ECONOMIC CONSIDERATIONS IN THE DECOMMISSIONING OF LIGHT WATER REACTORS

by

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1.0 INTRODUCTION

The decommissioning or dismantling of a nuclear reactor is one of the tasks facing every holder of an AEC license for operation of an utilization facility. After the facility has completed its useful life, the licensee may apply to the AEC to surrender its license as covered in Section 50.82 of Title 10 of the Code of Federal Regulations (1), which outlines the requirements for termination of the license.

The major requirements include: 1) safe disposal of radioactive material, 2) decontamination of the site, 3) assuring that the dismantling and disposal of material will not be inimical to the common defense, and 5) providing that the dismantling will be accomplished in accordance with the rules and regulations of the commission.

This study will investigate the factors which affect the economics of decommissioning Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) power generating stations. In order to discuss the problems that are generic to all nuclear power stations, a hypothetical 1000 MW(e) PWR (Plant X) with initial operation in 1980 will be used as a basis for properly assessing the magnitude of the costs of the various dismantling options as well as the consequences of scheduling variations. Though each individual plant is confronted with unique problems when decommissioned, the economic factors which affect the decommissioning of a nuclear power generating facility are common to all plants.

2.0 DECOMMISSIONING - THE REASONS

2.1 Major Plants Decommissioned to Date

In the reactors dismantled to date, many factors have contributed to the decision to decommission the facility. In each case, one of the primary reasons or the deciding factor was the inability of the facility to be operated economically or to be justified on the basis of operating experience.

Of the reactors decommissioned to date, most were either experimental reactors or first or second generation demonstration reactors built and licensed under the AEC Reactor Demonstration Program. No reactors built with the primary purpose of generating electric power have been decommissioned to date.

The power reactors decommissioned to date, CVTR, Piqua, Elk River, Pathfinder, BONUS, and Hallam, had relatively small power outputs, and the reactors had either encountered technical difficulties or had completed their test programs.

The Boiling Nuclear Superheater (BONUS) Power Station was a joint project between the AEC and the Puerto Rico Water Resources Authority (PRWRA) who operated the reactor which was located on the western tip of Puerto Rico. The plant was a 50 MW(th) boiling water reactor with an internal superheater. The reactor was decommissioned because the "back-fitting of the safety systems to comply with the 27 design criteria (now 50 plus design criteria) of the AEC, poor economics, low availability, combined with the decrease in AEC interest in pursuing the Superheater Program resulted in a joint decision of AEC and PRWRA to terminate the contract... (2)." The decommissioning plan (3) included: 1) removal of all special nuclear material (SNM) from the site, 2) entombing the reactor pressure vessel, its internals and a limited quantity of highly contaminated or activated equipment, and 3) decontaminating those systems and areas external to the entombment. After the plant was decommissioned, PRWRA utilized the decommissioned reactor as an Exhibition Center where the general public could tour a nuclear power plant.

The Carolinas Virginia Tube Reactor (CVTR) was constructed under the provisions of the third round of power demonstration reactors by the Carolinas Virginia Nuclear Power Association (CVNPA) and the AEC. The reactor facility was located at Parr, South Carolina, approximately 25 miles northwest of Columbia. The CVTR was a 54 MW(th), D_2O - cooled and moderated pressure tube reactor. The experimental program of the CVTR was completed after approximately four years of operation and a decision to decommission the reactor was made (4). The CVTR decommissioning plan entailed: 1) shipping fuel and heavy water off-site, 2) disabling the reactor control drives, 3) storing radioactive equipment within concrete buildings and vaults, 4) establishing a double security barrier on all areas containing radioactive material, and 5) decontaminating certain areas of the site to allow unrestricted access to those areas. The plant reactor and containment buildings will be used for long time storage of the radioactive material under Byproduct Material License Number 39-08625-02 (5). This long time storage will require periodic inspection by CVNPA to assure that the material does not impose a hazard to the general public.

The Piqua Nuclear Power Facility (PNPF) was built by the AEC as part of the AEC Power Demonstration Program. The city of Piqua, Ohio operated the plant for the AEC. The steam generated by the plant was mixed with steam from other plants and was used both for process steam and to generate electricity. The reactor was a 45 MW(th) organic-moderated and cooled power demonstration reactor. When decomposition of the reactor coolant resulted in the formation of coke-like material in the reactor, distortion of certain reactor components occurred. The AEC reviewed the situation and concluded that the PNPF technology had limited potential use in advanced reactor concepts and decided to terminate operation of the facility (6). The PNPF Dismantlement Plan (7) was based on storing radioactive materials within the confines of the biological shield. All fuel, excess radioactive material, and special nuclear material were shipped off-site for disposal. The organic moderator was drained from the reactor and burned. The reactor vessel and vessel cavity, including radioactive material stored therein, were sealed within a waterproof membrane covered by a concrete slab to prevent water intrusion and personnel access. The remainder of the site was decontaminated and used as office and warehouse facilities by the city of Piqua.

The Hallam Nuclear Power Facility (HNPF) was a joint venture between the AEC and the Consumers Public Power District (CPPD) at the Sheldon Station near Hallam, Nebraska. The nuclear steam generating portion of the plant was owned by the AEC, while CPPD owned the conventional facilities of the site including the turbine generator. The reactor was a sodium-cooled, graphite moderated reactor with an output of 256-MW(th) (54-MW(e)). When moderator elements of the reactor became defective in 1964, the reactor was shutdown, and the AEC decided to decommission the reactor. The decommissioning of the Hallam facility was based on the HNPF Retirement Plan (8).

The plan included: 1) removal of all fuel from the site, 2) removal of all bulk sodium, 3) reaction of residual sodium to insure that it was *essentially inert*, 4) shipment of radioactive materials for off-site burial or storage of wastes within three isolation areas, 5) decontamination of accessable areas, and 6) demolishment and removal of all reactor complex buildings from the site. The radioactive material is entombed within the three isolation areas; the reactor vessel, one of the fuel storage pits, and one of the moderator element storage cells. The site, though not abandoned, does not require surveillance, and CPPD has been released from all licensing obligations.

The Pathfinder Reactor located at Sioux Falls, South Dakota was owned and operated by the Northern States Power Company (NSP). It was the first power reactor to be decommissioned which was not owned at least partially by the AEC. The Pathfinder Reactor was a 50 MW(e) direct cycle boiling water reactor with an integral nuclear superheater. When the reactor reached initial full power, it was discovered tha the steam separators which were located around the periphery of the core had failed and would have to be replaced. Due to the small electrical output of the facility and the cost of repairing the reactor, NSP decided to replace the nuclear steam system with three package boilers and use the fossil fueled boilers to drive the turbine (9).

After the plant was converted to fossil operation, NSP decided to dismantle the facility to the extent where it was no longer necessary to maintain a Part 50 license on the facility. A decommissioning plan was generated (10) which included: 1) The reactor building and lower levels of the fuel handling building were sealed. 2) All unused systems within the fossil plant were removed. 3) The reactor was deactivated by disabling the control rod drives and filling the reactor with gravel. 4) The isolation area penetrations were sealed. 5) The fuel storage pool was sealed by a reinforced concrete slab. 6) The portions of the facility which were accessible were decontaminated to allow unrestricted access. The facility will serve as a storage site for radioactivity under a Part 30 license and NSP will conduct periodic inspections to insure that the storage areas remain safe (11).

The Elk River Reactor (ERR) Power Station was built by the AEC at the town of Elk River, Minnesota, The reactor was owned by the AEC and operated by the Rural Cooperative Power Association (RCPA) (now named the United Power Association (UPA)). The ERR was an indirect cycle boiling water reactor with separate fossil fuel superheater with a total thermal output of 73 MW(th) (58.2 MW(th) for the reactor). Upon expiration of the operating license between the AEC and RCPA, RCPA waived its option to purchase the plant, and the AEC was required by the terms of the contract to make the site useable without undue danger to the public health and safety. The dismantling plan required that the AEC dismantle the ERR and remove the entire facility to approximately one foot below grade level. The resulting cavities were to be filled with clean earth or with clean rubble topped with clean earth (12,13). When the dismantling is complete, the site will be returned to its original condition except for the subgrade foundations which will be left in place. All radioactivity will be removed from the site and no license or monitoring will be required.

There are presently plans underway to decommission other reactors including the Saxton Nuclear Power Facility (14) and Peach Bottom Unit #1 (15). The Marviken reactor in Sweden was decommissioned due to control stability problems and the containment was used for safety testing (16). It was the highest output reactor decommissioned so far with a 140 MW(e) generating capability.

2.2 Economic Incentives for Decommissioning

The operation of any utility power system is based on the premise that the newer plants with lower incremental power cost will be operated first or will displace older plants in the system base load. The more efficient plants will be loaded prior to the "older" and hence less efficient plants. This premise is offset by such factors as plants designed specifically for peaking with rapid start-up capabilities and the need to assure that the system is capable of adequate response to emergency outages, unscheduled maintenance requirements, refueling outages, and other nontypical conditions. Due to the need to maintain adequate power under all circumstances, utilities assess a capacity charge credit on the capability of each generating unit on the grid. As a result, older units which might be shutdown and demolished are left in standby status to be brought on line as needed. Typically on a fossil unit, standby status requires only a skeleton operating and maintenance crew and as much fuel as might be required by any possible contingency. This standby philosophy typical of fossil units encounters problems when applied to a nuclear unit. At the end of the forty year operating life of a nuclear reactor, several factors will influence the decision of whether to decommission the reactor or let it remain in standby status.

One factor is the cost of maintaining an adequate staff at a nuclear facility. 10 CFR 55 (1) deals specifically with the licensing of reactor operators. As the number of nuclear plants increases, the shortage of trained operators, engineers, technicians, and health physics personnel will increase (17,18). The cost penalty of maintaining a complete staff at a standby plant due to AEC regulations, when these personnel could be used in new plants, is significant. An example of the stringent requirements for staffing nuclear plants is given in Regulatory Guide 8.8 issued by the AEC in July 1973 (53) and titled "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactor)." It provides minimum guidelines on the requirements of reactor health physics staffs. Regulatory Guide 8.8 suggests that the health physics chief at a reactor be capable of being certified by the American Board of Health Physics (ABHP) and their requirements for certification are stringent (19). The staff at any reactor will require many highly trained professionals and personnel with adequate qualifications will be in high demand.

The fuel costs of a nuclear facility are based on more of a capital fixed charge rate than on fuel costs as encountered in fossil plants. Typically, the utility will purchase yellow cake and will pay for enrichment, conversion to uranium oxide, and fuel fabrication. This requires a large investment prior to loading the fuel in the reactor. Typically fuel contracts (including first replacement cores) let now for 1980 units (1000 MW(e)) may require fuel cost investments of $32,000,000 (20). This is based on yellow cake costs of $18,000,000, separative work charges of $7,000,000 ($36/SWU), and fabrication charges of $7,000,000. Fixed charges on this investment at 15% will run near $5,000,000 annually. A standby plant not generating power is thereby penalized by the "use" charges involved with the fuel.

The operating and maintenance costs of a standby nuclear unit will not be significantly different from the costs of an operating plant. The maintenance costs will increase as equipment wears out with age. The costs of repairing radioactive components and equipment is higher than costs associated with repairing noncontaminated equipment. In addition to AEC requirements on staffing, operator training, and health physics monitoring, the AEC through license Technical Specifications ("Tech Specs") and other regulations requires continuous monitoring of the facility status. In addition, the AEC annual licensing fee, though an insignificant cost to an operating plant, becomes a cost penalty based on unit capacity if the unit is not generating power.

In perspective, even if the nuclear unit is shutdown and the fuel unloaded, the AEC regulations and license requirements will remain in effect until the unit is decommissioned or until the license is modified to allow "possession only" status with appropriate modifications to the Tech Specs. Therefore, the licensee has an incentive to decommission the facility as soon as it is uneconomical to operate. This assumes that due consideration is given the capacity charge credit if the unit is capable of operation when needed.

2.3 The Decommissioning's Objective

The objects of the decommissioning are to: 1) alter the reactor so it is incapable of generating special nuclear material (SNM) which might be inimical to the public safety, 2) dispose of the radioactive material located on the site or render it harmless so it is incapable of harming the public health, 3) minimize license requirements consistent with items 1) and 2), 4) free qualified personnel for work at other facilities, 5) minimize future facility surveillance requirements and costs, and 6) leave the facility in a safe condition with consideration given to such possible factors as vandalism, acts of terrorism, and natural disasters such as earthquakes, floods, and tornadoes.

These objectives are constrained by the overriding requirement that they be accomplished as economically as possible. Furthermore, it is desirable to handle the decommissioning in a manner so that it will not cause public opposition.

3.0 STATUS OF THE PRE-DECOMMISSIONED FACILITY

3.1 The Physical Site-PWR

Pressurized water reactor (PWR) nuclear steam supply systems (NSSS) are presently marketed in the United States by Westinghouse Electric Corporation, Combustion Engineering, Inc., and the Babcock and Wilcox Company. Though each vendor's design is unique in some respects, the designs have common or generic systems. A typical PWR site plan is shown on Figure 1, while Figure 2 shows a section through the reactor building.

The PWR plant site shown in Figure 1 is based on the Middletown Nuclear Power Generating Station, a hypothetical PWR used in WASH - 1230 (45) to generate PWR construction costs. Modifications to the site structures have been made to include recent design guides but the structure components remain essentially as listed in WASH - 1230. Figures 3, 4, and 5 show details of the Calvert Cliffs Nuclear Power Plant. Though individual stations may include more or fewer buildings and a different arrangement, certain characteristics are common to all. Five buildings common to all are the reactor containment, the turbine generator building, the control building, the primary auxiliary building, and the fuel handling building. Other buildings may include the feedwater heater building, the diesel generator building, the service and maintenance building, and the waste treatment building. PWR's are indirect cycle units, thus radioactive contamination of the feedwater and main steam systems occur only when leakage takes place in one of the steam generators. The turbine generator and the steam and condensate systems therefore are not constrained by radiation protection or contamination considerations.

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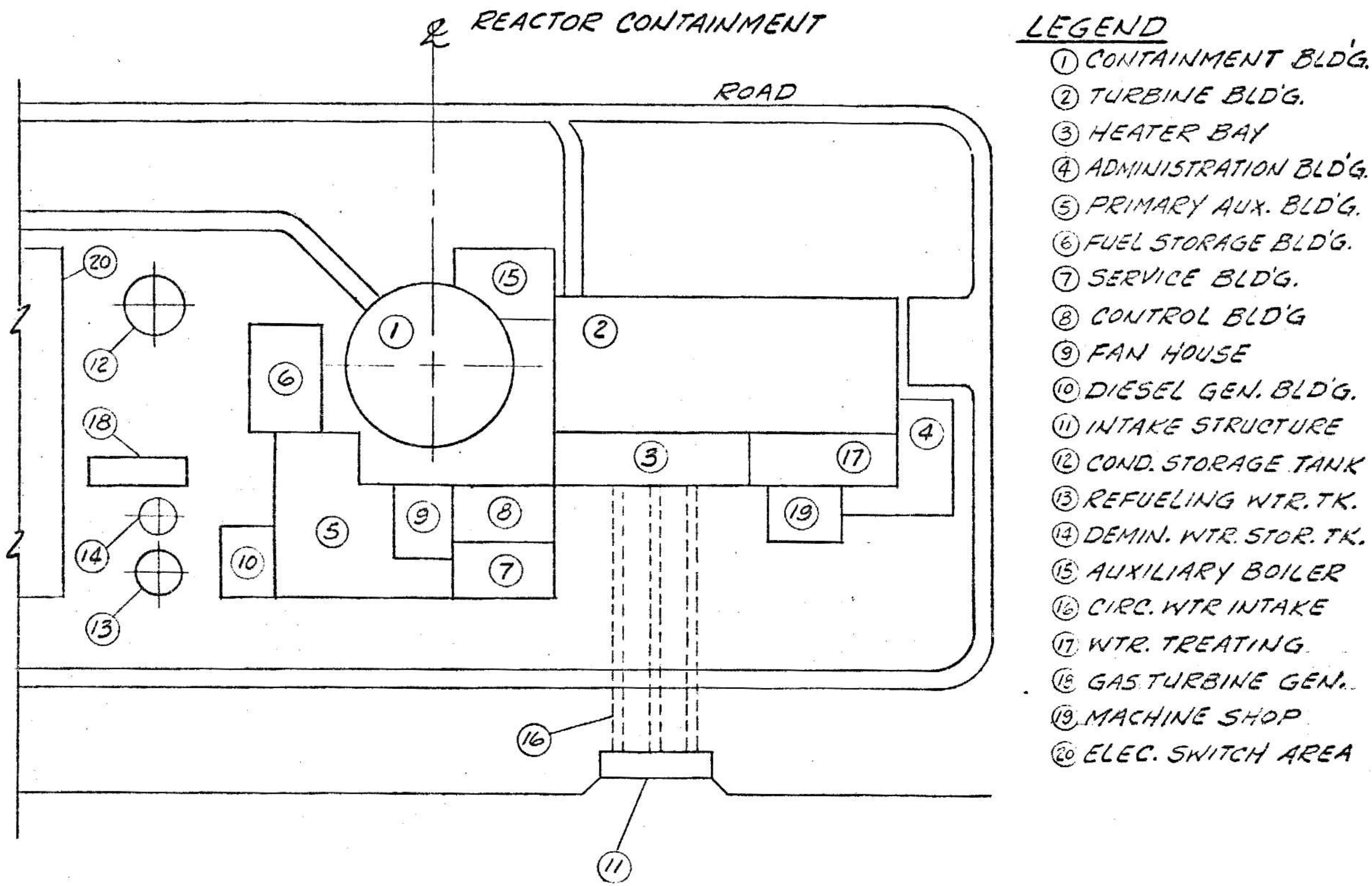


FIGURE 1 - Site Layout of Hypothetical PWR - Plant X

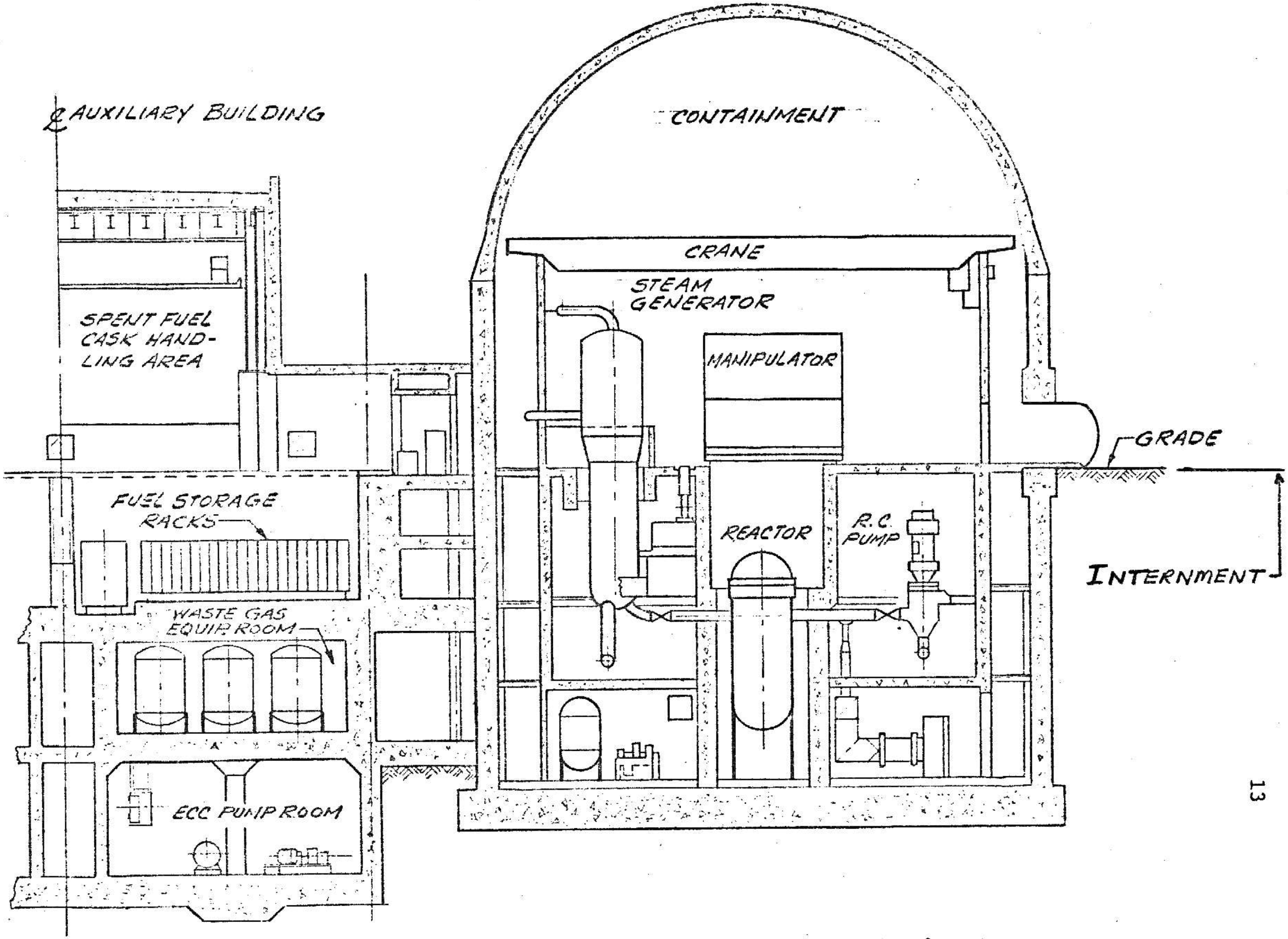
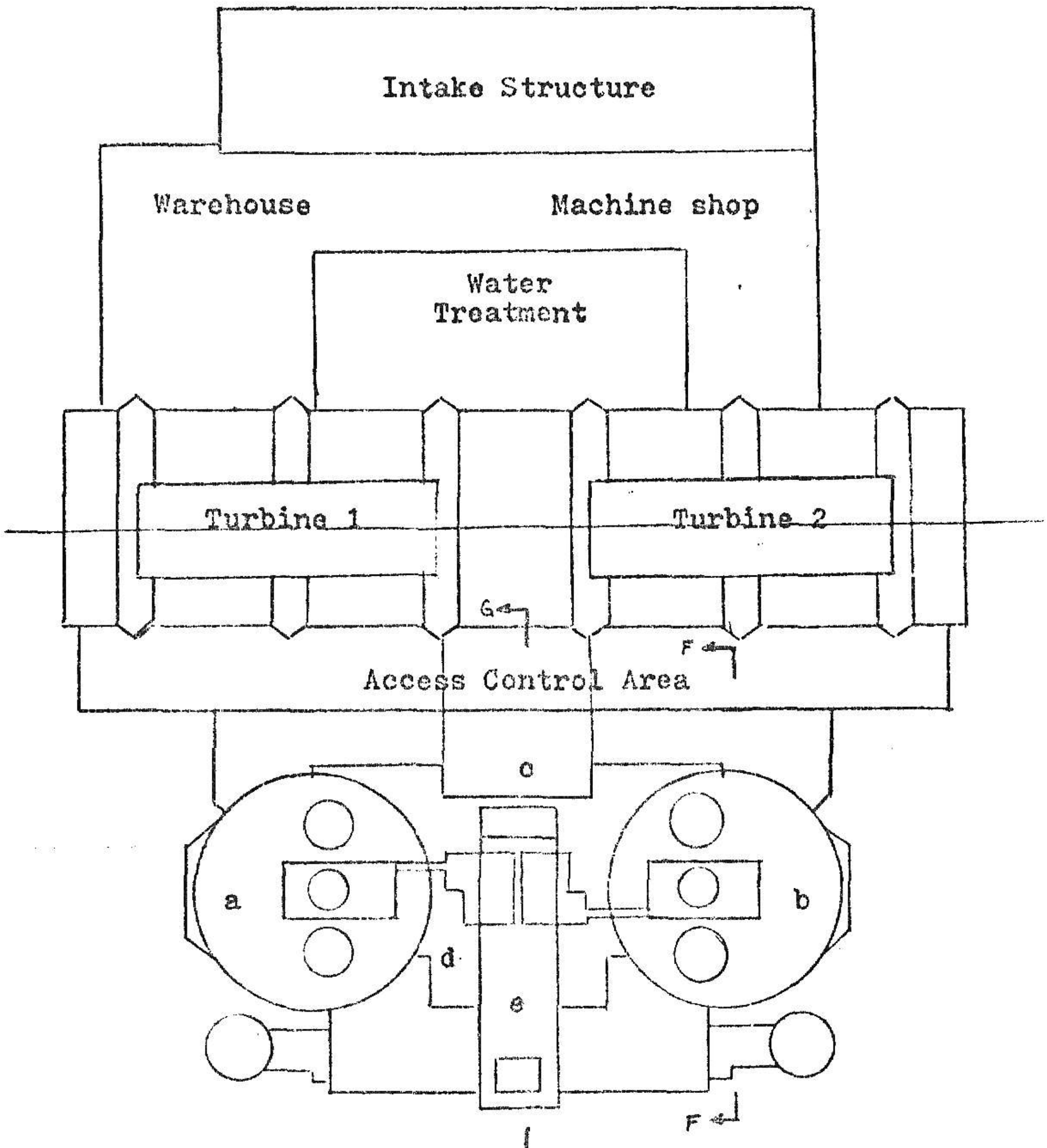


FIGURE 2 - Reactor Building - Plant X - Showing Internment Areas



a. Containment 1
b. Containment 2
c. Control Room
d. Auxiliary Building
e. Fuel Handling Area

FIGURE 3 - Plan of Calvert Cliffs PWR's Showing Major Building Layout

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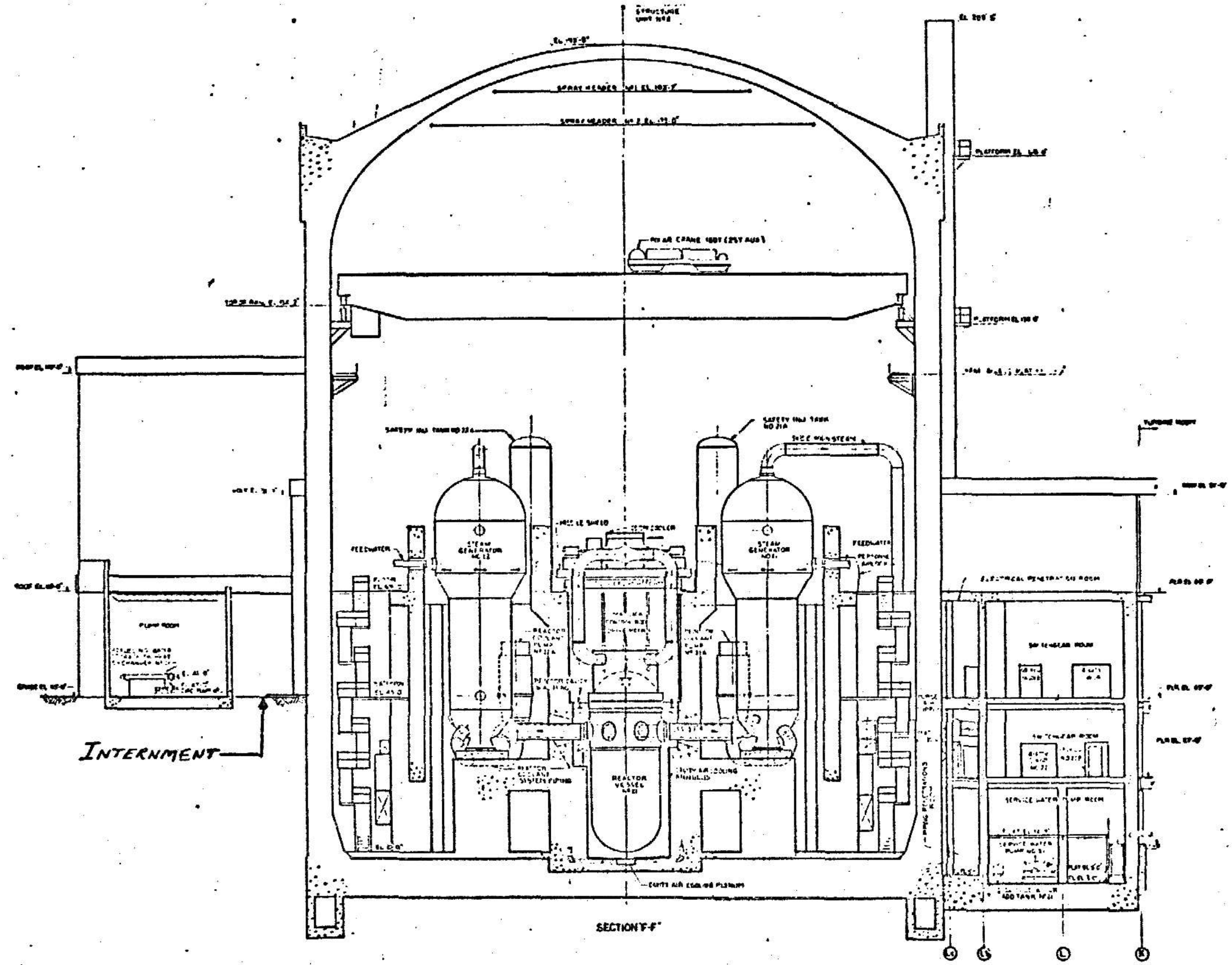

FIGURE 4 - Section of Calvert Cliffs Containment - Unit 2 - Showing Areas Available for Storage or Interment Dismantlings

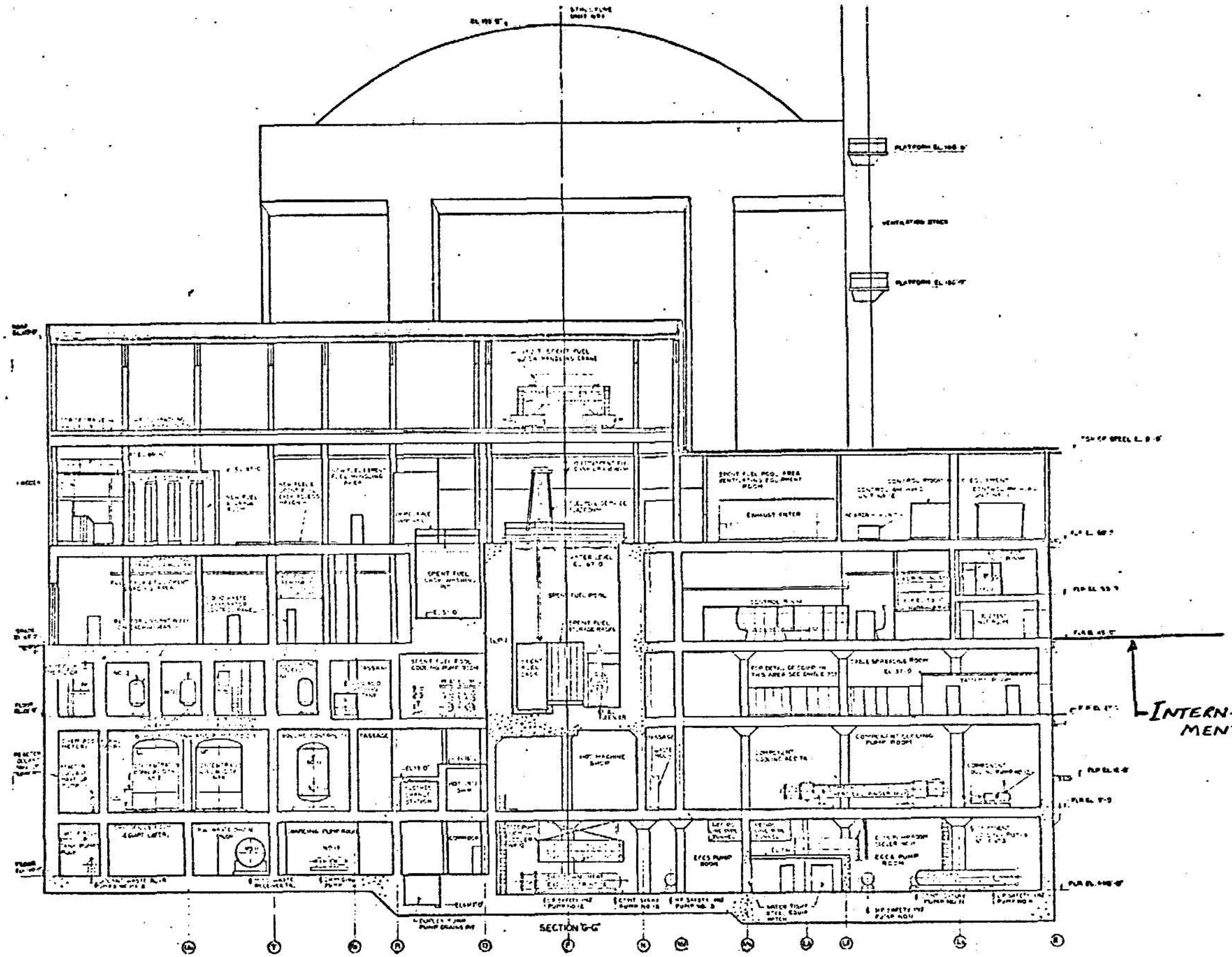

FIGURE 5 - Section of Calvert Cliffs Auxiliary Building Showing Areas Available for Storage or Internment

The auxiliary systems common to the PWR NSSS include, in addition to the primary system, the Chemical and Volume Control Systems (CVCS), the Safety Injection Systems (SIS) (which includes the various ECC systems), the Residual Heat Removal System (RHR), the Spent Fuel Storage and Cooling System (SFSCS), the Containment Cooling Systems, the Closed-Loop Cooling Systems, the Service Water Systems, the Ventilation Systems, and the Rad Waste Systems (21). The remaining systems are similar to those found in a conventional power plant and include Condensate System, Feedwater System, Extraction Steam System, Heater Drip System, Bearing Water System, Plant Air System, etc. A typical power generating station may have as many as 60 distinct systems to provide adequate operating conditions.

At the end of 40 years of operation, many of the plant components will be worn out and will be suffering from the effects of fatigue, corrosion, and erosion. They will require considerably more maintenance than comparable new equipment would require. For the components presently being manufactured under the rigid Section III requirements of the ASME Code, there is no valid information as to reliability at the end of life. Any conjecture about significant improvements in reliability and decreased maintenance requirements is not supported by data.

3.2 Radiological Status of the Site-PWR

The continuous operation of a nuclear plant over 40 years will result in the generation of large amounts of radioactive materials. In addition to the fission products which will escape from penetrations in the fuel cladding, the primary loop will be contaiminated from fission products generated from "tramp uranium" located outside the fuel cladding. Activation of the reactor vessel, thermal shield, fuel support structures, and biological shield will result from neutron capture. In addition to the activity induced in the reactor structural components, large amounts of corrosion products suspended in the primary coolant will be activated when they pass through the core. These corrosion products are typically referred to as "crud." The crud will circulate through the various reactor systems (the CVCS, RHR, SFSCS, rad waste systems, etc.) and will deposit in stagnant areas of the systems. The corrosion products will also plate out on the system's internal surfaces. The result of crud deposition is activation of all the reactor auxiliary systems. This activation may increase to the point where radiation levels are dangerous to operating personnel. A plant maintenance program which includes periodic decontamination of the contaminated systems coupled with adequate design of the reactor systems should minimize the exposure of operating personnel.

A general summary of the hypothetical specific inventory for Plant X, a hypothetical PWR, is shown on Table I. This summary is based on the plants decommissioned to date (3,4,10) and on calculations of the activities resulting from a 40 year operating life. Factors such as decontamination schedules and system components (component material and corrosion resistance) will have a large effect on the radioactive isotope inventory at the end of a plant's life. Table II lists the long-lived nuclides of importance in activation of the reactor vessel and components. In order to calculate the activation of the reactor components listed in Table I the Unit Operating History shown in Figure 6 was assumed. (The calculational procedure is given in Appendix A.)

TABLE I HYPOTHETICAL INVENTORY OF PLANT X
AFTER 40 YEARS OPERATION

		Curies
1.	Reactor	
	a. Lower Support Plate	1.96×10^7
	b. Upper Support Plate	1.65×10^6
	c. Core Shroud	3.04×10^7
	d. Thermal Shield	3.35×10^6
	e. Core Barrel	6.61×10^6
	f. Vessel and Accessories	3.1×10^6
2.	Primary Coolant Loop	1.6×10^5
3.	CVCS	20
4.	Safety Injection System	1
5.	Residual Heat Removal System	2
6.	Spent Fuel Cooling System	2
7.	Rad Waste System	55
8.	Spent Fuel Storage Pool	40
9.	Concrete	1500
10.	Miscellaneous	10

Refer to Appendix E for basis of inventory calculations.

The crud circulated in a reactor is principally spinel R_3O_4, where the R may be iron, nickel, or chromium. Crud formation in a large reactor may run as high as 50 to 100 g/day (22). Table II includes the half-lives of the long-lived radioactive isotopes found in the crud. The ^{60}Co isotope is the most critical due to its long half-life (5.3 years). Immediately after startup or shutdown, the short-lived isotopes (^{51}Cr, ^{59}Fe, ^{58}Co, etc.) contribute significantly to the total activity in the system. In addition to the activation of corrosion products, the PWR can be contaminated by activation of the primary water and the pH control reagents. The behavior of contamination, both crud and fission product, is treated in depth by Berry (22), Ayres (23), and Cohen (24). A typical analysis of crud is given in Table III.

The radiation levels generated in a nuclear power generating station are controlled by many interrelated parameters. Primary system corrosion resistance, the operation of the CVCS, the frequency and success of decontamination attempts and the design of the particular facility. Table IV lists the typical radiation levels one might expect to encounter in a generating station at the end of 40 years (Plant X).

The decontamination of any system has the primary goal of reducing the radioactivity and is assessed by the Decontamination Factor (DF) defined as the ratio of initial to final radioactivity. The procedures and chemicals used for decontamination are in general modifications of those used for pickling or descaling. Typical decontamination solutions and penetration rates are given in Table V. The decontamination of unit systems can result in DF's which vary from 1.5 to 60 (23) depending on the nature of the contamination.

TABLE II LONG-LIVED RADIONUCLIDES CONTRIBUTING SIGNIFICANTLY TO ACTIVATION

Nuclide	Half-Life, yr.	Decay Constant, $yr.^{-1}$	Parent Nuclides and Reactions
^{113m}Sn	0.323	2.145	$^{112}Sn(n,\gamma)$
			$^{114}Sn(n,2n)$
^{123}Sn	0.353	1.962	$^{122}Sn(n,\gamma)$
^{45}Ca	0.446	1.553	$^{44}Ca(n,\gamma)$
			$^{46}Ca(n,2n)$
^{119}Sn	0.685	1.018	$^{118}Sn(n,\gamma)$
			$^{120}Sn(n,2n)$
			$^{119}Sn(n,p)$
^{110}Ag	0.712	0.974	$^{109}Ag(n,\gamma)$
^{57}Co *	0.7447	0.931	$^{58}Ni(n,2n)$
			$^{58}Ni(n,d)$
^{54}Mn *	0.854	0.812	$^{55}Mn(n,2n)$
			$^{54}Fe(n,p)$
^{49}V	0.904	0.767	$^{50}Cr(n,2n)$
			$^{50}Cr(n,d)$
^{109}Cd	1.242	0.558	$^{112}Sn(n,\alpha)$
^{55}Fe *	2.4	0.289	$^{54}Fe(n,\gamma)$
			$^{56}Fe(n,2n)$
			$^{58}Ni(n,\alpha)$
^{22}Na	2.6	0.267	$^{23}Na(n,2n)$
^{125}Sb	2.7	0.257	$^{124}Sn(n,\gamma)$
^{60}Co *	5.24	0.132	$^{59}Co(n,\gamma)$
			$^{60}Ni(n,p)$
			$^{61}Ni(n,d)$
^{3}H	12.26	0.0565	$^{2}H(n,\gamma)$
^{113}Cd	14	0.0495	$^{116}Sn(n,\alpha)$
^{121}Sn	25	0.0277	$^{120}Sn(n,\gamma)$
^{63}Ni *	92	0.00753	$^{62}Ni(n,\gamma)$
			$^{64}Ni(n,2n)$
			$^{64}Ni(n,d)$

TABLE II (Continued)

Nuclide	Half-Life yr.	Decay Constant, yr.$^{-1}$	Parent Nuclides and Reactions
^{108m}Ag	100.	0.00693	$^{107}Ag(n,\gamma)$
			$^{109}Ag(n,2n)$
^{39}Ar	270.	0.00257	$^{42}Ca(n,\alpha)$
			$^{39}K(n,p)$
^{14}C	5730.	0.000121	$^{13}C(n,\gamma)$
			$^{17}O(n,\alpha)$

*Isotopes commonly found in the crud.

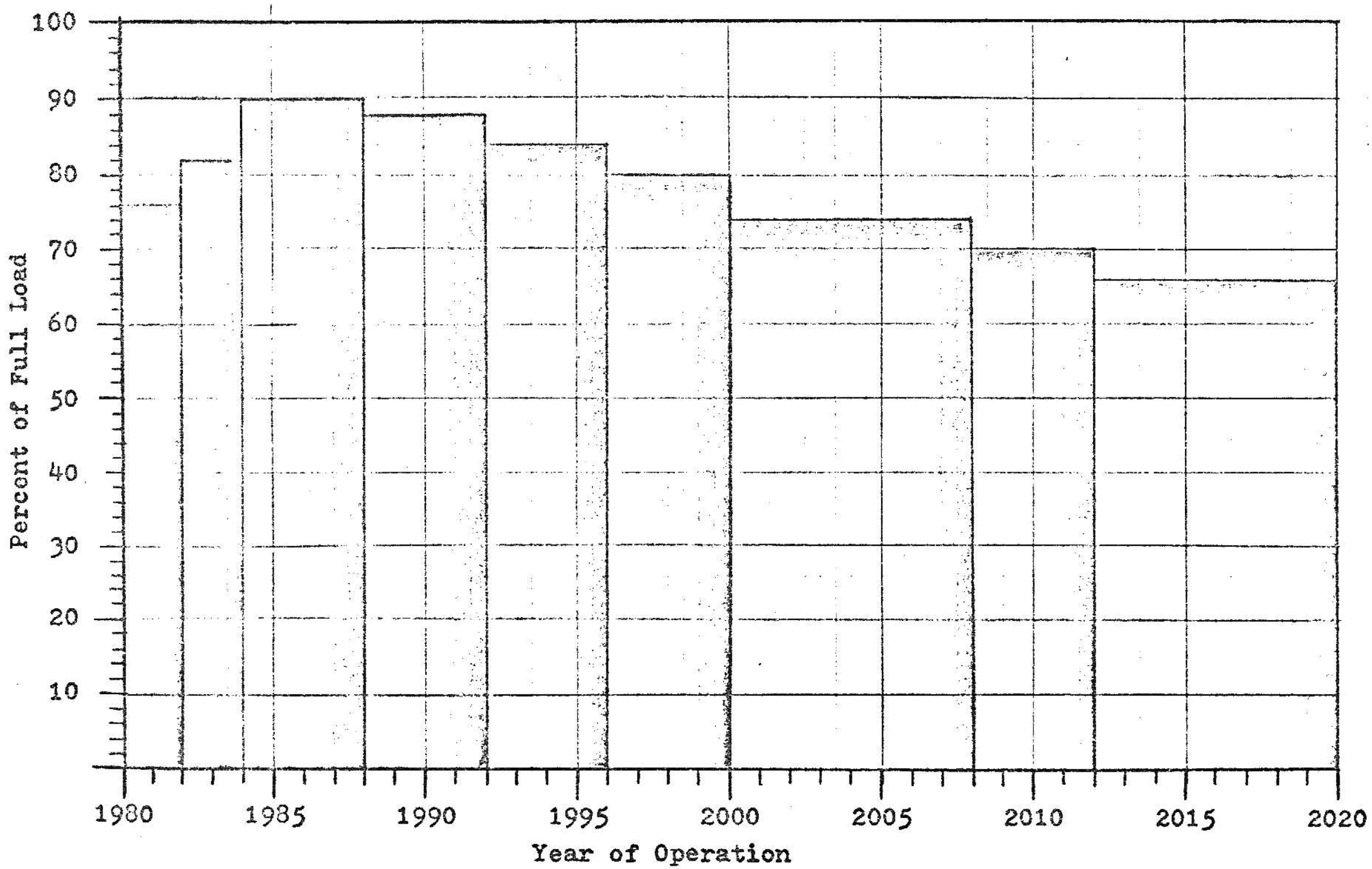

FIGURE 6 - Load Model in Percent of Total Capacity for Hypothetical Plant X

TABLE III TYPICAL CRUD ANALYSIS - PWR/BWR

Element	Percent of Sample - Chemical Composition of Oxide		
	A	B	C
Fe	68.1	43.1	49.9
Ni	9.9	7.10	22.6
Cr	0.88	1.57	4.9
Co	0.275	0.0420	13.3
Mn	0.51	0.21	
Nb	0.017		
Ag	0.0028	0.0330	
In	0.099	0.0420	
Cd	0.008	<0.0150	
B	0.0010	0.0080	5.8

A. Yankee Core I fuel assembly corrosion product deposit (22).

B. Yankee main coolant circulating crud chemical composition (22).

C. Chemical composition of crud in Dresden Reactor Steam generator (22).

TABLE IV HYPOTHETICAL RADIATION LEVELS IN PLANT X AFTER 40 YEARS OPERATION

	Area	Dose Rate (mR/hr)*
1.	Operating Floor of Containment	12.5 - 60.0
2.	Steam Generator	30.0 - 80.0
3.	Reactor Coolant Pump	50.0 - 175.0
4.	Pressurizer Quench Tank Area	24.0 - 120.0
5.	Spent Fuel Pool Service Platform	12.0 - 30.0
6.	Spent Fuel Pool Cooling System Pump Room	35.0 - 120.0
7.	Volume Control Tank	20.0 - 100.0
8.	Shut-down Heat Exchangers	15.0 - 50.0
9.	Waste Gas Decay Tanks	15.0 - 120.0
10.	Radioactive Laundry Tank	25.0 - 70.0
11.	Residual Heat Removal Heat Exchangers	15.0 - 50.0
12.	Liquid Rad-Waste System	40.0 - 2,500.0

*Dose rates are constant doses from various pieces of equipment. In general area dose rates will be at least a factor of four lower than contact dose rates. Hot spot readings may exceed the contact dose range by a factor of 100. (46,47,54,55) were used to generate relative dose levels.

TABLE V CORROSION RATES IN REACTOR DECONTAMINATING SOLUTIONS (22)

	Penetration rate, mils, for solution indicated				
Steel	APAC[(a)]	APAC[(b)]	$CrSO_4$[(c)]	ABF, KAP[(d)]	APACE[(e)]
AlSl 4340	—	—	—	4.4	—
ASTM A212 Grade B	0.12	—	—	—	—
AlSl Type 410	—	0.018	—	0.34	0.008
AlSl Type 302	—	—	—	—	0.005
AlSl Type 302, Sensitized	—	—	—	—	0.015
AlSl Type 304	0.0014	0.0017	—	0.015	0.002
AlSl Type 304 Sensitized	0.0047	—	—	0.045	0.002
AlSl Type 304L	—	—	0.16	—	0.003
AlSl Type 308, Sensitized	0.0048	—	—	—	—
AlSl Type 316	—	0.0016	0.02	—	—
AlSl Type 318	—	—	0.02	—	—
AlSl Type 321	—	—	0.06	—	0.002
AlSl Type 347	—	0.0016	0.13	0.018	—
Inconel 600	—	0.001	—	0.009	—

(a) 2 hr in 10 wt percent NaOH - 3 wt percent $KMnO_4$ at 103° C plus 2 hr in 10 weight percent dibasic ammonium citrate at 98° C. (APAC Method - alkaline permanganate-ammonium citrate.)

(b) 30 min in 18 wt percent NaOH - 3 wt percent $KMnO_4$ at 105° C plus 15 min in 10 weight percent ammonium citrate at 100° C (APAC Method.)

(c) 0.32 M $CrSO_4$ - 0.65 M H_2SO_4 at 85° C circulated for 4.4 hr. ($CrSO_4$ Method.)

(d) 3 to 5 hr in 1 wt percent ammonium bifluoride - potassium acid pthalate at 60° to 77° C. (ABF-KAP Method.)

(e) 2 hr in 18 wt percent NaOH - 3 wt percent $KMnO_4$ at 105° C plus 2 hr in 12 oz/gal ammonium citrate - 0.7 oz/gal 1 phenyl-2 thiourea circulated at 85° C. (APACE Method.)

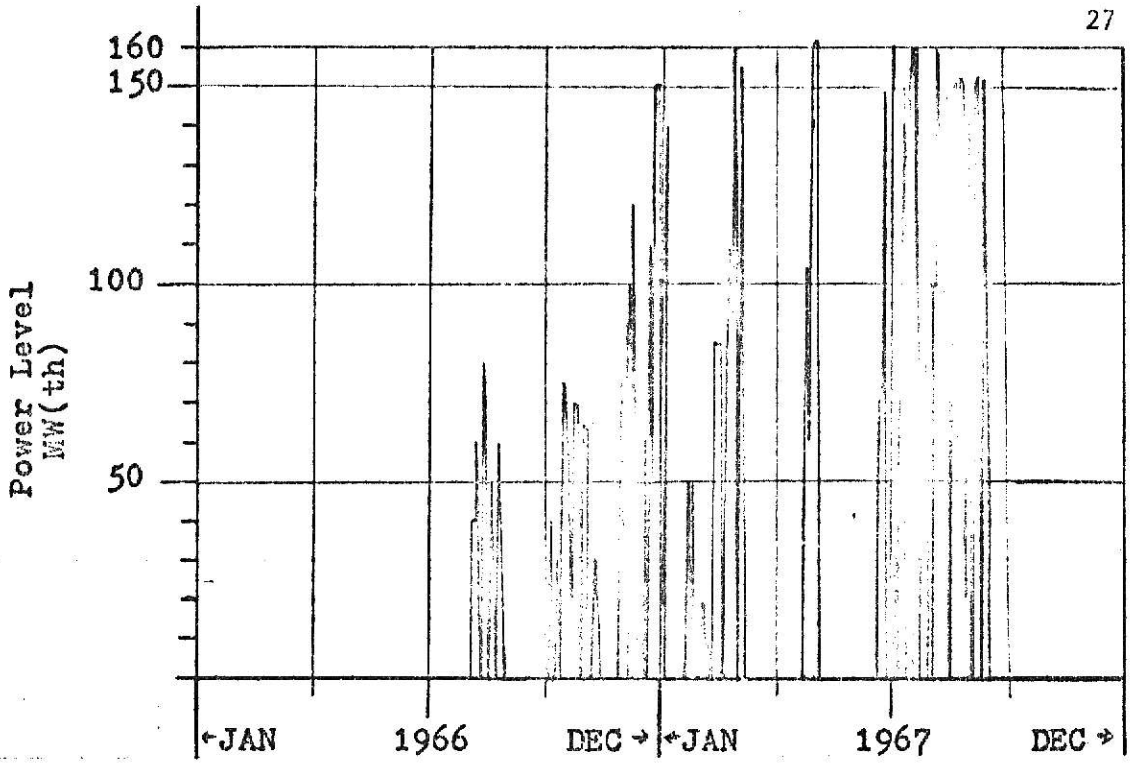

FIGURE 7 - Outline of Operating History - Pathfinder Reactor

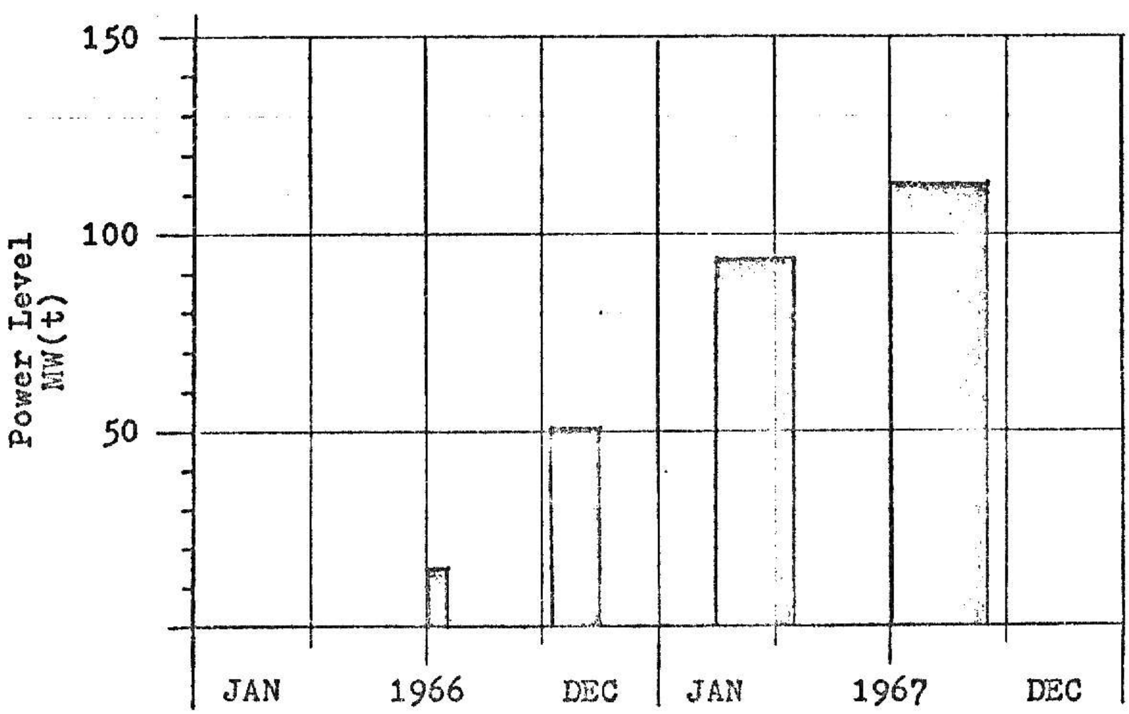

FIGURE 8 - Reactor History Used to Calculate Reactor Components Radioactivity Inventory - Pathfinder Reactor

The calculation of induced activity in the reactor vessel and components contains several conservative assumptions. In general, peak flux values are used for average fluxes throughout each component. Figure 7 shows the operating history of the Pathfinder Reactor and the model used to calculate the activity is shown in Figure 8.

The total megawatt hours of operation of the Pathfinder plant were used as the basis for generating Figure 8. The hours were broken into four groups to minimize computer time while still evaluating the relative decay of the radioactive nuclides formed. Using the model of Figure 8 and the program contained in Appendix B, the inventory of the Pathfinder reactor vessel and internals was checked to 15 months after the reactor was shutdown. The results of the calculation were compared to data NSP generated by physically measuring radiation levels. Of the eight components inventories checked, the activities calculated were found to be within a factor of four of the activities measured (25). The same type calculations were made for the ERR plant. In the ERR calculations, the vessel and internals were divided into discrete zones to generate a more accurate estimate of the activity (12). Appendix G contains a comparison of the calculated and measured radiation levels in the ERR. The calculated activities of the major radionuclides found in the reactor vessel, the steam separators and the grid plate of the Pathfinder Plant are given in Figures 9, 10, and 11. If Pathfinder had operated for 40 years, the activities would have been as shown on Figures 12, 13, and 14. The high content of Ni-63 in the grid plate and steam separators will result in a rather long-lived high activity. Nickel-63 decays by emitting a .077 MeV beta particle (92 year half-life) and as such it is only harmful when ingested. Though the total activity of the

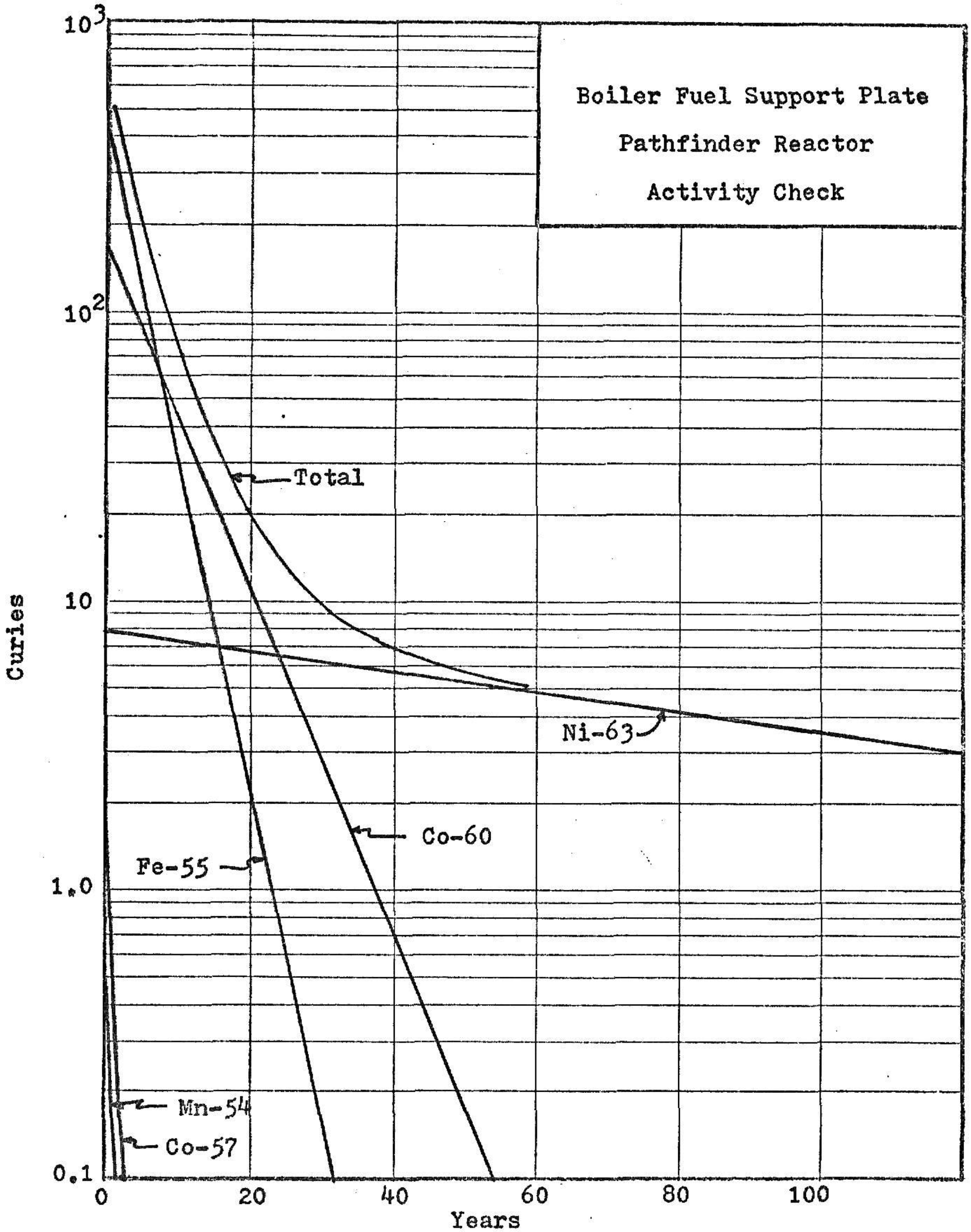

FIGURE 9 - Activity and Decay Calculated for Boiler Fuel Support Plant - Pathfinder Reactor

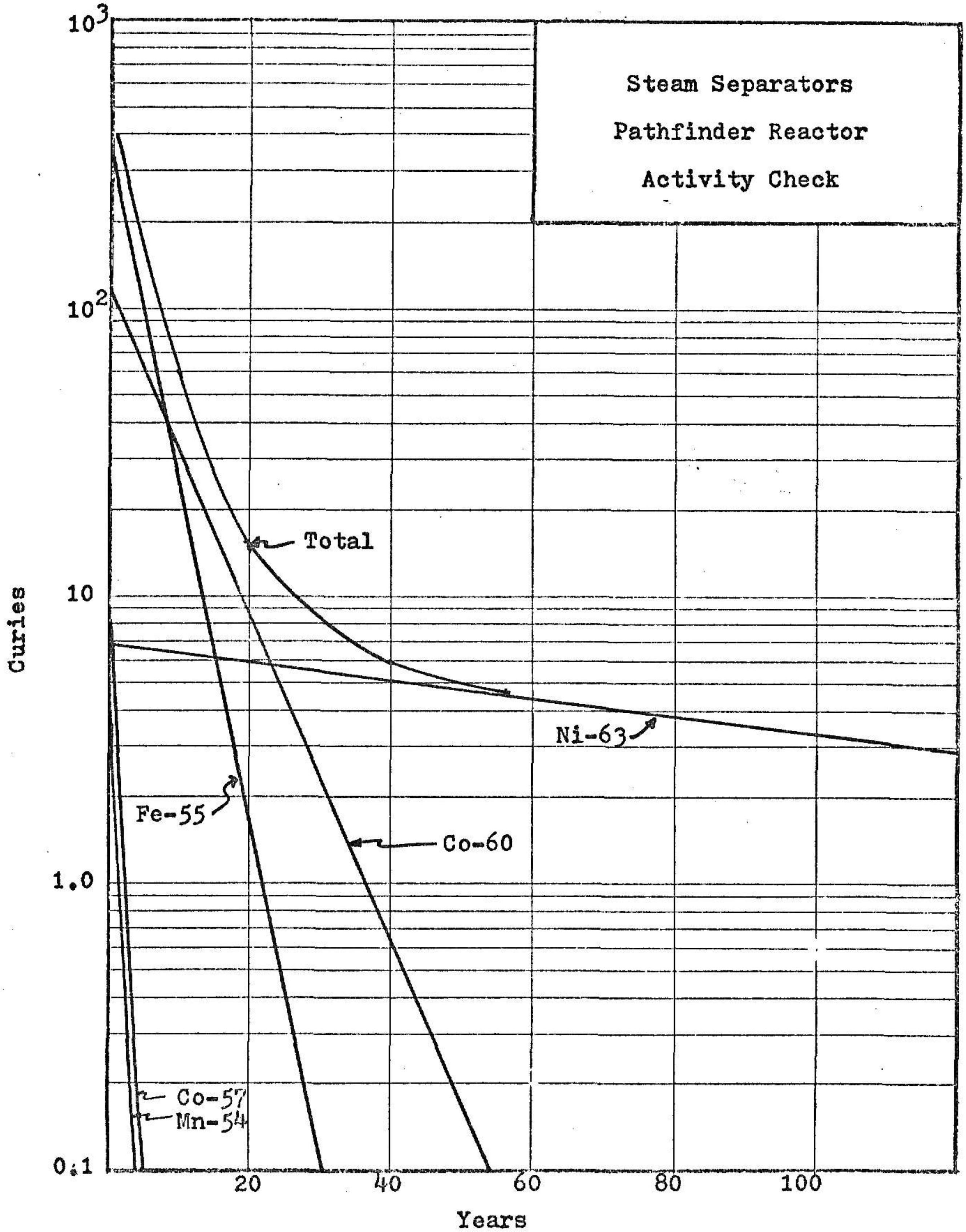

FIGURE 10 - Activity and Decay Calculated for Steam Separators - Pathfinder Reactor

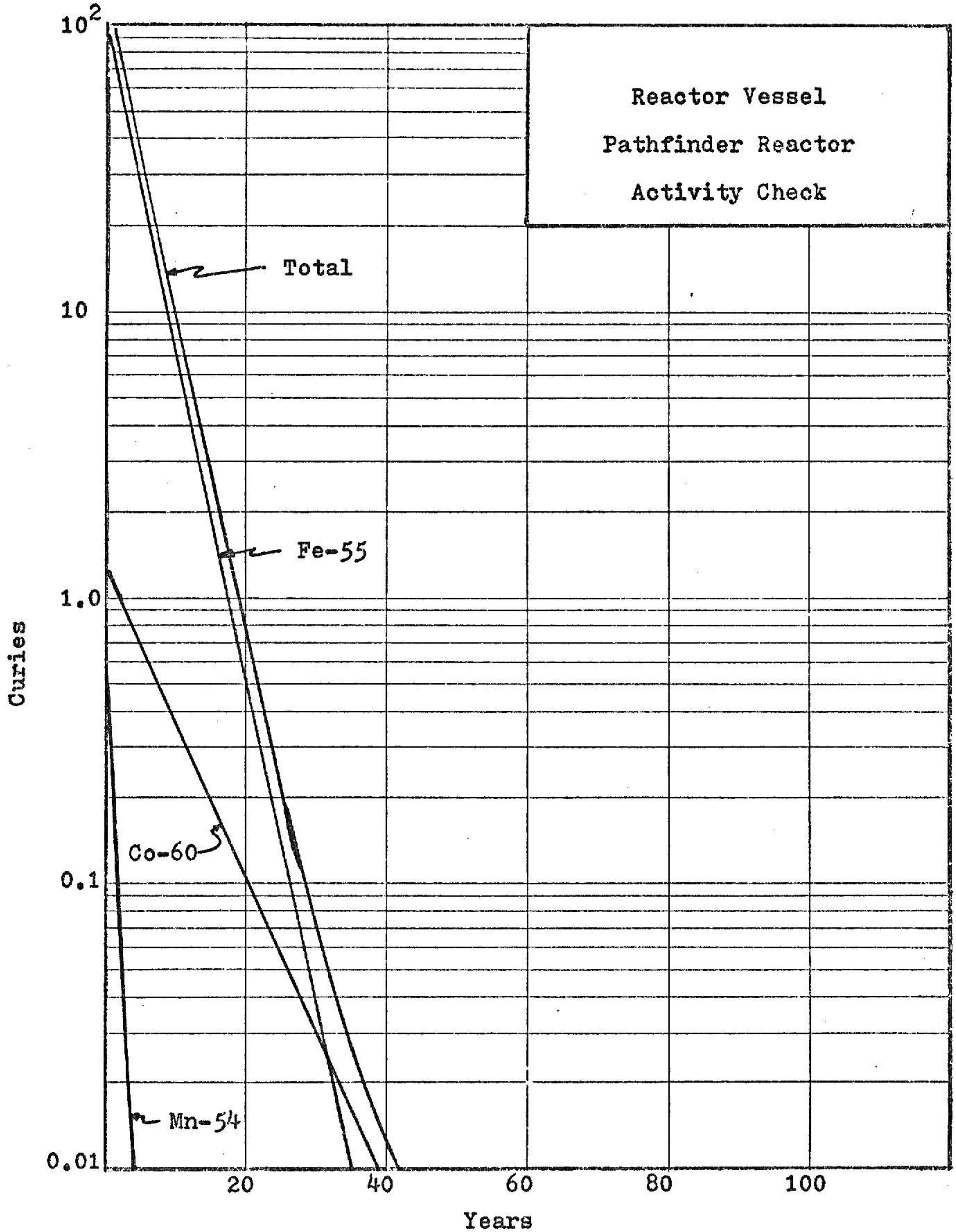

FIGURE 11 - Activity and Decay Calculated for Reactor Vessel - Pathfinder Reactor

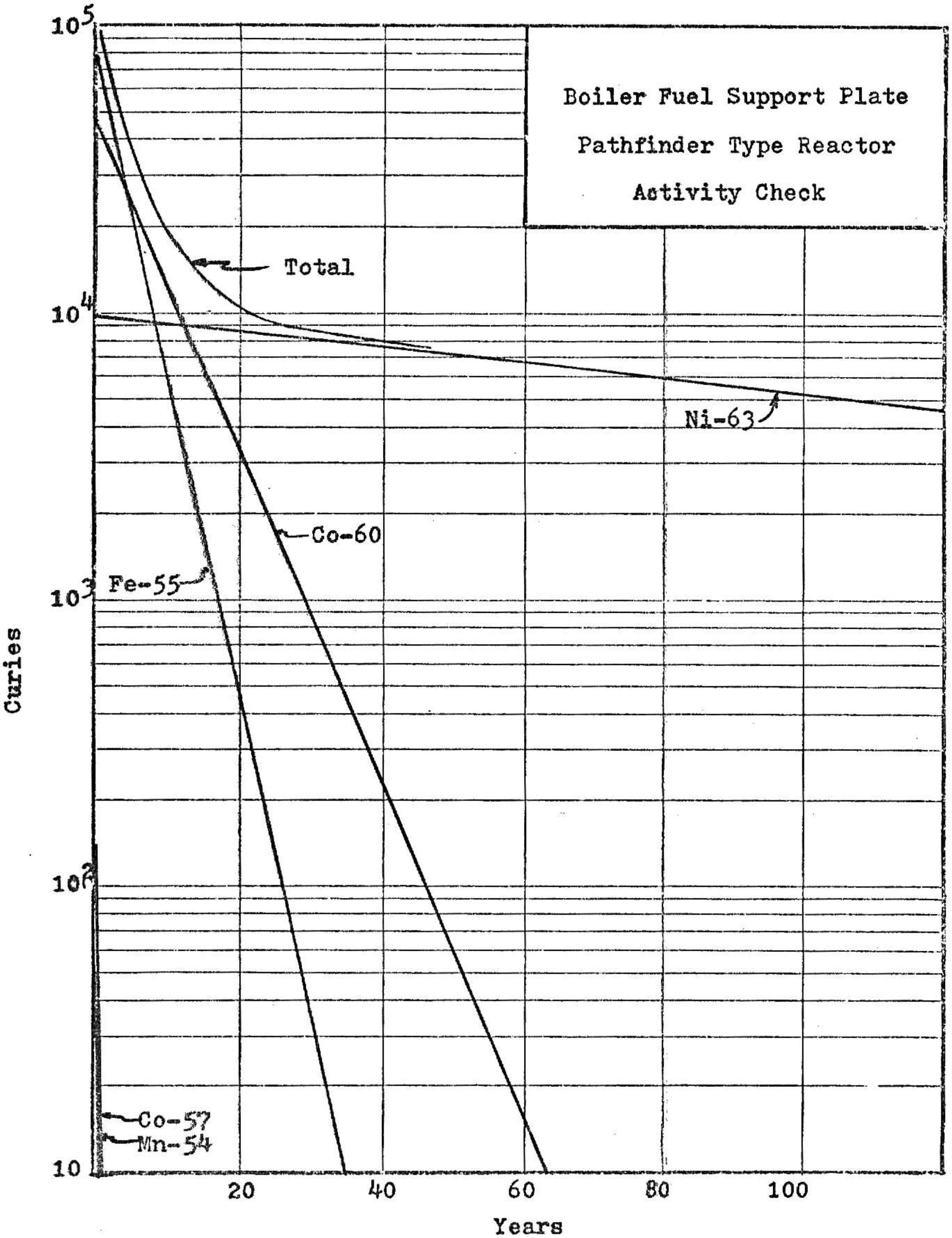

FIGURE 12 - Activity and Decay of Boiler Full Support Plant - Based on 40 Year Operating History (Figure 6)

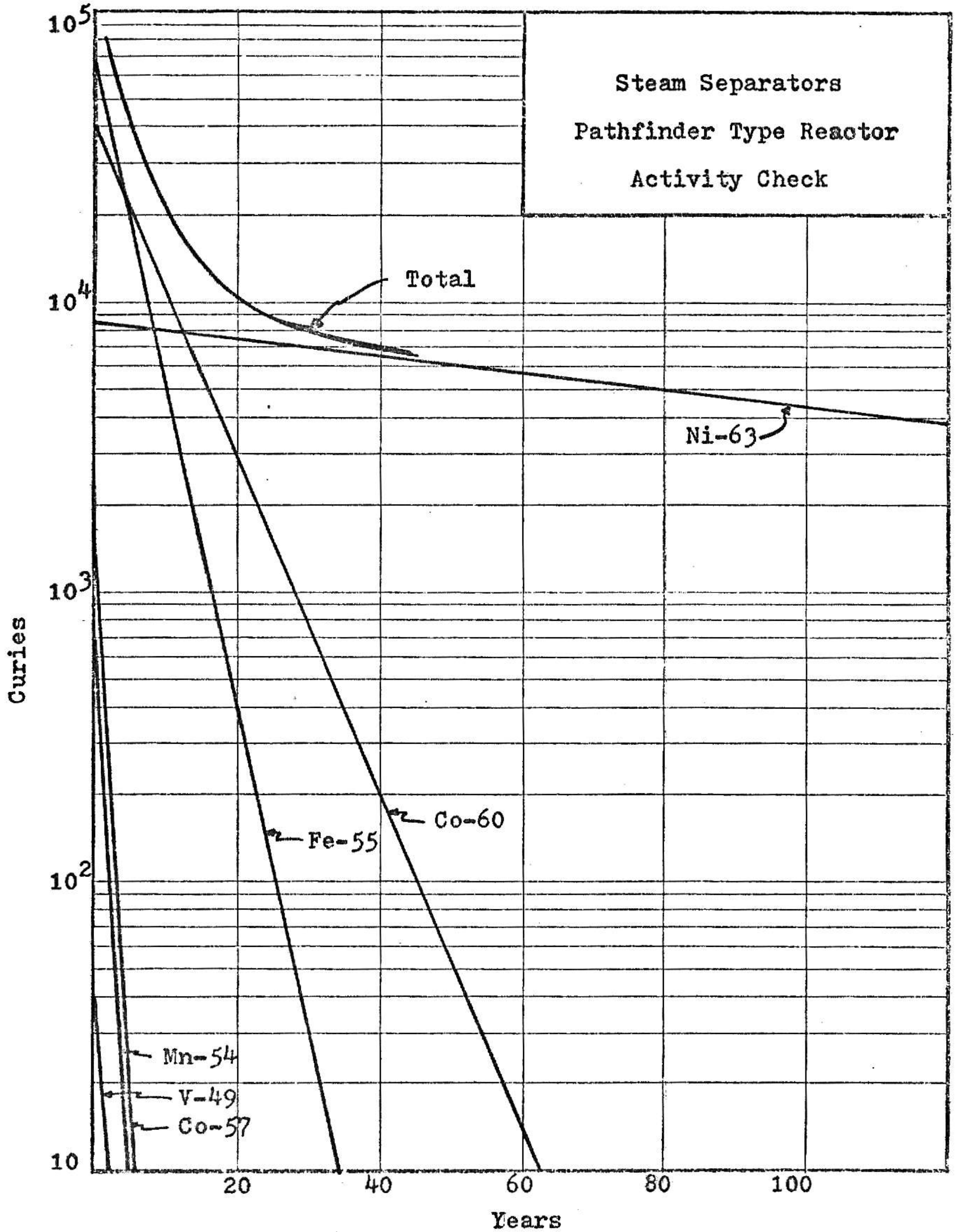

FIGURE 13 - Activity and Decay of Steam Separators - Based on 40 Year Operating History (Figure 6)

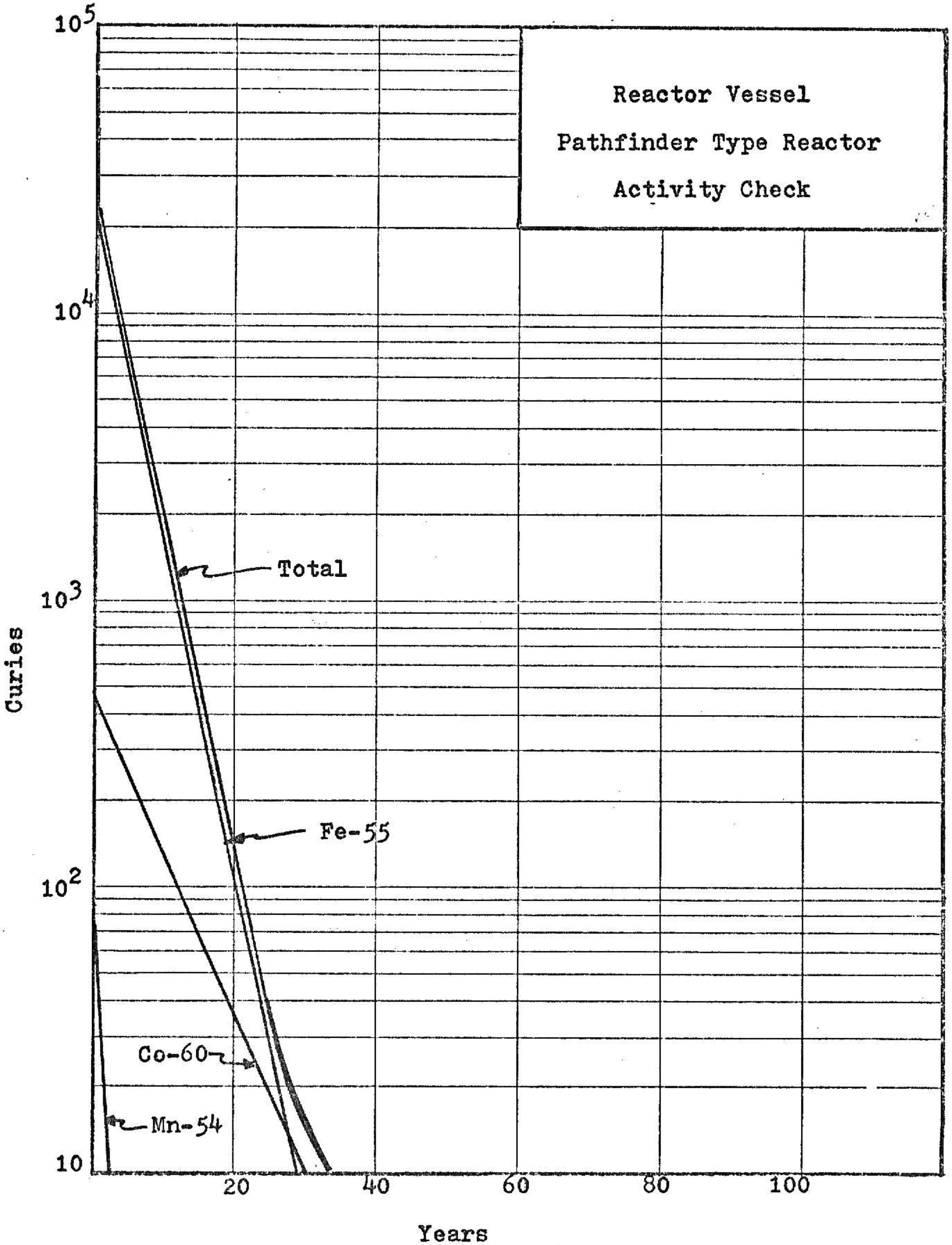

FIGURE 14 - Activity and Decay of Reactor Vessel - Based on 40 Year Operating History (Figure 6)

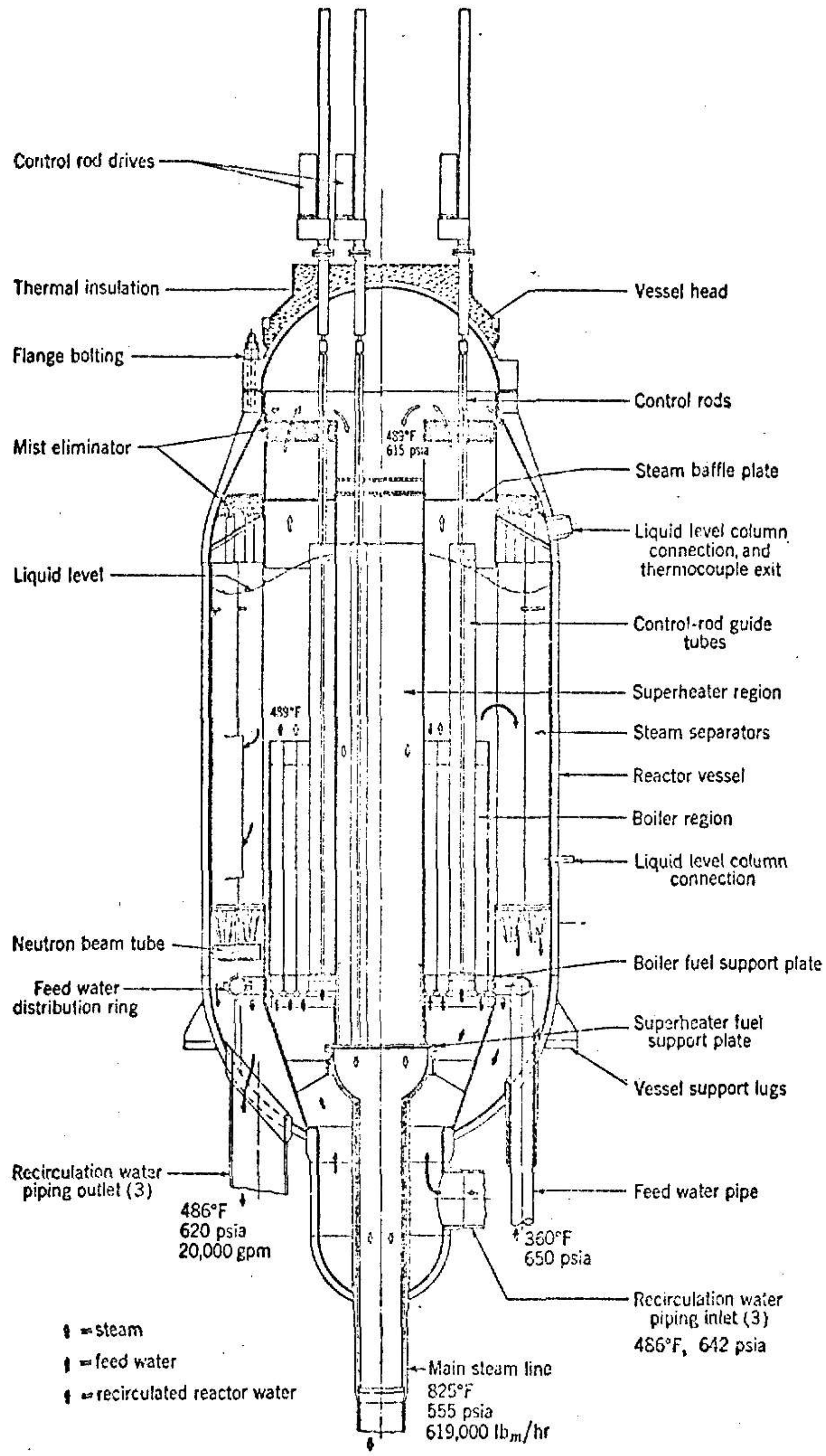

FIGURE 15 - Pathfinder Reactor Vessel and Internals

Ni-63 is high, the specific activity is low (ratio of Ni-63 to non-radioactive nickel). Therefore, if the entire body burden of nickel were replaced with the nickel in the reactor component (the standard man contains 0.01 grams of nickel), the component could be considered "safe" when .01 grams of the component nickel had an activity less than the maximum permissible body burden (MPBB) for nickel (26).

3.3 Utilization of Plant Facilities for Decommissioning

To minimize the cost associated with decommissioning a nuclear facility, use should be made of the existing plant equipment and facilities. The rad waste system of the plant including solidification and packaging equipment should be used as long as possible. The maintenance decontamination procedures and facilities should be utilized. If DF's of 10 to 16 can be reached by decontaminating the plant systems, the radiation levels should be reduced by an order of magnitude. This reduction will simplify working with the contaminated materials.

During the dismantling, extreme cleanliness will be required to insure that contamination spills can be cleaned with a minimum of cost. Recycling of the cleaning water through the demineralizers and plant clean up systems will minimize the quantity of low level liquid waste generated. Utilization of rad waste systems and demineralizers will contribute to lower costs. At some point in the dismantling work it may be necessary to dismantle the plant systems themselves and then temporary systems for waste handling will be required. Use may also be made of the plant's radiation monitoring and health physics facilities during the dismantling work. During the actual dismantling, sources will have to be kept at the site for instrument calibration.

3.4 Physical and Radiological Consideration-BWR

The only vendor presently marketing a BWR NSSS in the United States is General Electric Company. The GE BWR is a direct cycle unit and as such the turbine generator and associated steam and condensate systems become contaminated due to radioactive contaminates in the steam generated in the reactor.

A hypothetical BWR is shown on Figures 16 and 17 (51). The main plant buildings consist of the reactor building (which includes the containment and much of fuel storage facilities), the turbine building, the feedwater heater and demineralizer bay, the control building, and the rad waste building. In lieu of the primary loop found on the PWR, the BWR has a recirculation system which consists of a recirculation pump which furnishes water to multiple jet pumps located in the periphery of the reactor vessel. These jet pumps furnish the forced circulation of coolant water in the reactor. The auxiliary systems found in a BWR are similar to the systems of a PWR, though there are distinct differences. The BWR demineralizers are full flow units located in the feedwater lines and a CVCS system is not needed.

Since the BWR is direct cycle, the main steam, reheat, extraction steam, condensate, heater drains, and feedwater systems will be contaminated from radioactive particles suspended in the steam system. Radiation levels in the systems may range from 5 to 250 mR/hr even after shutdown. Hot spot readings as high as 2R/hr can be found. Actually, the contamination levels of the turbine and associated systems have been found to be very low due to the excellent steam-water separation factors that are encountered in the BWR reactor at the steam interface. Concentrations of radionuclides may be 10^4 to 10^6 times higher in the water than in the steam (23). The radiation levels of the steam and feedwater systems will also be affected by decontamination efforts and by the material constituents of the systems. In the initial BWR's constructed, extensive use of cupro-nickel, Monel and Admiralty brass in the feedwater systems resulted in considerable corrosion products in the reactor due to those materials. In particular, the Monel and cupro-nickel resulted in formation of Co-58, while the Admiralty brass resulted in Zn-65. Later units and units now under construction use stainless steel tubes in the feedwater systems and this reduces the concentration of Co-58 and Zn-65 being generated in the reactor.

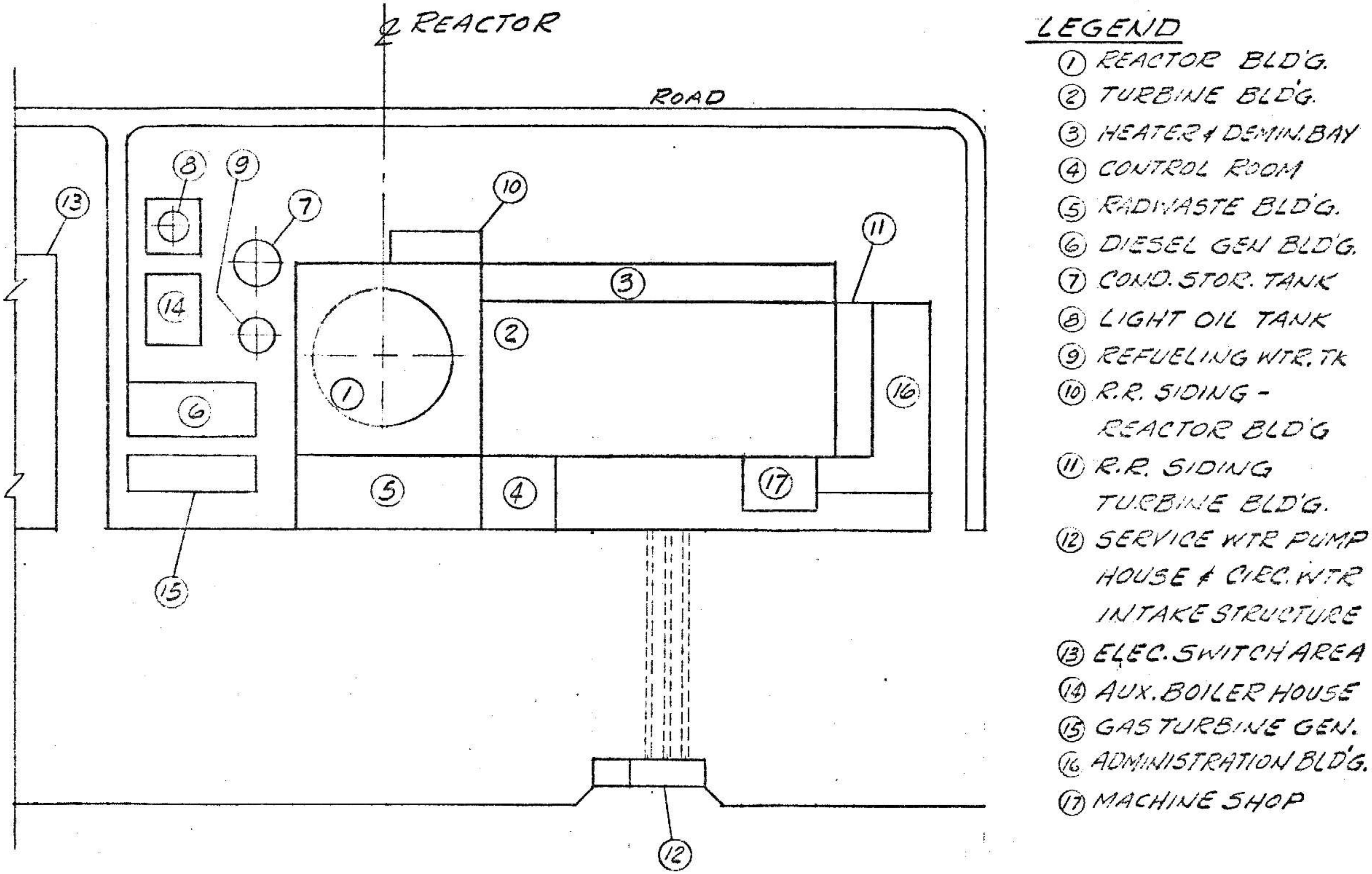

FIGURE 16 - Site Layout of Hypothetical BWR

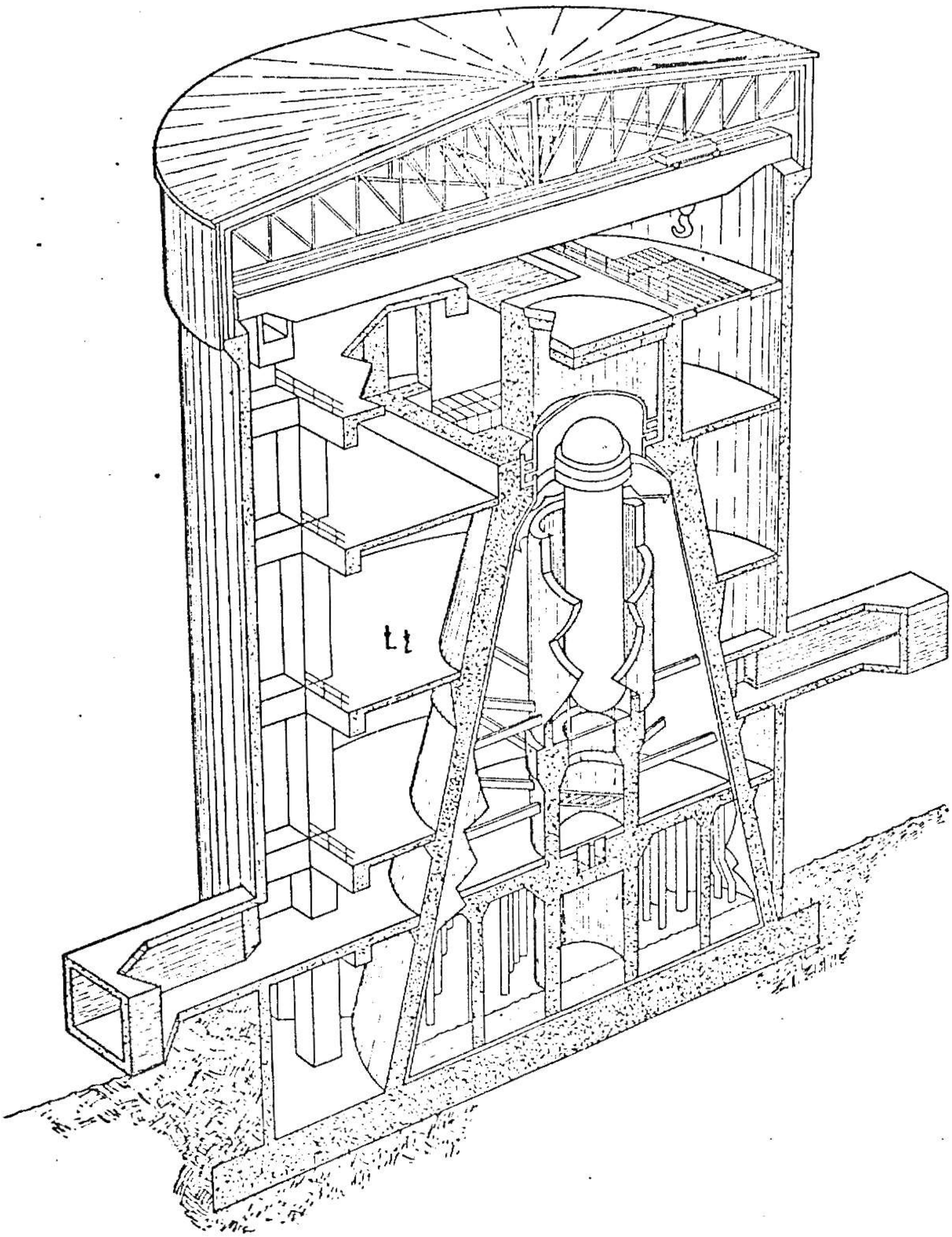

FIGURE 17 - BWR Reactor Building - (Over-Under)

4.0 DISMANTLING OPTIONS

The options available for dismantling a nuclear power plant can be broken into three distinct types. The AEC, in Regulatory Guide 1.86 (49), has listed the options as 1) Moth-balling, 2) In-Place Entombment, or 3) Removal of All Radioactive Components and Dismantling. The AEC has identified a fourth option which is Conversion to a New Nuclear System or a Fossil Fuel System. The probability of utilizing a new steam supply, such as was done with the Pathfinder Power Plant does not appear likely for large power stations, as will be discussed in Section 7.5. The other three options: Mothballing or "Storage," In-Place Entombment or "Internment," or complete "Removal" span the options available from past experience, present review of the situation, and from the position of the Regulatory branch of the AEC.

4.1 Storage

Utilization of the site structures for storage of radioactive components in their operating configuration is the "minimum" dismantling effort which is acceptable under present AEC regulations. The containment structures were designed to provide adequate safety for the public health during reactor operation and they should therefore be adequate for storage of radioactive materials once the plant has been shut-down. However, even with this minimum option there are certain work items which must be completed prior to modification of the facility license.

In order to proceed with minimum or "storage" dismantling, the licensee must remove all special nuclear material from the site, and the reactor must be deactivated and made incapable of being operated. All contaminated piping systems must be decontaminated, drained, and then sealed to prevent future spreading of contamination. The fuel storage pool, waste storage tanks, and demineralizers must be handled in the same manner. Open tanks, such as the fuel storage pool, may require covering with a concrete slab or they may be decontaminated by sealing the radionuclides to the tank surface by painting those surfaces. Painting the surface of a contaminated component will only be allowed after the licensee has demonstrated that adequate steps have been taken to decontaminate the surface and that the surface has resisted those efforts (49). Whether painting alone is adequate depends on whether a secondary boundary is available to control contamination spread. All radioactive resins and wastes must be shipped off-site for burial. All areas of high radiation, such as around the reactor vessel, must be sealed to prevent access and water intrusion. Monitoring facilities must be installed in the facility low spots to detect any water leakage into the storage areas. The storage areas must be sealed by welding shut all pipe penetrations, by sealing shut all windows, doors, electrical and other penetrations except for one access door and a storage area pressure equalization line. The equalization line must prevent storage area overpressures by letting the area breath through a filter.

In a PWR, the radioactive systems must be confined primarily to the reactor building, the primary auxiliary building, the fuel handling building, and the rad waste building though other buildings may contain stored activity depending on site layout. Those buildings which are not used as storage areas must be decontaminated to below levels required for unrestricted access

(27) while the storage areas must be sealed as described above and adequately locked to prevent unauthorized access. Prior to termination of the operating license, a complete and extensive survey of site radiation and contamination levels must be made so that future surveys will reveal any spread of radioactive materials. The licensee must instigate periodic radiation surveys and must keep the site under surveillance to prevent unauthorized personnel from gaining access to the storage areas.

During the initial dismantlings of CVTR, BONUS, Piqua, and Pathfinder (5,2,6,11) the AEC terminated the Part 50 or Reactor License and issued a Part 30 or Byproduct Material License for the radioactivity stored on site. Recent plans to decommission the Saxton Reactor have shown a shift in AEC policy and the Saxton Nuclear Experimental Corporation will maintain the Saxton site after dismantling under a Part 50 "possession only" license. Listing the high inventory of radioactive material on the site as the determining factor, the AEC has decided that all dismantled reactor sites which are used for storage or disposal of radioactive material will be required to maintain Part 50 licenses unless the licensee makes application to the State (if an Agreement State) for a byproduct materials license.

Advantages of the "storage" concept are minimization of cost and minimization of the worker's occupational radiation exposure. At a latter date the site can undergo a more permanent solution such as burial or complete removal of the radioactive materials. By undertaking the storage option a utility is "leaving its options open." Costs of the "storage" dismantling for a hypothetical 1000 MW(e) PWR is shown on Table VI. The schedule for the decommissiong work is given in Figure 18. The much shorter dismantling time results in savings for the licensee. In addition, the actual work, such as decontamination of the plant systems and buildings, can be done by plant maintenance personnel using standard maintenance procedures.

4.2 Internment

The "internment" dismantling option entails using the plant site as a burial ground for radioactive material. The actual burial will use those portions of the reactor building, primary auxiliary building, fuel handling building, and rad waste building which are below grade to serve as tombs or storage vaults. The radioactive components will be stored in the lower recesses of these burial buildings, and then the upper portions of these structures will be leveled, the tombs covered with earth, and the radioactive material permanently buried. When estimating the costs of the "internment" dismantling, it was assumed that the turbine generator building, the control building, and other auxiliary buildings which are not contaminated or do not contain significant amounts of radioactive materials will be decontaminated and modified to serve as warehouses or office space. This modification entails removal of all equipment, piping, and power wiring and altering the building as necessary to utilize as much of the facility as is practical. The turbine generator and condensers may be disposed of as scrap depending on the economics of complete removal at the time of dismantling.

Specific work items required by the dismantling include removal and sealing of all contaminated pipe and equipment. The radioactive components will be stored in the burial vaults or shipped for burial off-site, while the non-contaminated material will be disposed of as scrap. The upper portion of the reactor containment and the other storage or burial structures will be razed to below grade level. The storage areas of the burial vaults shall be filled with concrete to prevent access and contamination spread. The burial vaults must be covered with a water impervious membrane to prevent water intrusion and the membrane must be covered with backfill and the area graded and planted in grass. In the Hallam dismantling, the water impermeable membrane was Amercoat Nob-Lock and Amercoat 40 mil Plain Sheet and all joints were sealed using Amercoat No. 22 Adhesive (28). The membranes must be resistant to water (fresh, salt, or brackish), fungus, bacteria, dilute organic acids, dilute solutions of alkaline chemicals, and highly corrosive salts. Site ownership must remain with the licensee, and he must conduct periodic environment radiation surveys to assure that the site does not prove inimical to the public health.

In assessing the safety of the "internment" dismantling option, several items must be considered. The sanctity of the vault must be maintained for from 120 to 200 years. At the end of this time, the radiation levels will have decayed to levels below those requiring restricted access. Assuring that the site will remain impervious to corrosion and leaching is dependent on groundwater flow, water table level, soil conditions, concrete type used and cathodic protection provided for the burial structures. The groundwater may be passive or corrosive depending on its constituents. Groundwaters with high sulfate contents are particularly detrimental to most concretes. The mechanism of corrosion and leaching of concrete are not completely understood though the various concrete manufacturers, the National Bureau of Standards and the American Concrete Institute are doing extensive research to investigate the factors which affect concrete deterioration (29,30). Concrete type V which is low in tricalcium aluminate has demonstrated the best resistance to sulfate groundwater attack. The safety analysis of the entombed activity must demonstrate that the radioactive material remains safe even though the walls of the vault are breached. In addition, the environmental surveys conducted by the licensee must include analysis of water from wells located in the vicinity of the site to assure that no radioactivity is entering the water supply. A passive cathodic protection system utilizing sacrificial anodes may be installed to prevent rapid deterioration of the steel containment shell but it will not be practical to maintain an impressed voltage cathodic protection system on the containment for the 125 to 200 years necessary to insure that the burial vault remains unbreached. In the safety analysis, typically no credit is taken for the cathodic protection.

The actual dismantling work of the "internment" option will require careful scheduling to assure minimization of costs and effective use of all workers. By careful scheduling of work it will be possible to utilize plant facilities as long as possible (the rad waste system, etc.), and the storage of radioactive components in the burial vaults will proceed under careful guidelines to maximize the on-site storage and minimize the radioactive material which must be shipped off-site for burial, thereby saving shipping and burial costs.

The comparative schedule for the dismantling is shown on Figure 18 while costs are summarized on Table VI. "Internment," as demonstrated by the Hallam Dismantling (31), is a "permanent solution" to dispose of radioactive materials at a power plant site. Though the burial is by no means irretrievable, the possible reclamation of the buried material would only be attempted under the most extenuating circumstances.

4.3 Removal

The removal dismantling option entails removing all vestiges of the power plant from the site and returning the site to its original condition. Complete removal can be considered an "extreme" option,but the dismantling of the Elk River Reactor has shown that it can be done. At the ERR, all vestiges of the plant except the lower foundations of the major buildings, will be removed from the site. The foundations will be checked to insure that all radioactive concrete has been chipped out and shipped off-site for burial. Once it is assured that all nonnatural radioactivity has been shipped off-site, the foundations will be filled with rubble, covered with top soil, and planted with vegetation.

The actual dismantling will require extensive work in radiation fields. Since all radioactive components must be removed, it will be necessary to remotely dismantle the "hot" items such as the reactor vessel and internals.

The dismantling of the ERR reactor (32,33) demonstrated the capability of remotely cutting up a reactor vessel and internals. The reactor vessel and the inner and outer thermal shields were segmented remotely using a plasma arc cutting torch. The inner shield was cut up under water while the external shield and the pressure vessel were cut up in air. The remote tools required to operate the plasma torch and other dismantling tools were designed and fabricated by Oak Ridge National Laboratory. The inner shield took five weeks to segment and transfer; the pressure vessel, 15 weeks; and the outer shield, 10 weeks. Radiation levels were 1300 R/hr on the inner thermal shield surfaces.

The actual dismantling of ERR progressed from the pressure vessel out. By working from the highly contaminated reactor internals out to the containment, the dismantling forces made maximum use of existing plant systems and structures. The ERR pressure vessel contained an estimated 1200 curies of activity. However, at the end of a 40 year life, a conventional power plant reactor vessel will be considerably more active. In addition, the physical size of a PWR or BWR reactor vessel will require that it be cut into many pieces to allow shipment of the highly radioactive segments in appropriate shipping casks. Unlike irradiated fuel, which contains large amounts of fission products which are capable of contaminating large areas, the radioactive scrap from a "removal" dismantling will contain little potential for contamination spread in the event of a transportation accident. Prior to commencement of the actual dismantling, all SNM will be shipped off-site, all systems and contaminated areas will be extensively decontaminated to remove loose contamination. One factor which will increase the efficiency of pre-dismantling decontamination is the use of strong decontamination solutions. Since the plant will not be operated again, extremely strong decontamination solutions can be used. Steps will have to be taken to assure that the decontamination solutions do not rupture a line and inadvertantly spread contamination. After the plant is decontaminated as far as practical, all resins, radioactive wastes, and other highly contaminated materials will be shipped for burial. At the initiation of dismantling, the activity left on site will be either induced activity in the reactor materials, or contamination (probably corrosion products) which could not be removed by the decontamination solutions. These two types of radioactive material, though hazardous to personnel working with the equipment in which the material is found, are not potentially hazardous to the public health. In transporting the radioactive material to burial, the only dangers will result from the high radiation fields from the reactor vessel and components and the possibility of contamination due to the "crud" located in the piping system. Therefore, it will be necessary to ship the reactor component segments in shielded casks designed to minimize external gamma radiation, and the piping and equipment internal surfaces which are contaminated will have to be sealed. In most cases, taping the pipe openings shut will be sufficient to prevent the spread of contamination from the pipes.

Two large tasks required by the dismantling will be the removal of the reactor biological shield and the dismantling of the containment building. If the containment inner surface can not be decontaminated, it can be painted to fix the radioactivity to the building. As this procedure has been used extensively in handling contamination from accidents (23), it should be capable of preventing contamination spread while containment sections are shipped for burial. The dismantling of the biological shield will result in the production of large quantities of highly contaminated airborn dust. It will be necessary to protect the dismantling forces and prevent unnecessary spread of the dust.

The actual magnitude of the transportation and burial activities will be significant factors in the cost of the dismantling. Location of the nearest AEC licensed burial ground and the condition of roads leading from the site to the burial ground as well as railroad connections, will have to be taken into consideration.

The complete removal of radioactive material from a site results in no future surveillance or licensing requirements, but offset against this "ideal" end result is the cost and the exposure of the working force to radiation. The transporting of the radioactive material to a central AEC burial ground allows large quantities of contaminated material to be kept under surveillance by a small work force. The radioactive materials will still have to be kept under surveillance until the radioactivity decays to insignificant amounts. The schedule for the removal dismantling option is shown on Figure 18 and the costs are included in Table VI.

4.4 Overall Scheduling Consideration for Dismantling Work

The scheduling of the dismantling work should be optimized to minimize the cost of the entire decommissioning effort. Many factors, such as local public sentiment, availability of trained personnel, interest rates, and current AEC regulations must be considered by a utility in selecting the dismantling option to use in decommissioning a nuclear facility. Any of the options presented or a modification of them may be selected by a licensee upon consideration of the alternatives open to him at termination of his facility operating life or license. In every case there are certain scheduling considerations which will affect dismantling costs for the utility. The overall considerations such as fuel costs, maintenance costs, licensing requirements, and manpower shortage costs have already been mentioned. Since the major problems confronting the dismantling workers in each option are the presence of radioactivity and contamination and the necessity of working in a radiation field, possible methods of reducing these problems should be evaluated. The AEC has accepted as regulations the National

Commission on Radiological Protection recommendations for the maximum permissible occupational exposure to ionizing radiation. Table VII lists the present NCRP radiation dose limits (50).

Recently, along with the AEC's limitation of nuclear reactor radioactivity releases to as "low as practical," the AEC and others have begun to examine the possibility of reducing the maximum permissible dose limit for occupational exposure from its present limit of 5 rems in one year to some smaller value. Regulatory Guide 8.8 is directed toward minimizing occupational exposure as "far as practicable." Dr. Karl Morgan of Georgia Institute of Technology has stated that the NCRP is reviewing the basis of the present occupational dose with the intent of lowering the limits if possible (34).

Reducing the occupational exposure limits will increase the cost of all dismantling work activities. Extensive use of additional shielding and remote work tools will be required for work tasks done in radiation fields. The length of time for remaining in any work area will be lowered due to the smaller allowable dose.

A second factor affecting radiation levels and radioactivity on site, is the decay of the radioactive nuclides. Immediately after the reactor is shutdown, the activity due to short half-life isotopes will contribute significant radiation levels. By letting the plant "cool" from two to four years, most of the short-lived radionuclides will have decayed to insignificant levels. Figures 9 through 14 show typical decay curves and their constituents for the reactor vessel and internals of the Pathfinder and a hypothetical similiar unit that has operated for 40 years.

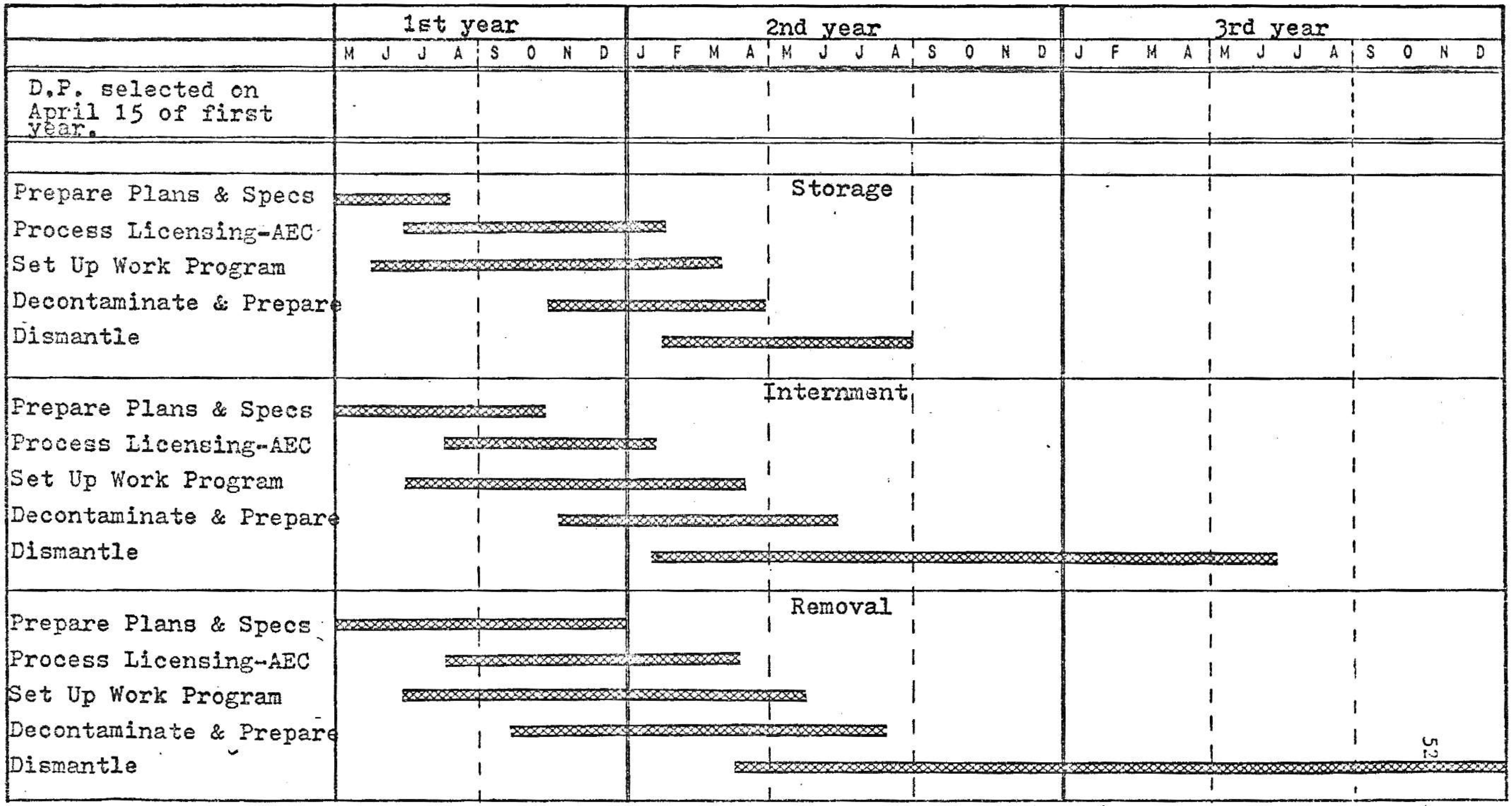

FIGURE 18 - Relative Schedules for Dismantling Options

TABLE VI DISMANTLING OPTION COSTS

COST ITEMS (evaluated at present dollar worth)	STORAGE $	INTERNMENT $	REMOVAL $
Labor	402,000	4,021,000	8,490,000
Materials and Equipment	161,000	1,407,000	3,306,000
Overhead	140,000	1,357,000	2,972,000
Subtotal	703,000	6,785,000	14,858,000
Engineering, Field Management and Contingency	450,000	1,900,000	4,458,000
Total	1,153,000	8,685,000	19,316,000
Present Worth of Surveillance and Maintenance	318,000*	212,000**	---
Evaluated Total Costs***	1,471,000	8,897,000	19,316,000

Cost estimates are "rough order of magnitude"

*Annual Cost of $15,000 escalated - 7% per year and present worthed at 7% per year for 20 years.
**Annual Cost of $10,000 escalated - 7% per year and present worthed at 7% per year for 20 years.
***No consideration has been given escalation, inflation or other economic factors.

See Appendix F for breakdown of labor cost item.

TABLE VII
NCRP RADIATION DOSE LIMITS

Category	Value
Maximum permissible dose equivalent for occupational exposure	
Combined whole-body occupational exposure	5 rems in any one year
Retrospective annual limit	10 to 15 rems in any one year
Long-term accumulation	(Age - 18) x 5 rems
Skin	15 rems in any one year
Hands	75 rems in any one year (25/quarter)
Forearms	30 rems in any one year (10/quarter)
Other organs, tissues, and organ systems	15 rems in any one year (5/quarter)
Fertile women (with respect to fetus)	0.5 rem in gestation period
Dose limits for the public, or occasionally exposed individuals	
Individual or occasional	0.5 rem in any one year
Students	0.1 rem in any one year
Population-group dose limits* (total exposure from "man-made" radiation above and in addition to natural background which averages about 0.1 rem per year)	
Genetic	0.17 rem average per year
Somatic	0.17 rem average per year
Emergency dose limits (life saving)	
Individual (older that 45 years if possible)	100 rems
Hands and forearms	200 rems, additional (300 rems, total)
Emergency dose limits (less urgent)	
Individual	25 rems
Hands and forearms	100 rems, total
Family of radioactive patients	
Individual (under age 45)	0.5 rem in any one year
Individual (over age 45)	5 rems in any one year

*Average for population group is intended to assure that no individual exposure exceeds 0.5 rem/year.

The cost of continued license possession must be evaluated against the savings received from delaying the dismantling work to wait until the short-lived radionuclide have decayed to insignificant levels.

The actual scheduling of the decommissioning work reveals that there are certain items which constrain a CPM schedule of the decommissioning work. During the initial stages of a decommissioning as the decommissioning order is being sought, the constraining items are preparation of the Decommissioning Plan and Safety Analysis Report, AEC review of these documents, and preparation of the plans and specifications for the actual work. Once the order has been received, removal of all fuel from the site and decontamination of the plant systems constrain the start of the dismantling work. Once the dismantling work has started, it will generally proceed through several concurrent "paths," any of which could become critical in relation to dismantling completion. The only obvious critical item is to assure that removal of shielding does not proceed removal of the shielded components.

5.0 DECOMMISSIONING ENGINEERING REQUIREMENTS

The following engineering requirements are extracted from the information available on previous decommissionings. In addition, the AEC has issued Regulatory Guide 1.86 "Termination of Operating Licenses for Nuclear Reactors, June 1974" to clarify the regulatory requirements for terminating an operating license. Though specific documents are identified, these documents may be combined by a licensee into one document as long as the relevant information is included. For example, the Dismantling Plan covered in Section 5.2 may include the Safety Analysis Report mentioned in Section 5.3. Regulatory Guide 1.86 does not specifically require an Environmental Report but issuance of the same is considered necessary by the author.

5.1 Initial Engineering Considerations

Recently there has been a general increase in concern for the environment. The AEC's mandate for control of the licensing of nuclear facilities was redirected by the Federal Court in the Calvert Cliffs decision to include complete environmental review of all phases of a nuclear plant's life and to examine the environmental impact of each phase in the licensing process. The AEC has responded in part to this by the issuance of Regulatory Guide 4.2 "Preparation of Environmental Reports for Nuclear Power Plants, March 2, 1973 (35)." Section 5.9 of this guide directs the applicant to "describe its plans and policies regarding the actions to be taken at the end of the plant's useful life. Information should be provided on the longterm uses of the land, the amount of land irretrievably committed, the expected environmental consequences of decommissioning, and an estimate of the monetary costs involved."

The guide goes on to request information on the considerations given decommissioning in the plant design and the expected dismantling steps the licensee will take, Though not requiring that the decommissioning plans be finalized, the guide is slanted toward assuring that the licensee applicant consider the costs and problems associated with the decommissioning of the plant early in the project planning. The Environmental Report is submitted in most cases with or shortly after the PSAR, and therefore, the decommissioning requirements should be given consideration throughout the plant design.

At the end of the plant's useful life, the licensee will review the options available to him and decide on a program for decommissioning the reactor. The review of options may be made by the licensee's internal engineering staff, or it may be handled by an architect-engineer (A-E). Selection of the program will follow a detailed review of the factors previously mentioned. After the program has been selected, the licensee will have his engineer prepare a Decommissioning Plan, a Safety Analysis Report (SAR), and an Environmental Report on the impact of the Decommissioning Plan. These will be submitted to the AEC with a Request for a Dismantling Order (36), and a Request for Modification of the Tech Specs. These documents formalize instigation of the decommissioning process.

5.2 The Decommissioning Plan

The Decommissioning Plan (10,12) formally outlines the program the licensee has selected for decommissioning the plant. The various dismantling activities are covered in enough detail to demonstrate the steps taken by the licensee to protect the public health and safety and to assure the AEC that adequate steps are taken to insure that the job proceeds in compliance with the appropriate regulations.

The Dismantling Plan (DP) opens with a review of the plant background. In addition to covering the plant design, the DP reviews the plant operating history. Any significant contamination accidents or significant spreads of radioactive material are summarized. It contains a summary of the current plant status, the dispostion of all SNM and radioactive sources, and it estimates the total plant radioactive inventory. The inventory contained in the reactor components is calculated using the procedure contained in Appendix A. The calculated inventory is compared to the measured inventory by measuring the radiation levels and back calculating the source strengths. A complete description of plant radiation levels is included in the DP as well as a description of the proposed plant final status.

The DP contains a detailed description of the dismantling activities including the proposed staff organization, and it delineates the responsibilities and lines of authority for the dismantling work. The health physics program, the safe work procedures, major work task descriptions, the work scheduling, and the work control procedures are outlined.

The DP is concluded with a review of the safety aspects of the actual dismantling operations. Control and protection of the dismantling workers, rad waste procedures, health physics procedures, and dismantling hazards are reviewed.

5.3 SAR, FSR and ER

The Safety Analysis Report (SAR) is submitted with the DP to the AEC, and it covers the safety considerations of the decommissioned plant. It reviews the structural design of the plant to confirm that it is adequate for the dismantling option selected. A review of radiological safety considerations is included to support the conclusions of the safety analysis.

A System Radiological Safety Analysis (SRSA) is generated by the Licensee's engineer. The SRSA reviews the various avenues of possible accidents which might occur. The consequences of floods, earthquakes, tornadoes, hurricanes, and other natural disasters and their interaction with the dismantled facility are reviewed to examine all possible avenues of contamination spread. Following the precedent established in operating license submissions, a worst conceivable hypothetical accident may be selected as a design basis accident, DBA. The safety analysis of the dismantled plant must confirm the adequacy of the dismantled plant to withstand the DBA (3).

The Final Status Report (FSR) is submitted to the AEC at the completion of the dismantling activities. It includes a description of the site's condition after completion of all dismantling work. A review of the radiological safety procudures which were followed during the dismantling as well as a complete review of the current plant radiological status is contained in the FSR. This includes a complete site radiation and contamination survey. The FSR concludes with a safety analysis of the plant as dismantled.

The Environmental Report (ER) addresses the environmental consequences of the dismantling (13). In addition to a plant history review and a dismantling operations summary, the ER contains a complete analysis of the environment impact of the proposed dismantling activities. In addition to considering the impact of the plant final status, the ER considers the effects of the actual dismantling work including shipping accident considerations. The ER contains an environmental surveillance program for the dismantled facility.

5.4 Revised Technical Specifications

Section 50.36 of 10 CFR 50 requires that the plant operation be defined by Technical Specifications (Tech Specs). The Technical Specifications include safety limits and safety system settings as well as limiting conditions for operation. They also include surveillance requirements, design features, and administrative controls as well as operating procedures for the discharge of effluents (50.36a) from the plant. These Tech Specs define the limits under which the reactor may be operated by the licensee, and they define the checks, analyses, surveys, and procedures which the plant operations staff must incorporate in their operating procedures. Failure to comply with the Tech Specs can result in the licensee being cited by the AEC, which can result in fines and other punitive actions. Compliance with the Tech Specs is therefore "mandatory." However, the Tech Specs are oriented toward an operating power plant, and they include many requirements which are irrelevant to a plant being decommissioned. This is particularly true once the actual dismantling work is started. Therefore, a request for complete revision of the Tech Specs is normally submitted with the DP.

The modification of the Tech Specs can be conditional. For example, when all fuel is shipped from the site, the Tech Spec requirements covering SNM safeguard procedures can be automatically deleted if the AEC agrees. There are normal procedures for the modification of the Tech Specs but these procedures are generally time consuming, and if possible, the licensee should submit streamlined modifying procedures along with the revised Tech Specs. This will allow the dismantling to proceed with as little delay as possible due to noncritical license requirements.

5.5 Plans and Specifications

Due to the nature of the dismantling work, it is extremely important to plan the work effectively. Inadequate plans will result in delays caused by insufficient training, lack of coordination, and inadequate equipment. Control of the work force and job tasks is necessary to minimize the possibility of contamination spread. The general work program must consider utilization of existing plant systems, equipment, and facilities as long as it is practical. The program must also consider minimization of worker radiation exposure. Material control and waste disposal must also be included in the work program as well as HP monitoring, record keeping, and decontamination procedures.

The licensee may not be staffed to handle the dismantling work, or the licensee may not want to subject his normal work staff to radiation doses which will limit their normal work capacity. The utility can contract the work to be done by an independent contractor. However, the utility will still have primary responsibility as licensee to see that the work is accomplished in accordance with AEC regulations.

In order to procure a meaningful bid from a vendor or contractor for the dismantling work, the exact work scope, the work management and HP responsibility interface, as well as all AEC requirements will have to be called out in the specifications. In addition, such items as decontamination and contamination removal costs, which vary from plant to plant and therefore are not easily estimated, will have to be priced on a cost plus basis to handle the inherent uncertainties in the tasks.

The licensee will probably employ an architect-engineer to generate the SAR, DP, and specifications. The specifications will need to cover the work scope and the safe work procedures to be followed to insure that the contractor does not violate the licensee's regulatory requirements. This inclusion of procedures in the specification is treading on grounds normally controlled by the contractor management, but the contraints of the AEC license impose requirements on the licensee to assure that the work will be done according to the regulations. Though the contractor will be responsible for his work force, it will be necessary for the licensee to assure that they follow safe work procedures.

5.6 Dismantling Supervision

As licensee, the owner of the facility has prime responsibility to the AEC to assure that the dismantling proceeds in accordance with AEC directives and regulations. Though the licensee may delegate his authority for job control to persons outside his organizations, the responsibility is still his. The actual dismantling work should be under the direction of a qualified project manager. Typical supervisory personnel are listed below.

The project manager must be familiar with construction work in radiation fields, health physics, monitoring, waste disposal procedures, and AEC regulations, in addition to being qualified to handle construction or demolition work. His primary function will be to plan and coordinate all phases of the dismantling so that the work proceeds as smoothly as possible.

The health physics chief will be responsible for monitoring radiation levels and establishing radiation areas. He shall assist the project manager and contract supervisor in preparing safe work procedures, and he shall provide data for work permits. He shall also be responsible for maintaining radiation exposure records and for developing effective decontamination procedures to deal with abnormal contamination problems.

The contract supervisor will have primary responsibility for issuing work permits for the individual work tasks and for getting the work done. He will be directly responsible for labor relations and work productivity.

Assisting these supervisory personnel will be a safety committee comprised of the plant superintendent, the plant health physics supervisor, and the various plant engineers and other qualified individuals, This group, some of whom must be completely familiar with the plant, will review the work progress, Tech Spec modifications, and work procedures to monitor independently for possible safety problems. They will report any findings to the project manager.

These supervisory personnel will have to be assisted by a highly trained staff to assure that the work is done expeditiously. Shortage of trained manpower may be a definite problem area. It may be necessary to institute a rigorous training program for the dismantling staff personnel to provide the training required.

The breakdown of job supervision into three areas of independent responsibility (project manager-license, health physics chief-radiation safety, and contract supervisor-dismantling work) with independent review of the safety questions, should provide proper control of this job. In this framework, proper considerations for radiation safety will be possible.

5.7 AEC Regulatory Guides

In addition to the requirements of 10 CFR 50, in particular paragraph 50.82, pertaining to the decommissioning of nuclear plants, the AEC has issued several regulatory guides which affect the decommissioning of nuclear plants. Regulatory Guide 1.86, "Termination of Operating Licenses for

Nuclear Reactors" was issued in June 1974 to clarify the methods and procedures considered acceptable to the AEC Regulatory staff for meeting the requirements of 10 CFR 50.82.

In addition, Regulatory Guide (RG) 1.8 on "Personnel Selection and Training" and RG 1.39 on "Housekeeping Requirements for Water-Cooled Nuclear Power Plants (3-16-73)" affect the economic considerations relative to dismantling the facility. Many other guides, including RG 5.20 on "Training, Equipping and Qualifying of Guards and Watchmen," (which may be modified due to safeguards considerations) and RG 7.1 on the "Administrative Guide for Packaging and Transporting of Radioactive Materials" affect the cost of dismantling a facility.

The Regulatory Guides of Division 8 which deal with occupational health are applicable to the dismantling work. In particular, RG 8.2 "Administrative Practices in Radiation Monitoring," RG 8.7 "Occupational Radiation Exposure Records System," RG 8.8 "Information Relevant to Maintaining Occupational Radiation Exposures as Low as Practicable (Nuclear Reactors)," and RG 8.10 "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Practicable," serve as guidelines for proper procedures and control of the dismantling activities.

In general, the Regulatory Guides as issued and the other regulations of the AEC combine to provide an adequate picture of what is acceptable to the AEC in general terms.

6.0 DISMANTLING ACTIVITIES

The dismantling activities reviewed in these sections result in increasing cost for the dismantling work. The activities are designed to minimize the consequences of radiation exposure and decrease the possibility of contamination spread. The activities are based on experience with work in radiation fields.

6.1 Safe Work Procedures and Work Permits

As the actual dismantling work will require work in radiation areas, general Safe Work Procedures (SWP's) will be generated for all repetitive work tasks. Typical tasks covered will include pipe cutting, pipe capping or closure, equipment decontamination, system draining, system decontamination, and waste disposal. These SWP's will be general in nature and they will define the procedure that will be followed to accomplish the task. The SWP's will be generated by the contract supervisor with assistance from the project manager and the health physics chief. They will incorporate the procedures included in the job specifications if the work is done by an independent contractor.

Each SWP will list prerequisites required before the work can commence. Equipment required will be listed to assure that there is no loss of time retrieving tools and other equipment after the work commences. The SWP will specify minimum radiation monitoring and personnel protection equipment needed. The equipment needed may include respirators, anti-contamination clothing, polyethylene sheets, and tools as required by the work. This list will be supplemented by the radiation work procedure (RWP) posted at the work area. The work method will be described in detail with special emphasis given to radiation monitoring requirements and cleanup procedures. The SWP's will include the work control procedures used to monitor the work and will be approved by the safety committee.

The primary work control will be exercised by issuing Work Permits (WP). A Work Permit will be issued by the contract supervisor for each item of work. The WP will refer to a SWP if the work item is typical, or it may contain a specific procedure to follow. It will cover disposal of the contaminated material and steps to be taken to minimize radiation exposure. The WP will be issued by the contract supervisor after being approved by the project manager and health physics chief. After the task is done, the WP will be certified by the work group supervisor to show completion of the task, and it will be returned to the contract supervisor.

6.2 Health Physics

Before a dismantling work task starts, it is necessary to examine the hazards, monitor the radiation and contamination levels, catalog the area as to radiation level, select the dress required (anti-C, etc.), and specify the safety precautions to be followed when working in the area. The health physics staff will be responsible for these items. After a work area has been surveyed and the allowable exposure time calculated, the health physics monitor (HP staff member) will issue a radiation work procedure (RWP) which will be posted at the work area. The RWP will list requirements for special monitoring equipment, exposure time limits, protective clothing, and protective shielding. The RWP will be issued before health physics can approve a WP.

The HP staff will also assist in the generation of SWP's and will assist in the generation of special decontamination procedures. In addition to normal instrument maintenance, radiation record keeping, and monitoring radiation levels, the HP staff will be required to provide radiation monitoring for each work group whenever there is a possibility of contamination spread. When cutting a pipe that is internally contaminated, the HP monitor will continuously monitor the cut area to detect any large activity releases. This monitoring will require a large HP staff initially. As the work progresses, if it is found that the monitoring isn't necessary, the HP work force can be reduced.

One of the work practices that must be discouraged is "burning out" the labor force. "Burning out" consists of letting a worker accumulate his maximum allowable radiation dose in a short work time by requiring him to work in high radiation areas without taking steps to lower the exposure rate, and then laying off the man because of the limits of 10 CFR 20. This philosophy of "burning out" the local labor force has long range repercussions that may not be immediately evident. In addition to limiting the local labor force's capacity to work for the licensee, it also could result in a negative public attitude toward the work and the licensee. Typically, "burning out" is more common when independent contractors with no attachments in the area are responsible for the work. In order to maximize the contract profits, time spent on shielding and radiation protection are minimized by the contractor. The AEC, with issuance of Regulatory Guide 8.8, has already taken steps to discourage this practice.

6.3 Material Control

Once all SNM has been removed from the plant site, the material control program should shift from a dual role of material safeguards and radioactivity control to one of radioactivity control. The safeguards portion of the material control program is oriented toward preventing diversion of SNM for clandestine purposes while the radioactivity control portion of the program is oriented toward preventing accidental release of radioactive material into the environment.

The material control effort during the dismantling should be directed toward controlling contaminated scrap, tools, and equipment. The work force may be tempted to pilfer or unknowingly take contaminated material from the site, particularly since the material is being disposed of as scrap. If the work force is familiar with the radiation safety aspects of the job, they may feel that the rigid safety program (material control) is unnecessary. (An example of such an attitute is the contempt many skilled craftsmen feel about OSHA regulations.)

The disