified by whole body counter examination were exclusively ^{137}Cs and ^{60}Co. No internal deposition exceeded 1% of a maximum permissible body burden even though airborne levels of radioactivity were significant at times. This may be attributed to detailed preplanning, training, and the liberal use of various types of respiratory protective equipment.

VII. Program Costs

The total project cost (including $418,000 of technical support services provided by Gulf United Nuclear Fuels Corporation under a separate prime contract with AEC) was $6.148 million. A summary of costs by major tasks (exclusive of technical support services) is shown in Figure 14.

B I B L I O G R A P H Y

1. Final Elk River Reactor Program Report, COO-651-93, September 1974, Revised November 1974.

2. Final Report of the Safety Review Committee for Decommission and Dismantling of the United Power Association's Elk River Reactor, dated July 26, 1974.

3. Final Elk River Reactor Site Survey, Results and Summary, COO-651-92, July 23, 1974

FIGURES

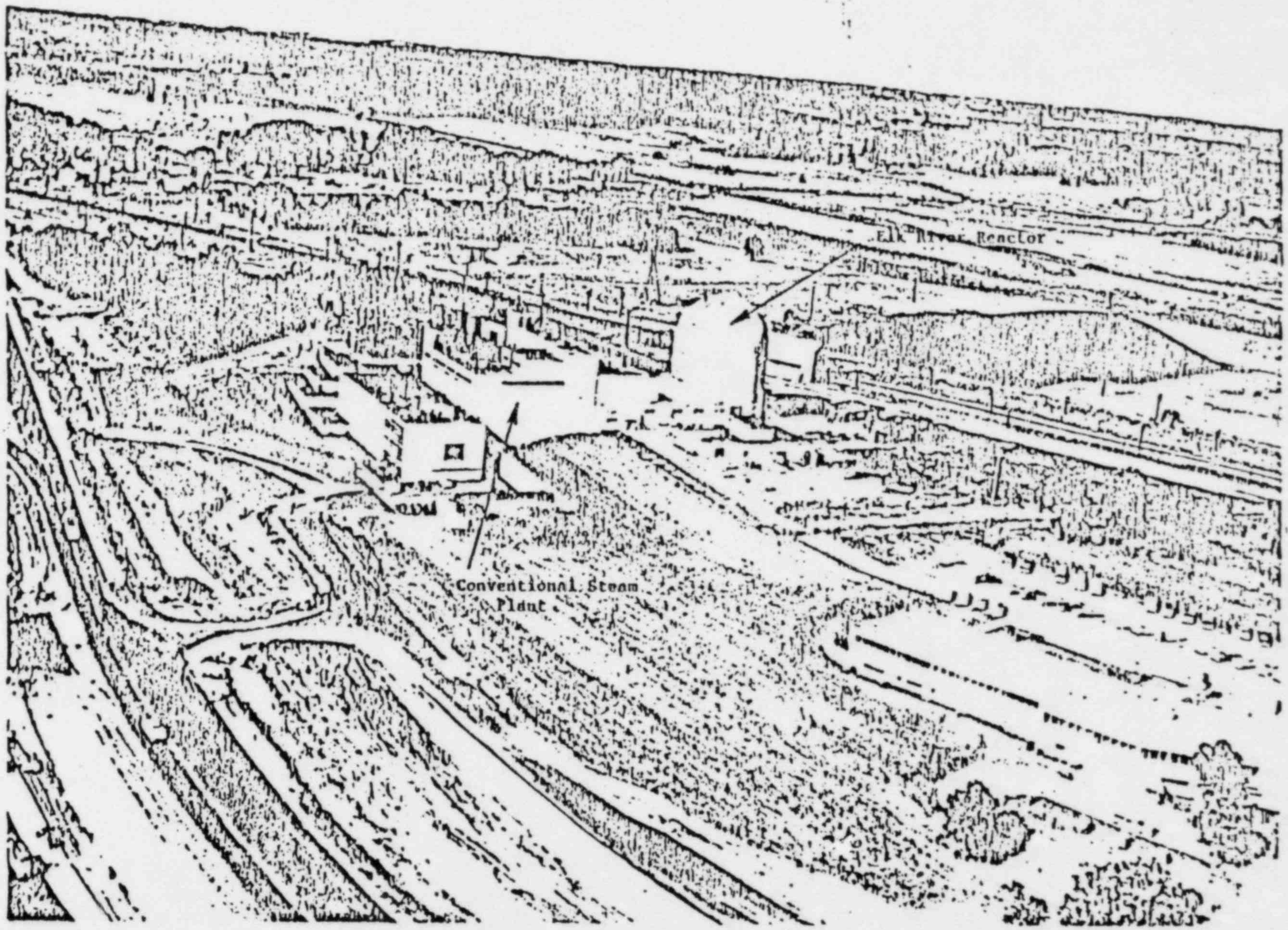
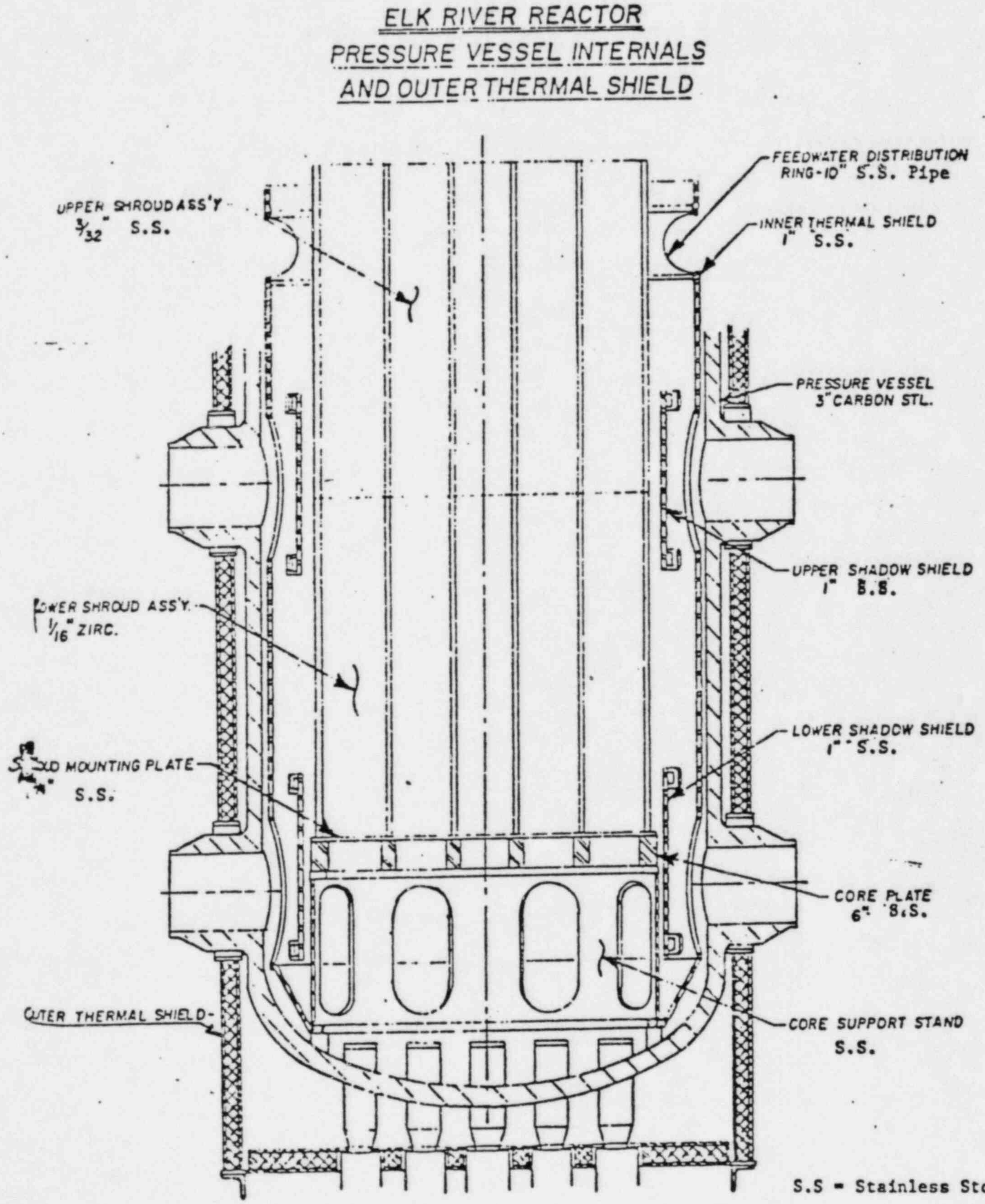


Figure 1
Elk River Reactor Plant



S.S - Stainless Steel

Figure 3

REACTOR INTERNALS RADIATION LEVELS AND RADIOACTIVE INVENTORY

COMPONENT	MAXIMUM CONTACT RADIATION LEVEL, R/hr	INVENTORY (CURIES)
S. S. SHROUD	2800	770
ZIRCALOY SHROUD	175	35
CORE & SHROUD PLATES	8000	2370
CORE SUPPORT STAND	150	100
INNER THERMAL SHIELD	1000	3090
SHADOW SHIELDS	3000	2330
FEEDWATER DISTRIBUTION RING	60	75

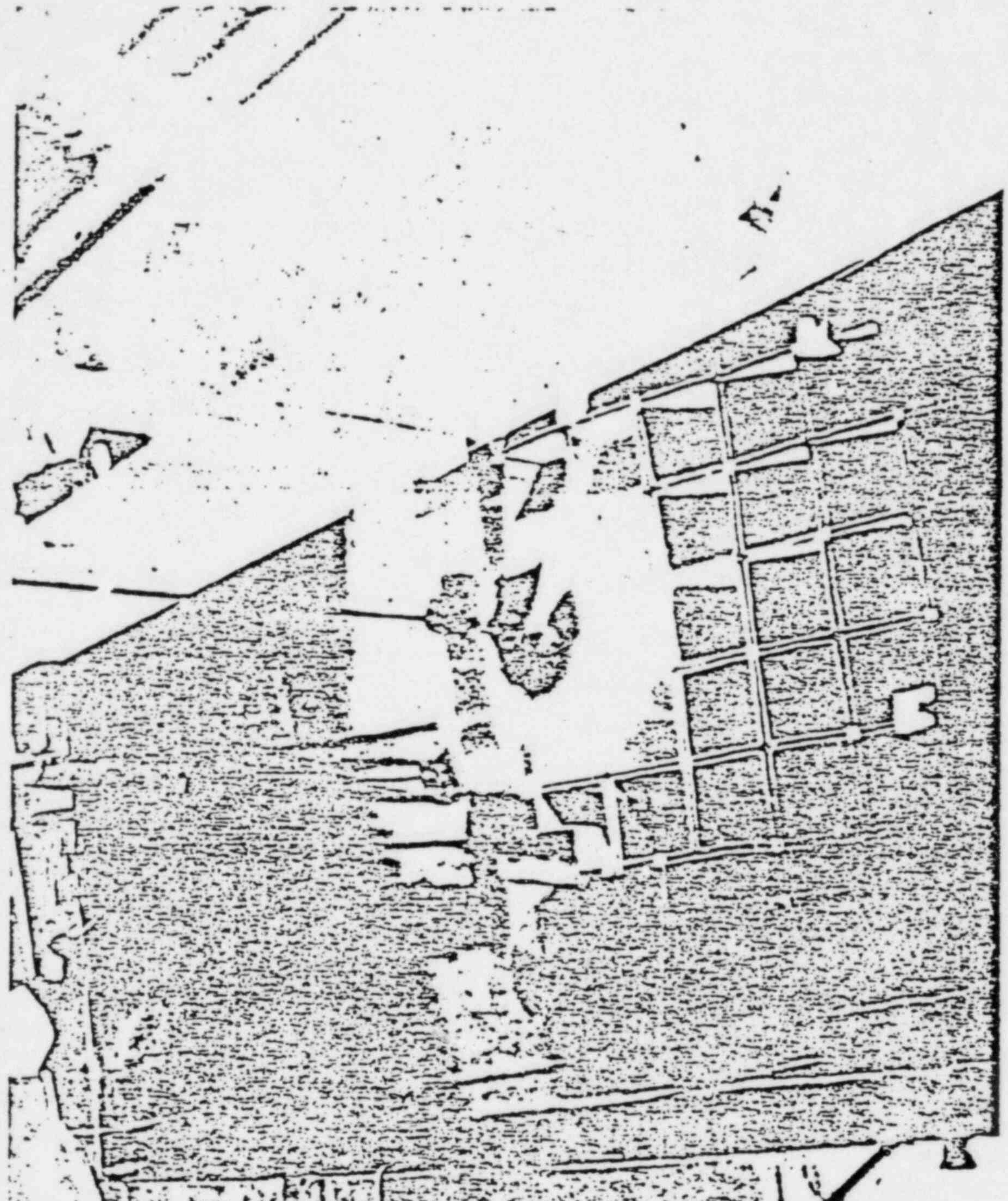


Figure 4
Segmentation of Shroud

Figure 5
Installation of Plasma Torch Manipulator

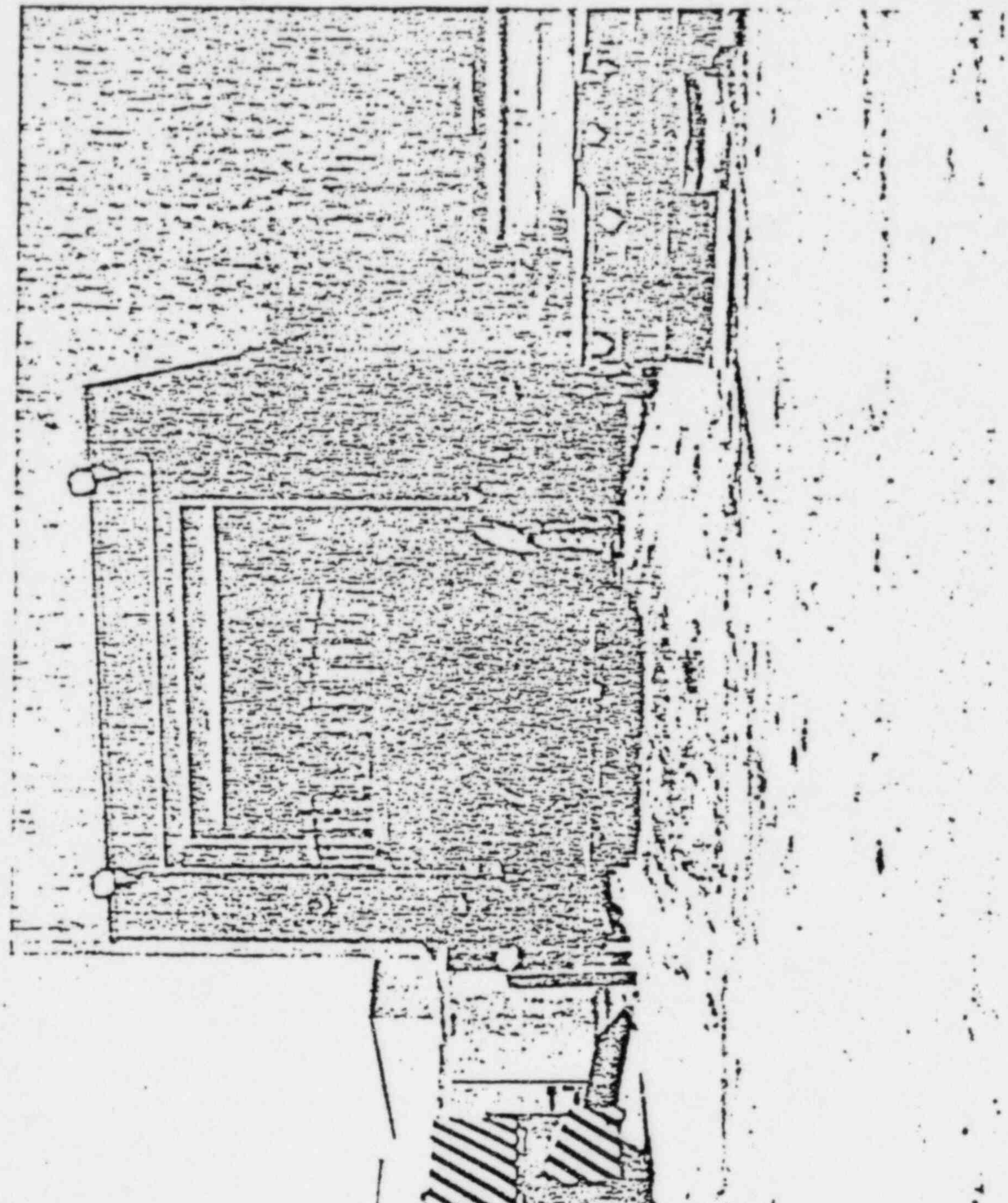


Figure 6

Upper Pressure Vessel Section

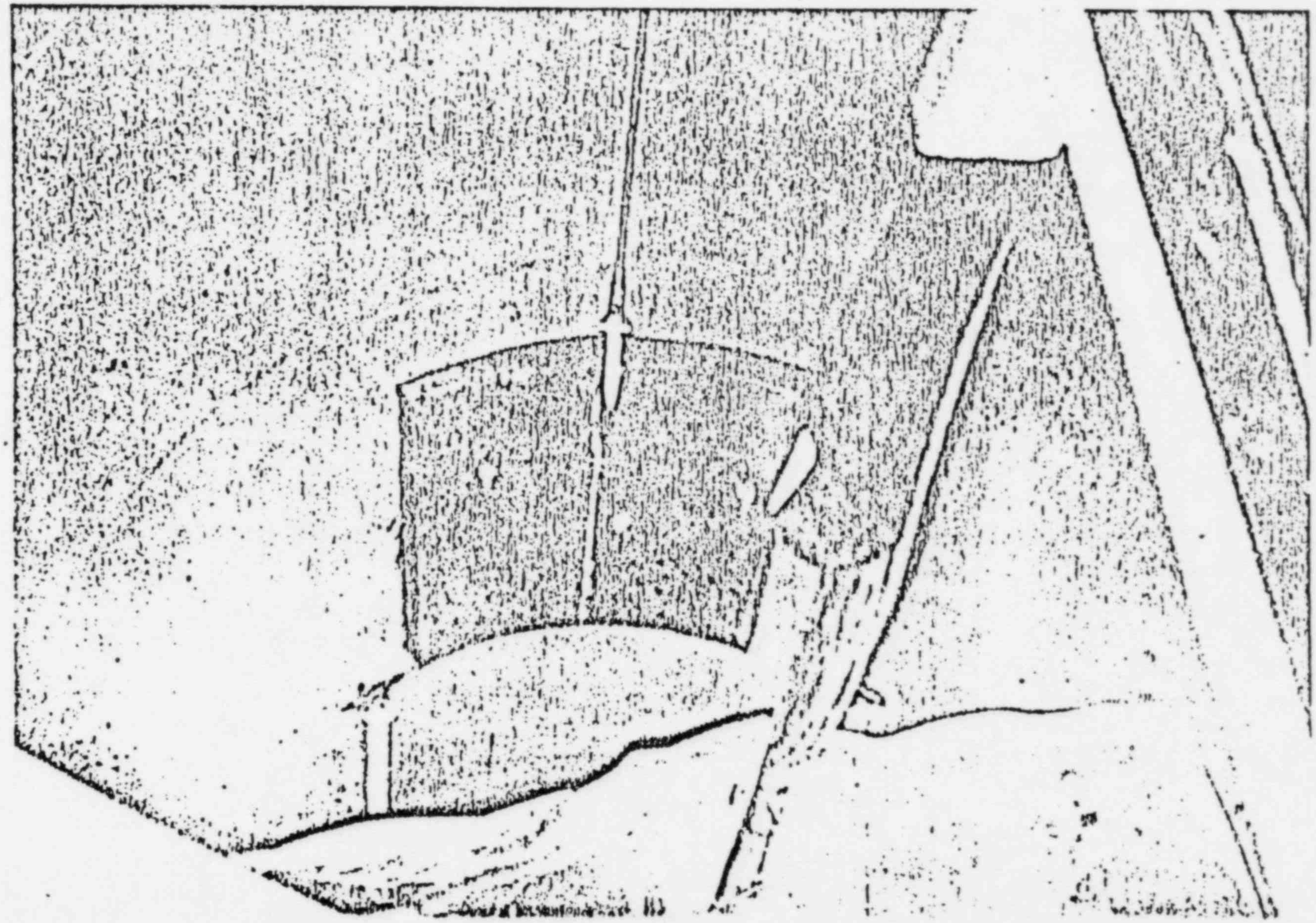


Figure 7

Pressure Vessel Segment Transfer

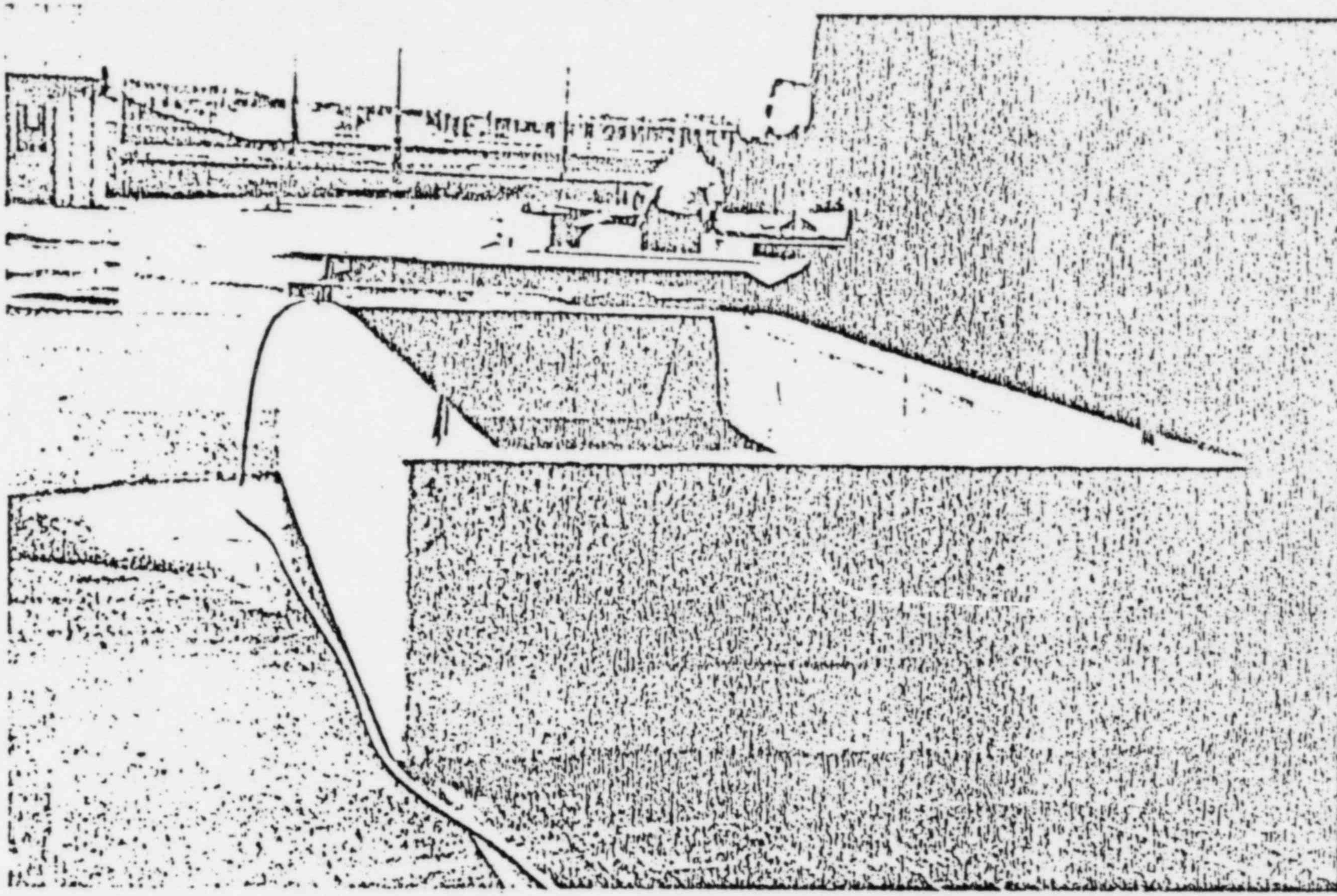
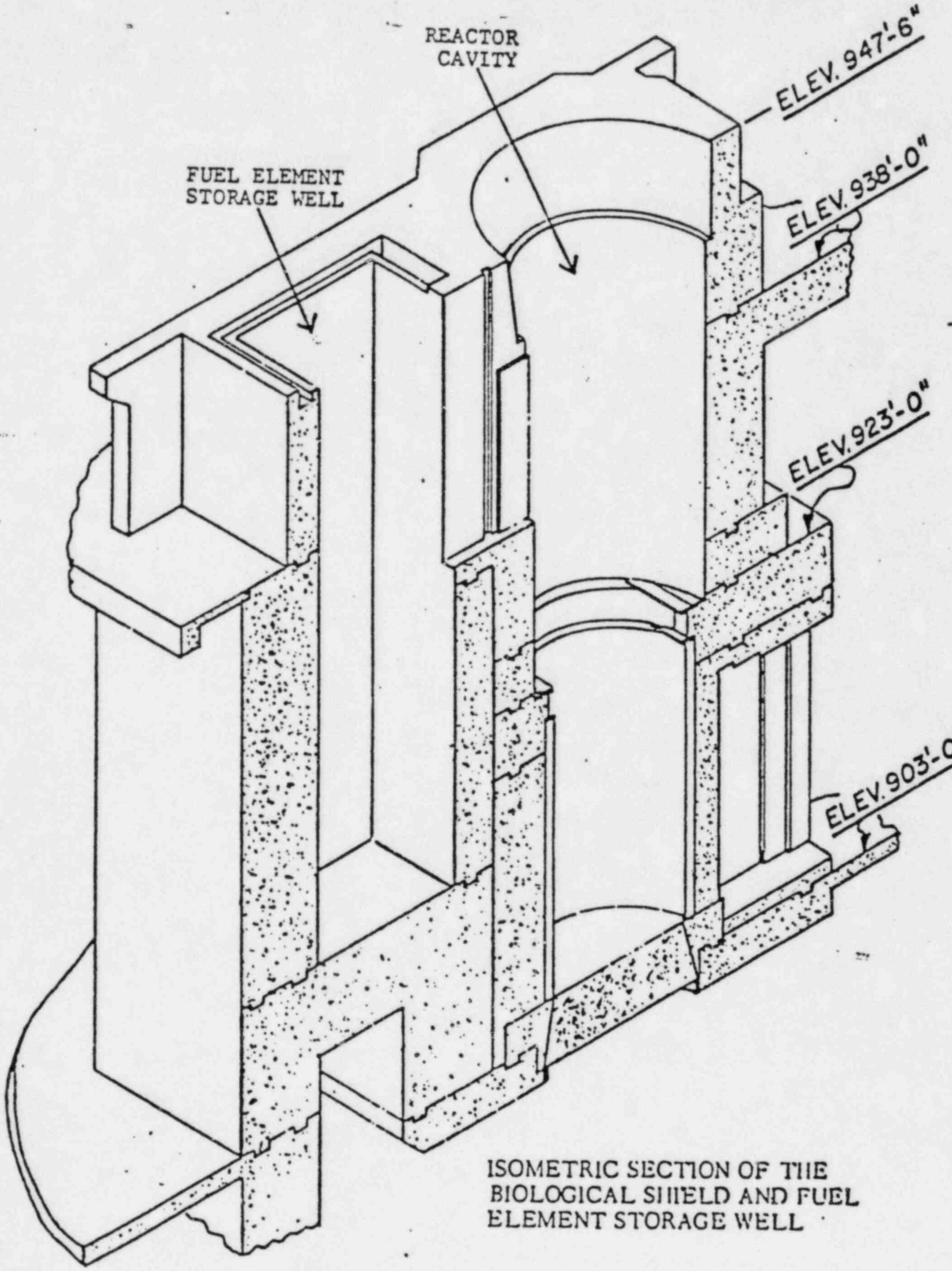


Figure 8

Outer Thermal Shield Concrete Shipping Vault

FIGURE 9



ISOMETRIC SECTION OF THE BIOLOGICAL SHIELD AND FUEL ELEMENT STORAGE WELL

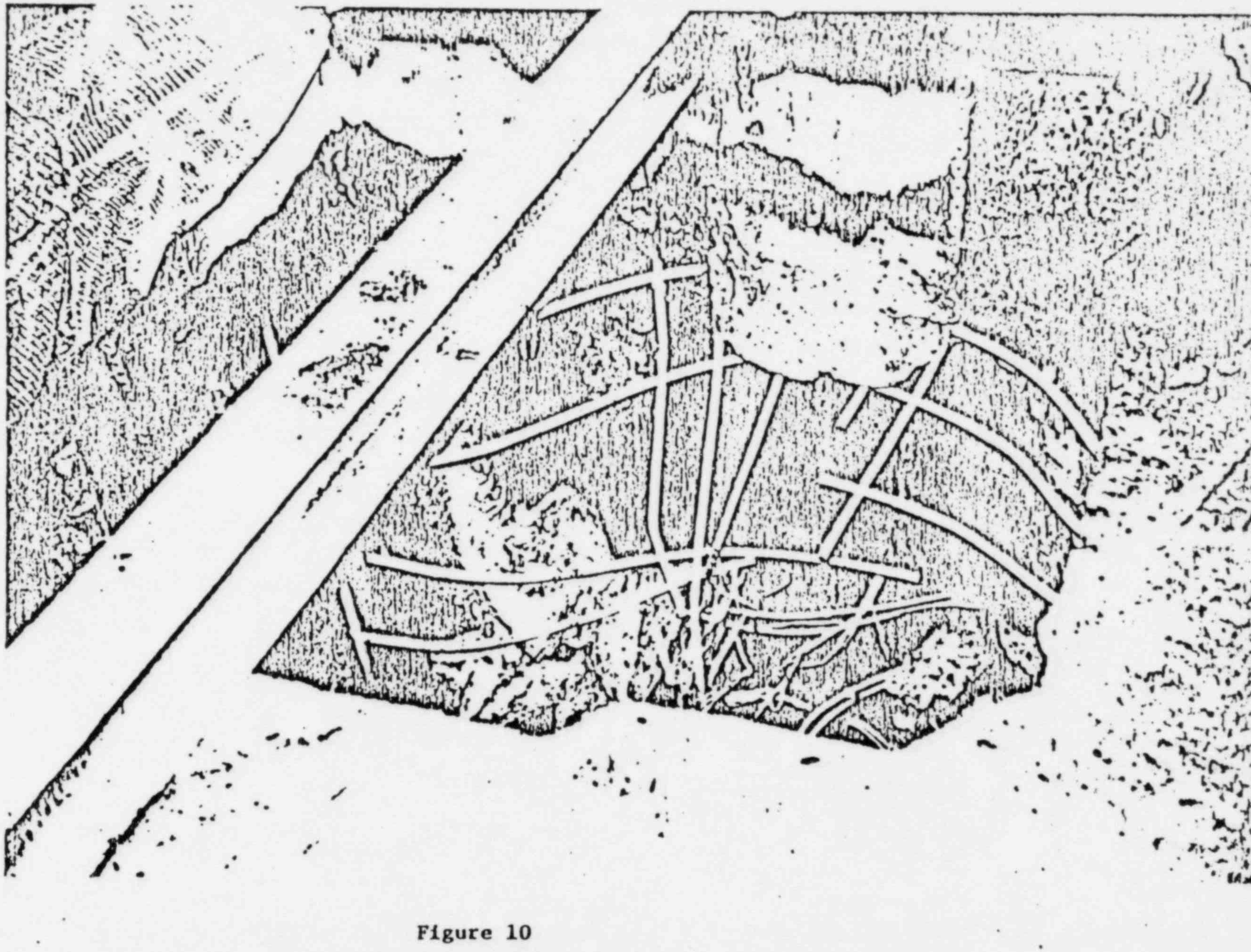


Figure 10

Explosive Removal of Biological Shield Concrete

Figure 11
Removal of Subbasement Walls

Figure 12

Explosive Scarfing of Subbasement Floor and Walls

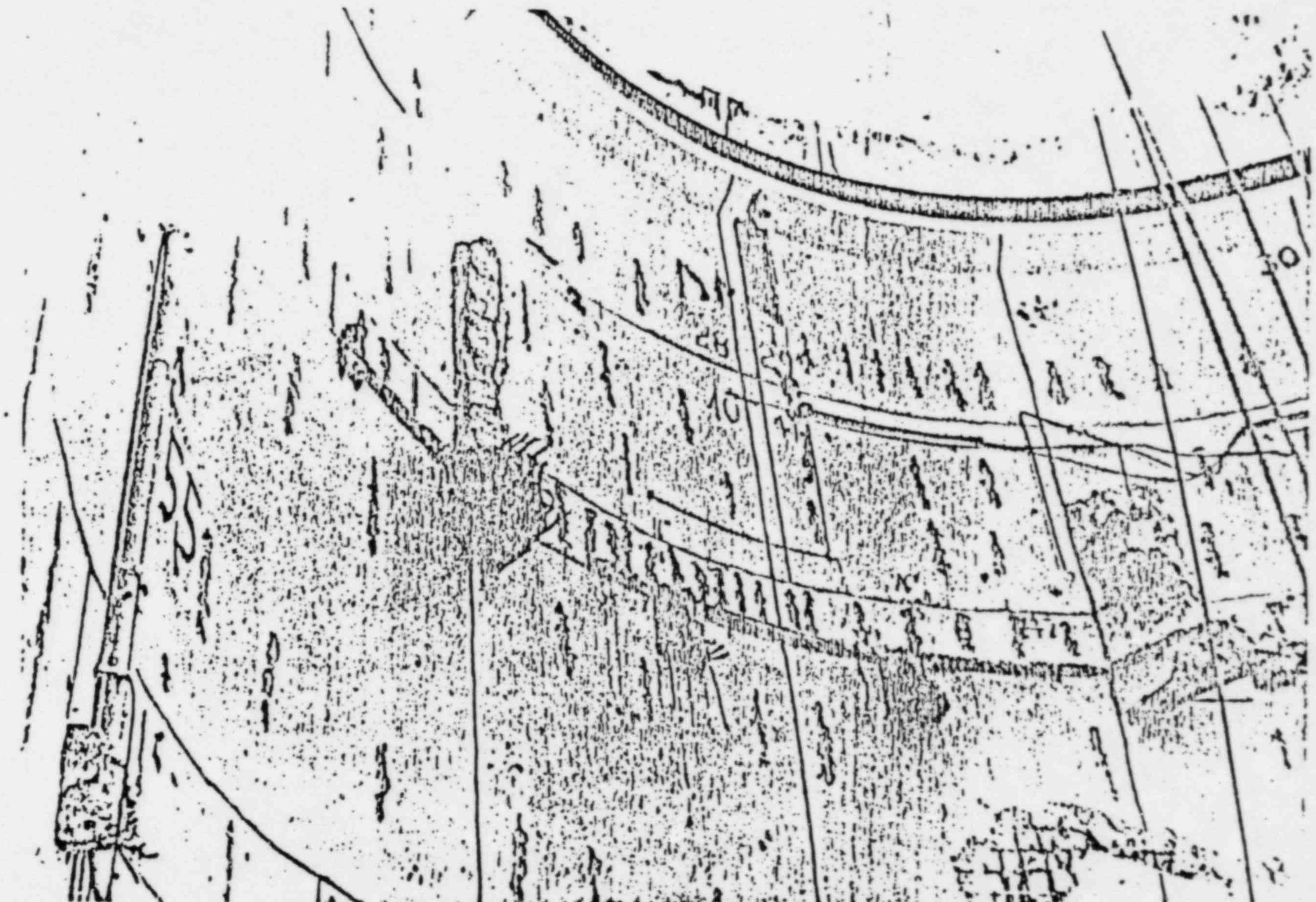


Figure 13

Wall Sampling for Detectable
Reactor Originated Radioactivity

Figure 14

PROJECT COSTS

	ACTIVITY SPECIFICATION NO.	ACTUAL OR ESTIMATED ACTUAL COST
1.	Final Facility Plans	$ 460,372
2.	Removal of Piping & Equipment from Containment	301,736
3.	Removal of Superheater & Superheater Building	86,988
4.	Removal of Passageway & Equipment	67,987
5.	Removal and Disposal of Vessel Internals	785,110
6.	Removal and Disposal of Pressure Vessel	1,057,337
7.	Removal and Disposal of Biological Shield	1,233,969
8.	Removal of Containment Building & Structure	391,468
9.	Removal of Miscellaneous Equipment	19
10.	Material Disposal	1,248,888
11.	Facility Closeout	103,901
	TOTAL	$5,737,775

Attachment 4

October 1973

THE EFFECT OF RESIDUAL ELEMENTS ON THE SENSITIVITY OF PRESSURE VESSEL STEELS TO IRRADIATION EMBRITTLEMENT

Prepared by

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Materials Engineering Branch

Directorate of Licensing

The Effect of Residual Elements on the Sensitivity of Pressure Vessel Steels to Irradiation Embrittlement

INTRODUCTION

Pressure vessels for light water reactors are designed to operate at 550°F (288°C) for a projected lifetime of 40 years. During this period, the neutron exposure in the beltline region of the vessel may reach fluences as great as 6×10^{19} n/cm^2 ($E \geq 1$ MeV).

Reactor pressure vessels are shop fabricated (1) from plates and forgings of manganese-molybdenum (Type A302B) or nickel-modified manganese-molybdenum (Type A533B and A508-2) low alloy steels. A nickel-chromium-molybdenum (Type A543) steel has been approved by the ASME Code for vessel construction, but has not yet been used. Although great care is used in steel processing, and the vessels are fabricated to conform to Codes approved by the ASME, the steels normally contain low percentages of residual unspecified tramp elements which may increase the susceptibility to irradiation-induced embrittlement.

The objective of this report is to review the literature and to recommend design curves for estimating the shift in the reference nil-ductility transition temperature in pressure vessel steels as functions of the residual element content and the neutron fluence at reactor operating temperature.

LITERATURE REVIEW

The properties of pressure vessel steels are changed as a result of neutron irradiation. This, of course, will restrict the temperature-pressure regime under which a reactor may operate. It necessitates the inclusion of surveillance test specimens in the beltline region to monitor the change in properties as a function of vessel lifetime. Neutron irradiation is known to increase the yield strength, increase the brittle to ductile transition temperature, decrease the total shelf energy and decrease the rate of change from brittle to ductile fracture of pressure vessel steels (2). Irradiation damage is assessed by measuring the shift in Charpy V-notch impact properties. This shift has previously been determined at the 30 ft-lbs energy level, but Appendix H, 10 CFR 50, specifies that 50 ft-lbs energy values will be used.

Programs have been conducted at the Naval Research Laboratory to study the effect of variation in composition, annealing, initial ferritic grain size, metallurgical structures, and processing variables, including welding procedures, on the irradiation embrittlement of pressure vessel steels (3). The studies were jointly sponsored by the Atomic Energy Commission, steel producers and vessel fabricators. The programs have established that the copper and phosphorus content can significantly contribute to the sensitivity of pressure vessel steel to embrittlement at 550°F (288°C.) (4).

The reduction of the sensitivity of steel to irradiation embrittlement by controlling the composition has been viewed with considerable interest by the vessel fabricators, and applied to welds (5), advanced alloy composition (6), and commercial scale melting practice (7).

MATERIAL SPECIFICATION STATUS

Steel producers and vessel fabricators have generally approved of the use of ferrous materials containing low residual tramp elements for pressure vessel construction. The ASTM approved specification A533-72a for heat treated manganese-molybdenum and manganese-molybdenum-nickel steels for reactor vessels construction. This specification limits the copper and phosphorus contents to 0.10 and 0.012 percent (maximum), for heat analysis and to 0.12 and 0.017 percent (maximum) for product analysis, respectively.

In addition a series of ASTM specifications for nuclear materials are in process of being approved. These are specifications for castings (A613), bolting materials (A614), plates (A647), forging and bars (A654), and piping and tubing (A655). The latter specifications are cross-referenced to A533-72a.

The AWS also reacted favorably by issuing specification A-5.17 for controlling the composition of weld metal for reactor pressure vessel construction. AWS specification A-5.17 is replacing ASTM specification

A558 and will control the copper, phosphorus, vanadium, and sulfur contents in weld metal deposits to 0.10, 0.12, 0.05, and 0.015 percent maximum, respectively.

Westinghouse, Combustion Engineering, and Babcock and Wilcox are specifying low residual tramp elements in material for the beltline region in the reactor vessel. Although General Electric has not adopted the ASTM and AWS specifications, the necessity in the boiling water reactor is not as great due to the lower fluence.

EFFECT OF RESIDUAL IMPURITIES

Copper and Phosphorus - The impact properties of irradiated and unirradiated ASTM A-543 weld metal were statistically analyzed in a cooperative program between NRL and B&W (5). The specimens were irradiated at 550°F to a fluence of 2.8×10^{19} n/cm^2 ($E \geq 1$ MeV). The work indicated that the shift in transition temperature may be expressed in terms of the copper and phosphorus contents by the following equation:

$$\Delta TT\ (°F) = -118 + 14800\ (\%P) + 990\ (\%Cu)$$

Fundamental research indicates that copper in iron alloys promotes formation of defect aggregates at relatively low neutron fluence. The defects are a result of vacancy trapping and the formation of vacancy loops (8). While the effect of phosphorus appears to be akin to temper brittleness, resulting from the irradiation induced diffusion of the element to the ferrite-carbon interface (8, 9).

Copper - The shift in the reference nil-ductility transition temperature for pressure vessel steel as a function of fluence at 550°F and the residual copper content for base and weld metal is shown in Fig. 1. The data for the curves are shown in Table 1. The curves were drawn to predict the maximum shift in the transition temperature at 30 ft-lbs "fix" Charpy V-notch impact energy for four levels of copper content. The copper content varied from ≤ 0.10% (ASTM specification A533-72a) to ≥ 0.25% (unrestricted residual). The data indicates that the weld metal was slightly more sensitive to irradiation damage than the base metal. The increased sensitivity was approximately equivalent to an increase of 0.05% in the residual copper content of the base metal. However, sensitivity may be influenced by the welding procedure. A general improvement in weld metal has been observed (5).

The design curves for the maximum shift in the reference transition temperature are shown in Fig. 2. The curves conservatively estimate the maximum shift at 550°F at fluences from 10^{18} to 10^{20} n/cm^2 (E > 1 MeV) for pressure vessel steel containing <0.10% and uncontrolled copper. The upper bound curve in Fig. 1 is above all the data available to us at this time, which include the results of recent surveillance test specimens. The lower bound curve shows the shift for <0.10% copper base material. A shift 40°F higher at 10^{20} n/cm^2 (E > 1 MeV) was observed for weld metal containing 0.10% copper. The magnitude of this shift is reflected in the location of the intermediate curves.

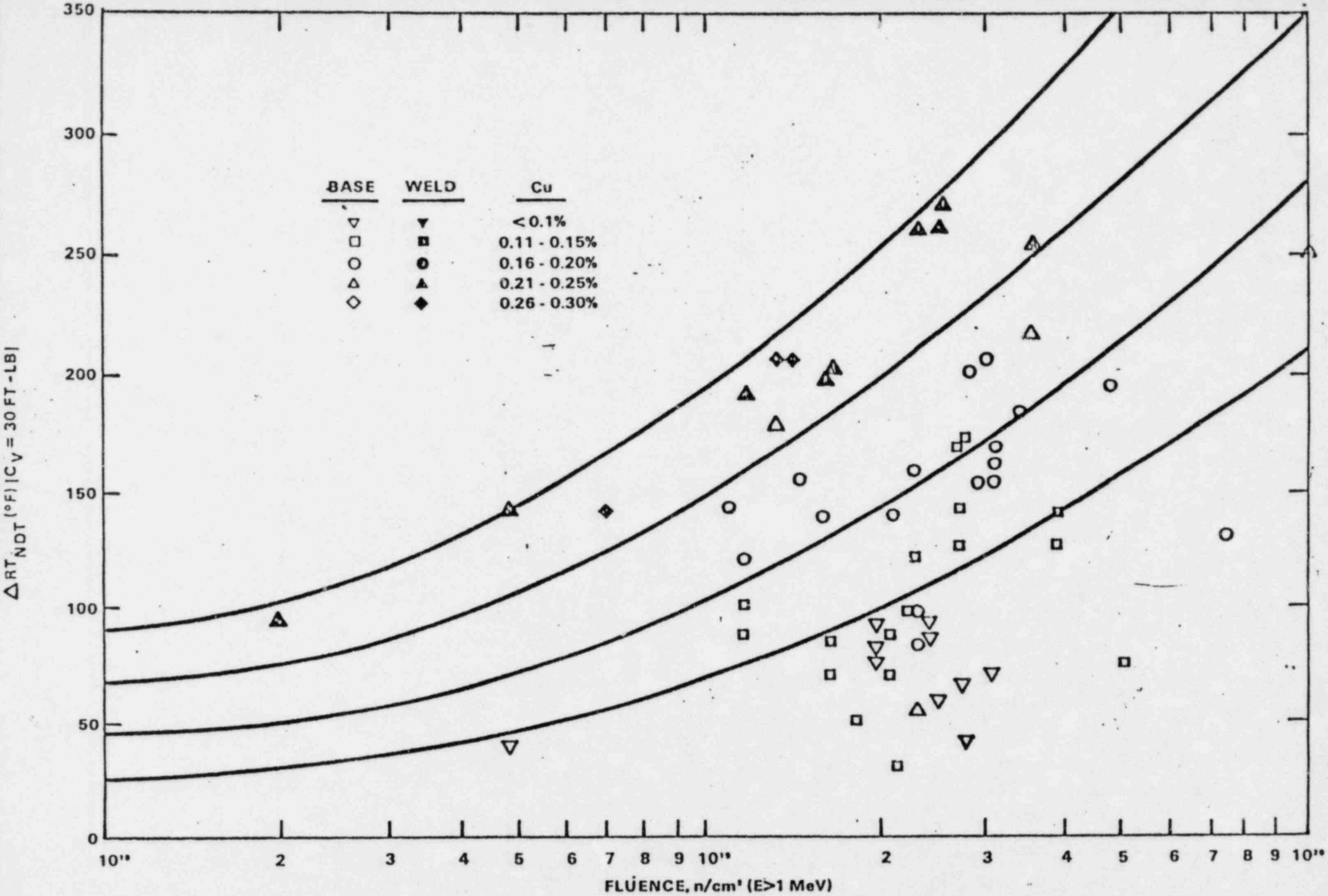

Fig.1 - SHIFT IN RT_{NDT} AS FUNCTION OF FLUENCE AT 550°F AND COPPER CONTENT OF PRESSURE VESSEL STEELS

TABLE 1

Shift In The Nil-Ductility Transition Temperature of Pressure Vessel Steels As A Function of Fluence and Copper Content

ΔRT_{NDT} (°F)	Fluence $x\ 10^{19}$ @ 550°F (n/cm^2, E>1 MeV)	Copper Content (%)	Type Steel (ASTM)	Literature Reference
120	2.3	0.15	A533B, 1	10
95	2.3	0.14	"	10
80	2.0	0.09	"	10
90	2.0	0.09	"	10
35	0.5	0.09	A533B, 2	10
75	2.0	0.09	"	10
50	0.2	0.20	A302-B	10
140	1.1	0.20	"	10
155	1.5	0.20	"	10
140	1.7	0.20	"	10
140	2.1	0.20	"	10
160	2.3	0.20	"	10
155	3.0	0.20	"	10
160	3.1	0.20	"	10
170	3.1	0.20	"	10
155	3.1	0.20	"	10
180	3.4	0.20	"	10
195	4.8	0.20	"	10
50	1.8	0.12	A533B, 1	10

TABLE 1 (CONT'D)

ΔRT_{NDT} (°F)	Fluence x 10^{19} @ 550°F (n/cm^2, E>1 MeV)	Copper Content (%)	Type Steel (ASTM)	Literature Reference
70	1.7	0.14	A533B, 1	10
85	1.7	0.11	"	10
180	1.4	0.22	A533C, 2	10
205	1.4	0.27	A533C, 2	10
200	1.7	0.22	A533B, 1	10
40	2.8	0.03	A533B, 1	7
65	2.8	0.03	A533B, 2	7
70	3.1	0.03	A533B, 1	7
140	2.8	0.13	A533B, 1	7
125	2.8	0.13	A533B, 2	7
55	2.3	0.21	A302-B	4
95	2.3	0.20	A302-B	4
80	2.3	0.20	A302-B	4
155	3.0	0.20	A302-B	4
75	4.52 - 5.59	0.14	A533B, 1	11
130	3.64 - 4.24	0.14	"	11
85	1.18 - 1.33	0.14	"	11
256	2.73 - 4.25	0.22	A533B, 1	11
125	3.64 - 4.24	0.14	A533B, 1	11

TABLE 1 (CONT'D)

ΔRT_{NDT} (°F)	Fluence x 10^{19} @ 550°F (n/cm^2, E>1 MeV)	Copper Content (%)	Type Steel (ASTM)	Literature Reference
120	1.2	0.20	A302-B	12
100	1.2	0.15	A302-B	12
80	1.2	-	Weld	12
205	3	0.20	A302-B	13
85	2.1	0.11	A302-B	14
30	2.1	0.11	A302-B	14
70	2.1	0.11	A302-B	14
0	0.5	0.09	A533B, 1	15
85	2.4	0.09	A533B, 1	15
0	0.5	0.09	A533B, 1	15
90	2.4	0.09	A533B, 1	15
60	2.5	0.09	A533B, 1	16
215	3-4	0.25	A533B, 1	17
255	3-4	0.25	"	17
260	2.5	0.23	A533B, 1	18
165	2.6	0.14	A533B, 1	18
170	2.7	0.14	"	18
200	2.8	0.18	"	18
95	0.2	0.21	A302-B	23
140	0.7	0.27	"	24
140	0.48	0.23	A508-2	25

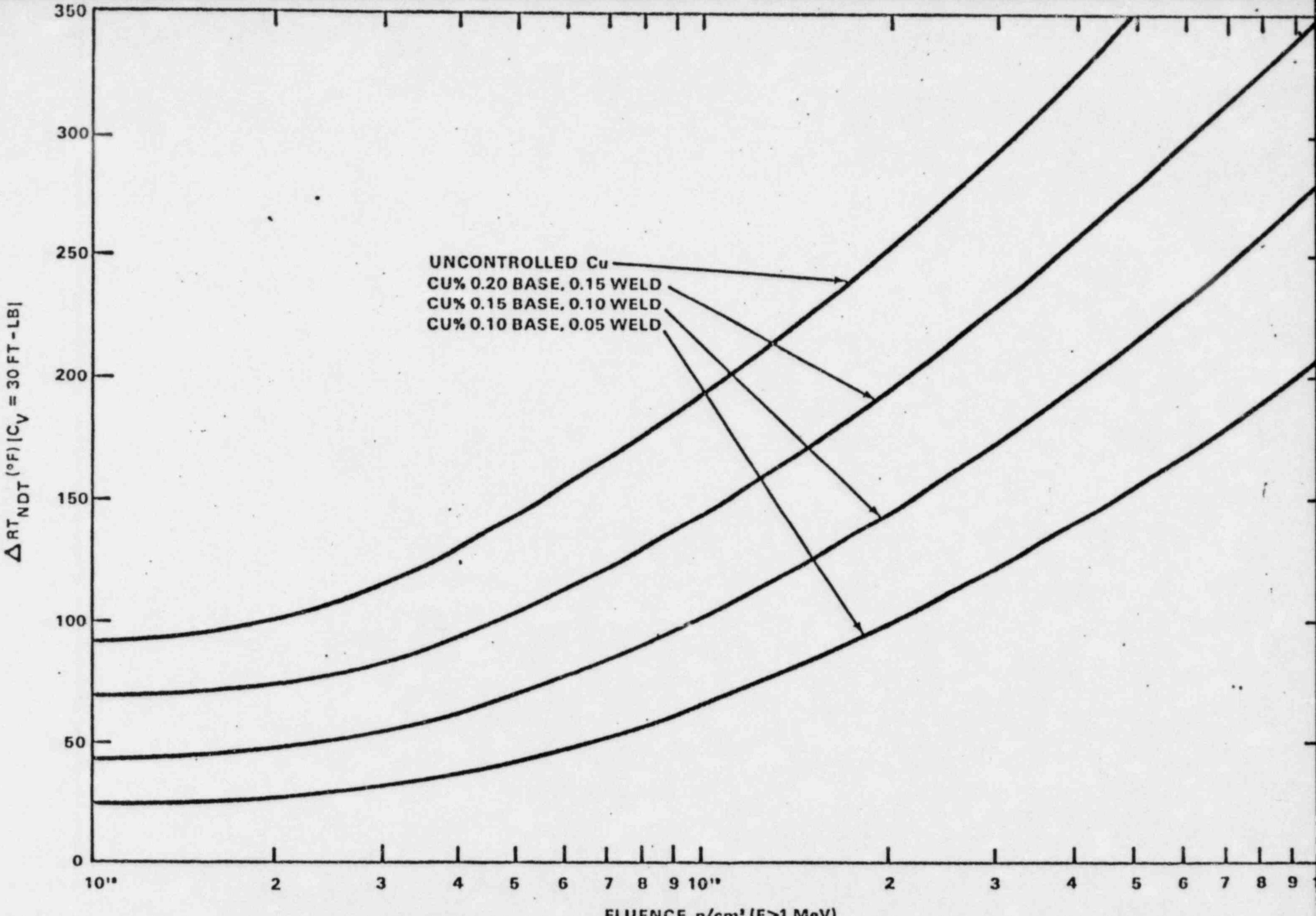

Fig. 2 - DESIGN CURVES FOR THE MAXIMUM INCREASE IN REFERENCE TRANSITION TEMPERATURE AS FUNCTION OF FLUENCE AND COPPER CONTENT AT 550°F

Phosphorus - Although phosphorus appears to increase the shift in the reference nil-ductility transition temperature, its effect may not be as great as predicted by the equation in the previous section. For example, 0.021% P was observed to increase the transition temperature by 80°F in an experimental Type A302-B pressure vessel steel (4). Furthermore, Palme (19) reported an increase of 70°F in two cupriferous (Type A302-B) steels, respectively, when the phosphorus content was increased from 0.004 to 0.015 percent.

These experimental values are lower than predicted by the equation. In combination with 0.06% Cu, the equation predicts transition temperature increases of 252°F and 104°F, respectively, for steel containing 0.021 and 0.011 percent phosphorus.

Tin, Arsenic, Antimony - The detrimental effect of phosphorus on the sensitivity of pressure vessel steels to irradiation - induced embrittlement has made tin, arsenic and antimony suspect elements. Iron ore from the southern district of the United States may have a high arsenic content. The effect of tin, arsenic, and antimony were investigated on experimental Ni-Cr-Mo-V forgings containing low (0.006%) and high (0.019%) phosphorus contents (20). Charpy V-notch specimens were exposed to a fluence of 3.1×10^{19} n/cm^2 (E $\geq$ 1 MeV) at 550°F..

The results indicated that tin, arsenic, and antimony did not directly affect the shift in the nil-ductility transition temperature in the

maximum concentration range investigated (170 ppm). The increase in the transition temperature for low phosphorus specimens (0.006%) was 10°F, compared to 170°F for high phosphorus specimens (0.016%). It was concluded that a tin-phosphorus interaction occurred in specimens containing both high (0.019%) phosphorus and high (137 ppm) tin, which reduced the detrimental effect of phosphorus.

Sulfur - The addition of sulfur (~0.02%) to Type A 302B steel reduced the full shear energy absorption (125 to 75 ft/lbs) without influencing the irradiation-embrittlement sensitivity (4).

Vanadium - The addition of vanadium (~ 0.08%) to Type A 302B steel either had a small or secondary effect on the irradiation-embrittlement sensitivity (4).

Nitrogen and Aluminum - Powers indicated that the irradiation-embrittlement sensitivity of Type A 336 forging steel was related to the uncombined nitrogen content (21). The steel was insensitive when nitrogen was in solid solution and not combined with aluminum as the nitride (AlN). Potapovs and Hawthorne observed that nitrogen and aluminum addition (either singly or in combination) to 3-1/2 Ni-Cr-Mo and 7-1/2 Ni-Cr-Mo base compositions resulted in no marked effect on irradiation-embrittlement sensitivity (4). The irradiations were conducted at 550°F to a fluence of 2.2×10^{19} n/cm^2 ($E \geq 1$ MeV).

CURRENT RESEARCH PROGRAM

A research program is currently in progress at NRL in cooperation with Combustion Engineering and AEC (RDT) to provide maximum design curves to confirm the effect of fluence and the residual copper content on the irradiation damage of Type A533B pressure vessel steel. The program consists of three tasks, which are defined as follows:

Task A - The change in Charpy V-notch impact properties will be determined on the plate, weld and heat affected zone for three steel compositions: (a) normal copper content ($\geq$ 0.15% Cu) - standard, nonimproved commercial productions; (b) low copper content (0.10% Cu max.) with 0.012% P max. - improved production to meet ASTM specification A533-72a; (c) extra low copper content (0.06% Cu max.) - the practical lower limit for copper control. The irradiation will be conducted at 550°F to $\sim 2.5 \times 10^{19}$ and $\sim 5.0 \times 10^{19}$ n/cm^2 (E > 1 MeV) fluences, respectively.

Task B - The change in tensile properties will be determined on plate and weld metal for each of the above compositions after irradiation to a fluence of $\sim 5.0 \times 10^{19}$ n/cm^2 (E > 1 MeV) at 550°F.

Task C - Post irradiation heat treatments (650°F) will be conducted on specimens from the normal copper content ($\geq$ 0.15% Cu) plate, weld metal and heat affected zone to determine the extent of recovery of the Charpy V-notch impact and tensile properties.

CONCLUSIONS

1. Investigations have shown that small amounts of copper and phosphorus markedly increase the sensitivity of pressure vessel steels to irradiation-induced embrittlement.

2. Specifications have been developed to limit the content of residual elements in reactor pressure vessel steels and welds, and are being used for their fabrication.

3. The change in RT_{NDT} as a function of fluence and copper content may be predicted from Fig. 2. Fig. 2 is considered subject to change when further data are developed.

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U.S. NUCLEAR REGULATORY COMMISSION

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REGULATORY GUIDE

July 1975

Attachment 5 (Part 1)

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.99

EFFECTS OF RESIDUAL ELEMENTS ON PREDICTED RADIATION DAMAGE TO REACTOR VESSEL MATERIALS

A. INTRODUCTION

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," which were added to 10 CFR Part 50 effective August 16, 1973, to implement, in part, Criterion 31, necessitate the prediction of the amount of radiation damage to the reactor vessel of water-cooled power reactors throughout its service life.

This guide describes general procedures acceptable to the NRC staff as an interim basis* for predicting the effects of the residual elements copper and phosphorus on neutron radiation damage to the low-alloy steels currently used for reactor vessels.

B. DISCUSSION

The principal examples of NRC requirements that necessitate prediction of radiation damage are:

*Research and construction experience with low-residual-element compositions of these steels is accumulating rapidly and is expected to provide a firm basis for acceptable procedures in the near future.

1. Paragraph II.H of Appendix G defines the beltline in terms of a predicted adjustment of reference temperature at end of service life in excess of 50°F; paragraphs III.C and IV.B specify the additional test requirements for beltline materials that supplement the requirements for reactor vessel materials generally.

2. Paragraph II.C.3 of Appendix H establishes the required number of surveillance capsules on the basis of the predicted adjusted reference temperature at the end of service life. In addition, withdrawal of the first capsule (when four or more are required) is to occur when the predicted adjustment of reference temperature is approximately 50°F or at one-fourth of the service life, whichever is earlier.

3. Paragraph IV.C of Appendix G requires that vessels be designed to permit a thermal annealing treatment if the predicted value of adjusted reference temperature exceeds 200°F during their service life.

4. Paragraph II.B of Appendix H incorporates ASTM E185-73 by reference. Paragraph 4.1 of ASTM E185-73 requires that the materials to be placed in surveillance be those that may limit operation of the reactor during its lifetime, i.e., those expected to have the highest adjusted reference temperature or the lowest Charpy upper-shelf energy at end of life. Both measures of radiation damage must be considered.

5. Paragraph V.B. of Appendix G describes the basis for setting the upper limit for pressure as a function of temperature during heatup and cooldown for a given service period in terms of the predicted value of the

adjusted reference temperature at the end of the service period..

The two measures of radiation damage used in this guide are obtained from the results of the Charpy V-notch impact test. Appendix G to 10 CFR Part 50 requires that a full curve of absorbed energy versus temperature be obtained through the ductile-to-brittle transition temperature region. The latter is located by the reference temperature, RT_{NDT}, which is defined in paragraph II.F of Appendix G. The "shift" of the adjusted reference temperature is defined in Appendix G as the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material, measured at the 50-foot-pound energy level or measured at the 35-mil lateral expansion level, whichever temperature shift is greater. In using published data that report only the temperature shift measured at the 30-foot-pound energy level, it has been assumed herein that the adjustment of the reference temperature is equal to the 30-foot-pound shift.

The second measure of radiation damage is the decrease in the Charpy upper-shelf energy level. In the absence of a standard definition, the upper-shelf energy is defined herein as the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. Normally, at least three specimens should be included; more specimens should be included when the shelf level appears to be marginal.

The measure of fluence used herein is n/cm^2 ($E > 1$ MeV), consistent with the data base* for the expression relating fluence to neutron damage. This procedure is not intended to preclude future use of data that are given in terms of neutron damage fluence.

As used herein, references to "% Cu" and "% P" mean the weight percent of copper and phosphorus as measured in the surveillance program per ASTM E 185-73. However, if such results are not available, the results of a product analysis may be used.

Use of the procedures for prediction of radiation damage given in the regulatory position should be limited to irradiation at 550 ± 25°F, because temperature is important to damage recovery processes. As a guideline, irradiation at 450°F has been shown to cause twice the adjustment of reference temperature and irradiation at 650°F, about half the adjustment produced by irradiation at 550°F for the steels cited in the regulatory position when the copper content is about 0.15%. The effects of irradiation temperature on decrease in shelf energy should be considered qualitatively similar to those cited for the adjustment of reference temperature.

Sensitivity to neutron embrittlement may be affected by other residual elements such as vanadium and by deoxidation practice, as indicated by the findings of current research. In predicting radiation damage for materials that differ in residual element content or deoxidation practice from those that make up the data base, such findings should be considered. Other residual elements, notably sulfur, impair the initial Charpy shelf energy of these materials, and their content should be kept low. Clearly, it is the remaining toughness at end of life or at some other critical period that is important. Such toughness may be given in terms of the margin between the operating temperature (nominally 550°F) and the limiting temperature based on toughness. A margin of 200 degrees is desirable to permit safe management of system transients. At full power, the limiting temperature based on toughness is generally 150-200 degrees above RT_{NDT}; hence, the latter should not exceed 150-200°F at end of life. This limit also avoids the problems of providing for annealing, per paragraph IV.C of Appendix G. The levels of residual elements such as copper, phosphorus, sulfur, and vanadium that are required to achieve the limit of 200°F adjusted reference temperature at end of life in a given reactor vessel will depend on the initial values of RT_{NDT} of the beltline materials and on the predicted fluence at the particular locations in the vessel where the materials are used.

C. REGULATORY POSITION

1. Prediction of neutron radiation damage to the beltline of reactor vessels of light water reactors should be based on the following procedures. When surveillance data from the reactor in question become available, both sets of information should be considered in making the prediction.

Reference temperature should be adjusted as a function of fluence and residual element content in accordance with the following expression, within the limits listed below:

$$A = [40 + 1000(\%\,Cu - 0.08) + 5000(\%\,P - 0.008)]\,[f/10^{19}]^{1/2}$$

where

A = predicted adjustment of reference temperature, °F

f = fluence, n/cm^2 ($E > 1$ MeV)

$\%\,Cu$ = weight percent of copper. If $\%\,Cu \leq 0.08$, use 0.08.

*The data base for this guide is that given by Spencer H. Bush, "Structural Materials for Nuclear Power Plants," 1974 ASTM Gillett Memorial Lecture, published in ASTM Journal of Testing and Evaluation, Nov. 1974 and its addendum, "Radiation Damage in Pressure Vessel Steels for Commercial Light-Water Reactors."

$\%P$ = weight percent of phosphorus. If $\%P < 0.008$, use 0.008.

If the value of A obtained by the above expression exceeds that given by the curve labeled "Upper Limit" in Figure 1, the "Upper Limit" curve should be used. If % Cu is unknown, the "Upper Limit" curve should be used.

As illustrated in Figure 1 for selected copper and phosphorus contents, the above expression should be considered valid only for $A > 50°F$ and for $f < 6 \times 10^{19}$ n/cm^2 ($E > 1 MeV$).

Charpy upper-shelf energy should be assumed to decrease as a function of fluence and copper content as indicated in Figure 2, within the limits listed below. Interpolation is permitted.

Application of the foregoing procedures should be subject to the following limitations:

a. The procedures apply to those grades of SA-302, 336, 533, and 508 steels having minimum specified yield strengths of 50,000 psi and under and to their welds and heat-affected zones.

b. The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater damage, and irradiation above 575°F may be considered to produce less damage. The correction factor used should be justified.

c. The expression for A is given in terms of fluence as measured by units of n/cm^2 ($E > 1$ MeV); however, the expression may be used in terms of fluence as measured by units of neutron damage fluence, provided the constant 10^{19} n/cm^2 ($E > 1$ MeV) is changed to the corresponding value of neutron damage fluence.

d. Application of these procedures to materials having residual element content beyond that represented by the current data base should be justified by submittal of data.

2. For new plants, the reactor vessel beltline materials should have the content of residual elements such as copper, phosphorus, sulfur, and vanadium controlled to low levels. The levels should be such that the predicted adjusted reference temperature at end of life is less than 200°F.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

This guide reflects current regulatory practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the positions described in this guide will be used by the NRC staff as follows:

1. The method described in regulatory position C.1 of this guide will be used in evaluating all predictions of radiation damage called for in Appendices G and H to 10 CFR Part 50 submitted on or after January 15, 1976; however, if an applicant wishes to use the recommendations of regulatory position C.1 in developing submittals before January 15, 1976, the pertinent portions of the submittal will be evaluated on the basis of this guide.

2. The recommendations of regulatory position C.2 will be used in evaluating construction permit applications docketed on or after January 15, 1976; however, if an applicant whose application for construction permit is docketed before January 15, 1976, wishes to use the recommendations of regulatory position C.2 of this regulatory guide in developing submittals for the application, the pertinent portions of the application will be evaluated on the basis of this guide.

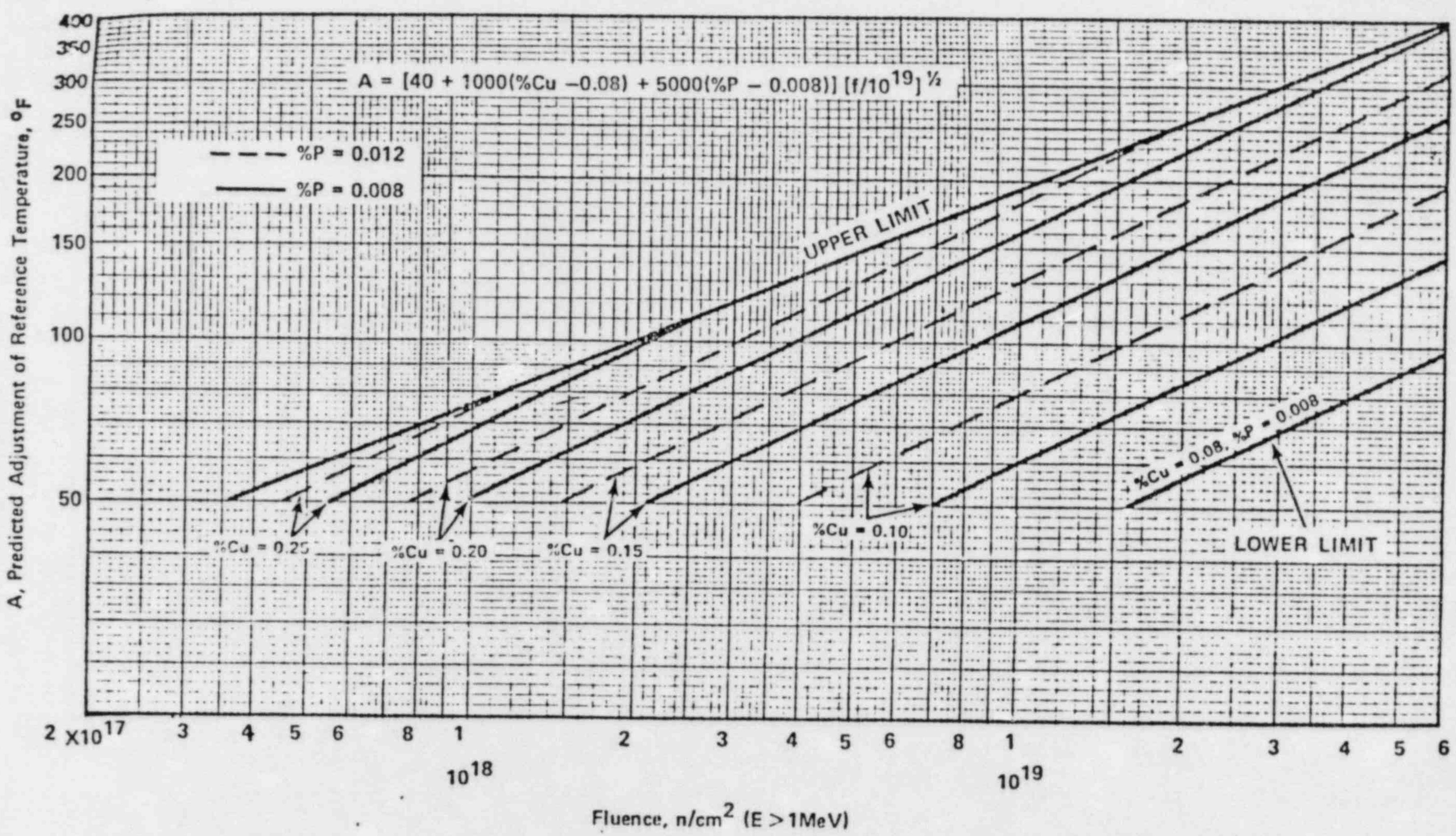

Figure 1 Predicted Adjustment of Reference Temperature as a Function of Copper and Phosphorus Content and Fluence.

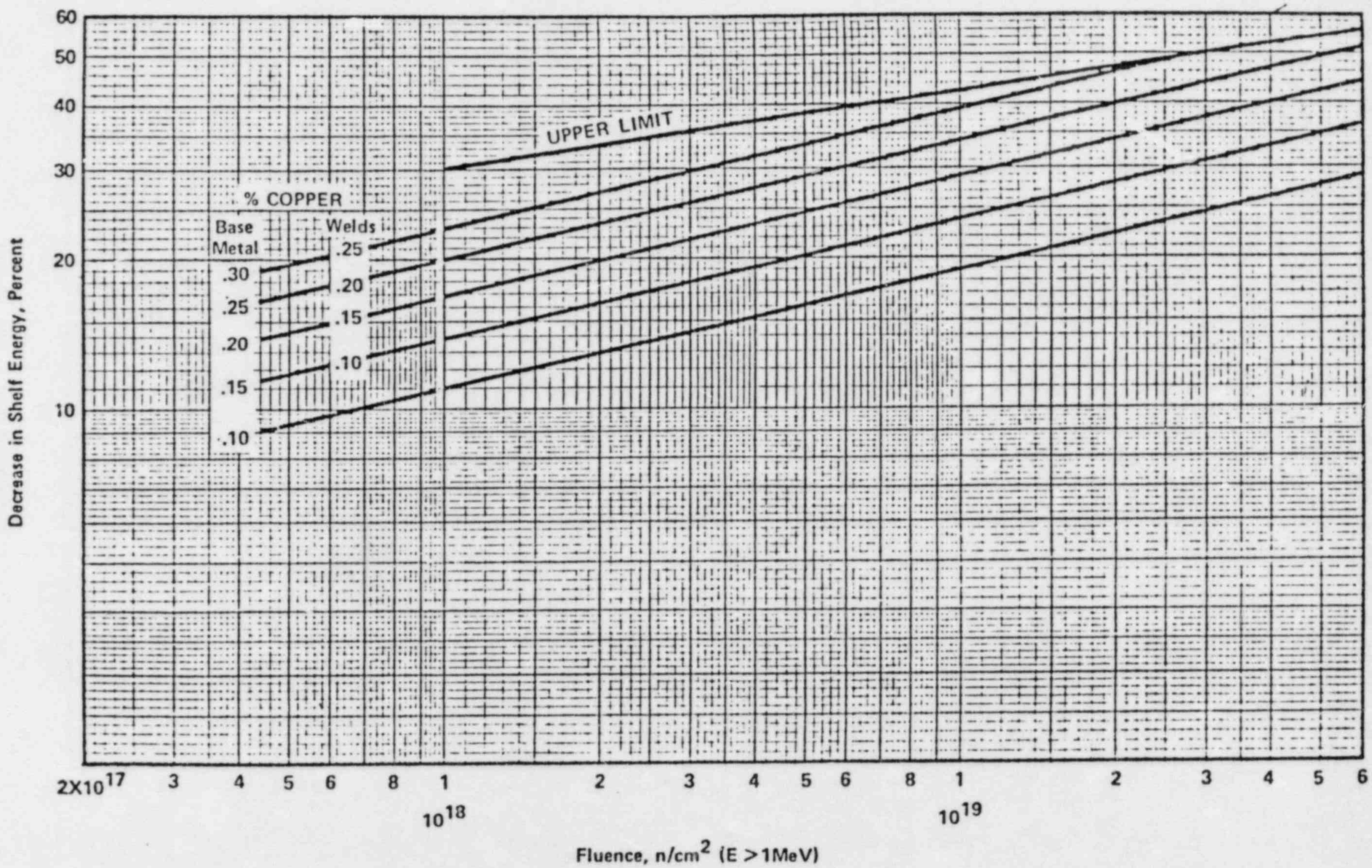

Figure 2 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence.

Attachment 5 (part 2)

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
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REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.99

EFFECTS OF RESIDUAL ELEMENTS ON PREDICTED RADIATION DAMAGE TO REACTOR VESSEL MATERIALS

A. INTRODUCTION

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," which were added to 10 CFR Part 50 effective August 16, 1973, to implement, in part, Criterion 31, necessitate the prediction of the amount of radiation damage to the reactor vessel of water-cooled power reactors throughout its service life.

This guide describes general procedures acceptable to the NRC staff as an interim basis* for predicting the effects of the residual elements copper and phosphorus on neutron radiation damage to the low-alloy steels currently used for light-water-cooled reactor vessels. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position. **

B. DISCUSSION

The principal examples of NRC requirements that necessitate prediction of radiation damage are:

1. Paragraph II.H of Appendix G defines the beltline in terms of a predicted adjustment of reference temperature at end of service life in excess of 50°F; paragraphs III.C and IV.B specify the additional test requirements for beltline materials that supplement the requirements for reactor vessel materials generally.

2. Paragraph II.C.3 of Appendix H establishes the required number of surveillance capsules on the basis of the predicted adjusted reference temperature at the end of service life. In addition, withdrawal of the first capsule (when four or more are required) is to occur when the predicted adjustment of reference temperature is approximately 50°F or at one-fourth of the service life, whichever is earlier.

3. Paragraph IV.C of Appendix G requires that vessels be designed to permit a thermal annealing treatment if the predicted value of adjusted reference temperature exceeds 200°F during their service life.

4. Paragraph II.B of Appendix H incorporates ASTM E185-73 by reference. Paragraph 4.1 of ASTM E185-73 requires that the materials to be placed in surveillance be those that may limit operation of the reactor during its lifetime, i.e., those expected to have the highest adjusted reference temperature or the lowest Charpy upper-shelf energy at end of life. Both measures of radiation damage must be considered.

5. Paragraph V.B of Appendix G describes the basis for setting the upper limit for pressure as a function of temperature during heatup and cooldown for a given service period in terms of the predicted value of the adjusted reference temperature at the end of the service period.

The two measures of radiation damage used in this guide are obtained from the results of the Charpy V-

*Research and construction experience with low-residual-element compositions of these steels is accumulating rapidly and is expected to provide a firm basis for acceptable procedures in the near future.

**Lines indicate substantive changes from previous issue.

notch impact test. Appendix G to 10 CFR Part 50 requires that a full curve of absorbed energy versus temperature be obtained through the ductile-to-brittle transition temperature region. The latter is located by the reference temperature, RT_{NDT}, which is defined in paragraph II.F of Appendix G. The "shift" of the adjusted reference temperature is defined in Appendix G as the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material, measured at the 50-foot-pound energy level or measured at the 35-mil lateral expansion level, whichever temperature shift is greater. In using published data that report only the temperature shift measured at the 30-foot-pound energy level, it has been assumed herein that the adjustment of the reference temperature is equal to the 30-foot-pound shift.

The second measure of radiation damage is the decrease in the Charpy upper-shelf energy level. In the absence of a standard definition, the upper-shelf energy is defined herein as the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. Normally, at least three specimens should be included; more specimens should be included when the shelf level appears to be marginal. However, if specimens are tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper-shelf energy.

The measure of fluence used herein is the number of neutrons per square centimeter (E>1MeV). An assumed fission-spectrum energy distribution was used in calculating the fluence for most of the data base.* However, for application to a reactor vessel, the calculated spectrum is used to predict fluence at a given location in the wall. This procedure is not intended to preclude future use of data that are given in terms of neutron damage fluence.

As used herein, references to "% Cu" and "% P" mean the weight percent of copper and phosphorus as measured in the surveillance program per ASTM E185-73. However, if such results are not available, the results of a product analysis may be used.

Use of the procedures for prediction of radiation damage given in the regulatory position should be limited to irradiation at 550±25°F, because temperature is important to damage recovery processes. As a guideline, irradiation at 450°F has been shown to cause twice the adjustment of reference temperature and irradiation at 650°F, about half the adjustment produced by irradiation at 550°F for the fluence levels and the steels cited in the regulatory position when the copper content is about 0.15%. The effects of irradiation temperature on decrease in shelf energy should be considered qualitatively similar to those cited for the adjustment of reference temperature.

Sensitivity to neutron embrittlement may be affected by other residual elements such as vanadium and by deoxidation practice, as indicated by the findings of current research. In predicting radiation damage for materials that differ in chemical content or deoxidation practice from those that make up the data base, such findings should be considered. Other residual elements, notably sulfur, impair the initial Charpy shelf energy of these materials, and their content should be kept low. Clearly, it is the remaining toughness at end of life or at some other critical period that is important. Such toughness may be given in terms of the margin between the operating temperature (nominally 550°F) and the limiting temperature based on toughness. A margin of 200 degrees is desirable to permit safe management of system transients. At full power, the limiting temperature based on toughness is generally 150-200 degrees above RT_{NDT}; hence, the latter should not exceed 150-200°F at end of life. This limit also avoids the problems of providing for annealing, per paragraph IV.C of Appendix G. The levels of residual elements such as copper, phosphorus, sulfur, and vanadium that are required to achieve the limit of 200°F adjusted reference temperature at end of life in a given reactor vessel will depend on the initial values of RT_{NDT} of the beltline materials and on the predicted fluence at the particular locations in the vessel where the materials are used.

When surveillance data from the reactor in question become available, the weight given to it relative to the information in this guide should depend on the credibility of the surveillance data as judged by the following criteria:

1. Materials in the capsule should be those judged most likely to be controlling with regard to radiation damage according to the provisions of this guide.

2. Scatter in the Charpy data should be small enough to avoid large uncertainty in curve fitting.

3. The change in yield strength should be consistent with the shift in the Charpy curve.

4. The relationship to previous surveillance data from the same reactor should be consistent with the normal trends of such data.

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

*The data base for this guide is that given by Spencer H. Bush, "Structural Materials for Nuclear Power Plants," 1974 ASTM Gillett Memorial Lecture, published in ASTM Journal of Testing and Evaluation, Nov. 1974, and its addendum, "Radiation Damage in Pressure Vessel Steels for Commercial Light-Water Reactors."

C. REGULATORY POSITION

1. When credible surveillance data from the reactor in question are not available, prediction of neutron radiation damage to the beltline of reactor vessels of light water reactors should be based on the following procedures.

a. Reference temperature should be adjusted as a function of fluence and residual element content in accordance with the following expression, within the limits below and in paragraph 1.c.

$$A = [40 + 1000(\% Cu - 0.08) + 5000(\% P - 0.008)]\,[f/10^{19}]^{1/2}$$

where

A = predicted adjustment of reference temperature, °F.

f = fluence, n/cm² (E>1 MeV).

% Cu = weight percent of copper. If % Cu ≤ 0.08, use 0.08.

% P = weight percent of phosphorus. If % P ≤ 0.008, use 0.008.

If the value of A obtained by the above expression exceeds that given by the curve labeled "Upper Limit" in Figure 1, the "Upper Limit" curve should be used. If % Cu is unknown, the "Upper Limit" curve should be used.

As illustrated in Figure 1 for selected copper and phosphorus contents, the above expression should be considered valid only for A > 50°F and for $f < 6 \times 10^{19}$ n/cm² (E > 1 MeV).

b. Charpy upper-shelf energy should be assumed to decrease as a function of fluence and copper content as indicated in Figure 2, within the limits listed in paragraph 1.c. Interpolation is permitted.

c. Application of the foregoing procedures should be subject to the following limitations:

(1) The procedures apply to those grades of SA-302, 336, 533, and 508 steels having minimum specified yield strengths of 50,000 psi and under and to their welds and heat-affected zones.

(2) The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater damage, and irradiation above 575°F may be considered to produce less damage. The correction factor used should be justified.

(3) The expression for A is given in terms of fluence as measured by units of n/cm² (E > 1 MeV); however, the expression may be used in terms of fluence as measured by units of neutron damage fluence, provided the constant 10^{19} n/cm² (E > 1 MeV) is changed to the corresponding value of neutron damage fluence.

(4) Application of these procedures to materials having chemical content beyond that represented by the current data base should be justified by submittal of data.

2. When credible surveillance data from the reactor in question become available, they may be used to represent the adjusted reference temperature and the Charpy upper-shelf energy of the beltline materials at the fluence received by the surveillance specimens.

a. The adjusted reference temperature of the beltline materials at other fluences may be predicted by:

(1) extrapolation to higher or lower fluences from credible surveillance data following the slope of the family of lines in Figure 1 or

(2) a straight-line interpolation between credible data on a logarithmic plot.

b. To predict the decrease in upper-shelf energy of the beltline materials at fluences other than those received by the surveillance specimens, procedures similar to those given in paragraph 2.a may be followed using Figure 2.

3. For new plants, the reactor vessel beltline materials should have the content of residual elements such as copper, phosphorus, sulfur, and vanadium controlled to low levels. The levels should be such that the predicted adjusted reference temperature at the ¼T position in the vessel wall at end of life is less than 200°F.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

This guide reflects current regulatory practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the positions described in this guide will be used by the NRC staff as follows:

1. The method described in regulatory positions C.1 and C.2 of this guide will be used in evaluating all predictions of radiation damage called for in Appendices G and H to 10 CFR Part 50 submitted on or

after June 1, 1977; however, if an applicant wishes to use the recommendations of regulatory positions C.1 and C.2 in developing submittals before June 1, 1977, the pertinent portions of the submittal will be evaluated on the basis of this guide.

2. The recommendations of regulatory position C.3 will be used in evaluating construction permit applications docketed on or after June 1, 1977; however, if an applicant whose application for construction permit is docketed before June 1, 1977, wishes to use the recommendations of regulatory position C.3 of this regulatory guide in developing submittals for the application, the pertinent portions of the application will be evaluated on the basis of this guide.

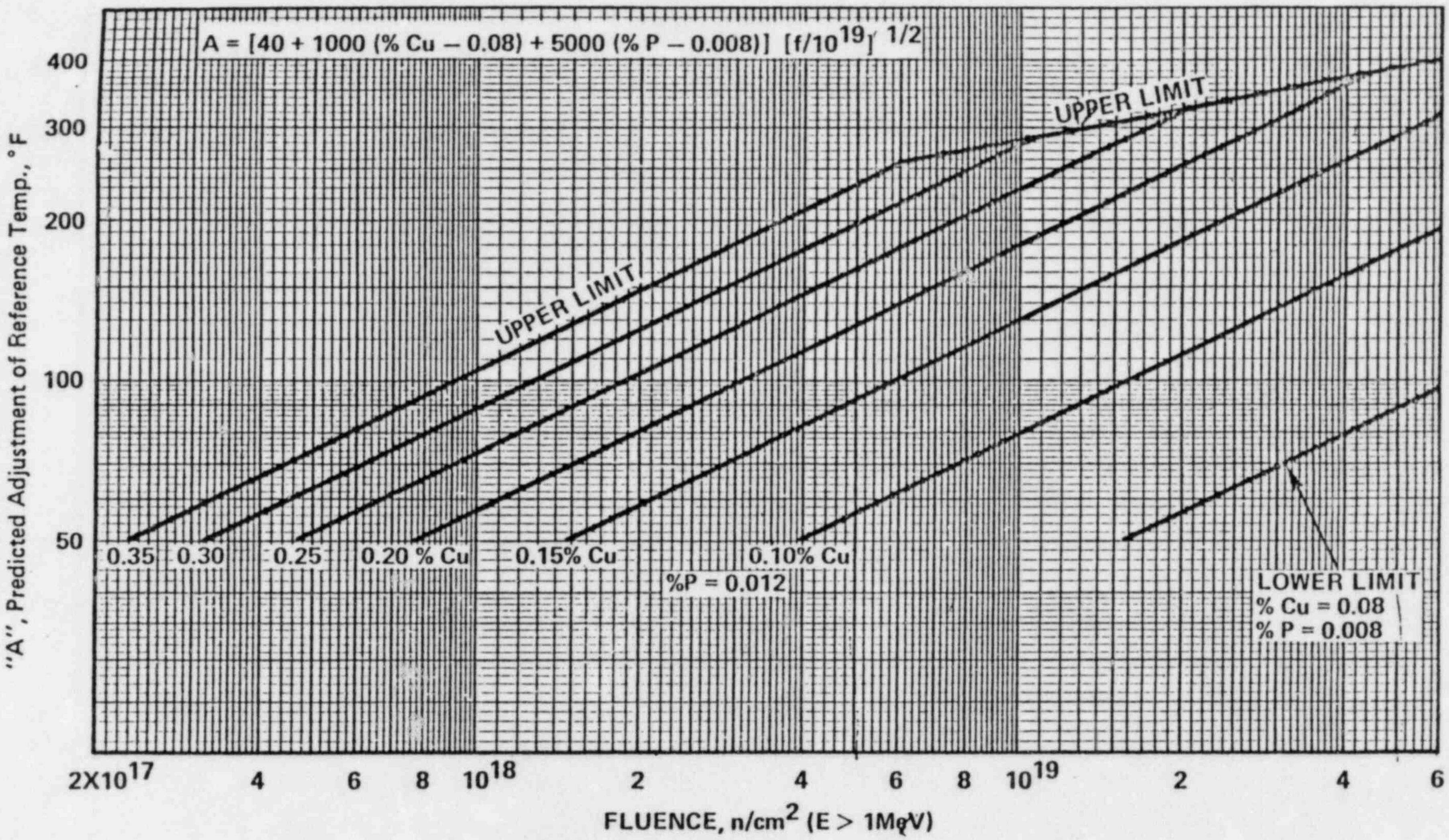

Figure 1 Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content.
For Copper and Phosphorus Contents Other Than Those Plotted, Use the Expression for "A" Given on the Figure.

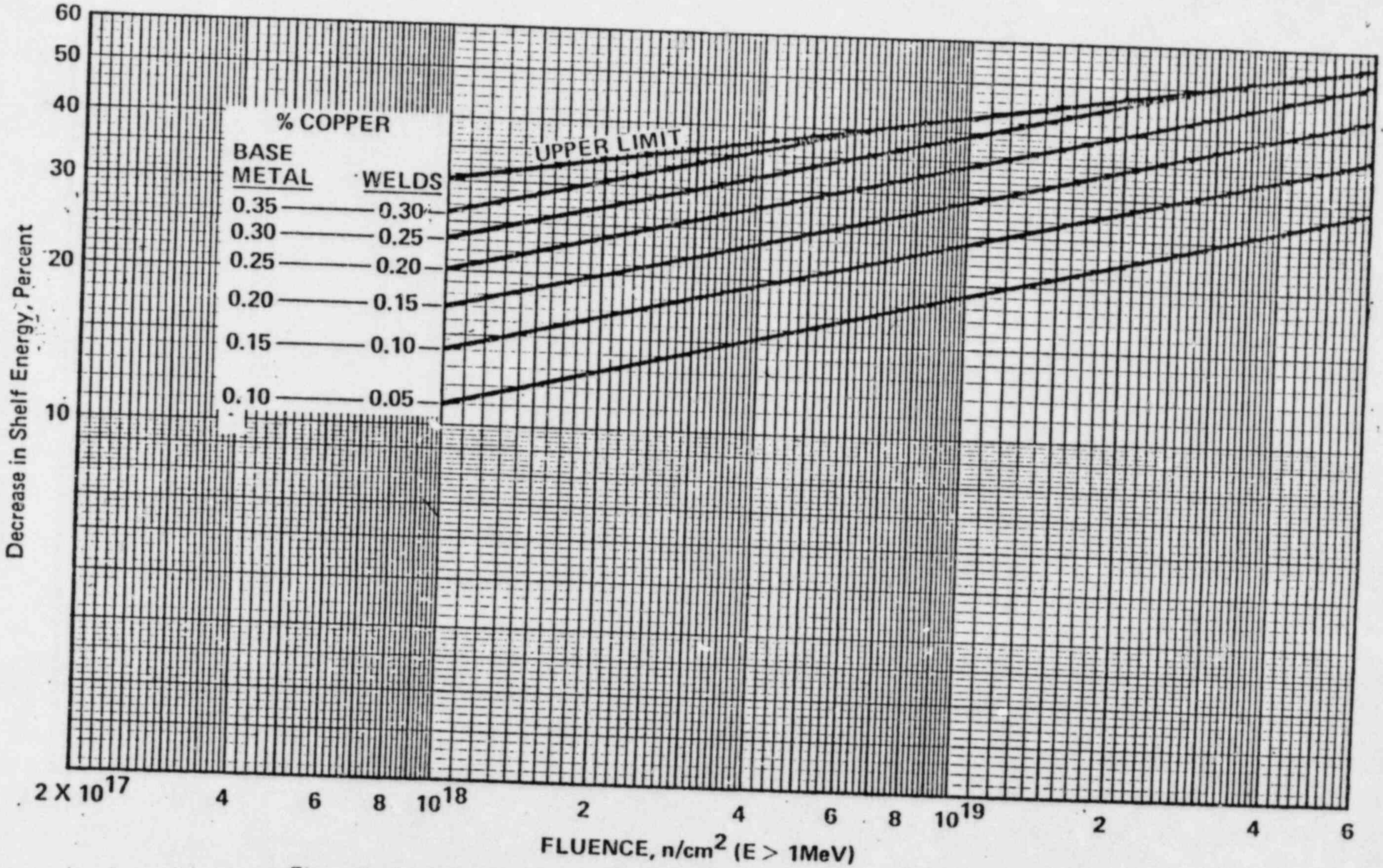

Figure 2 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence.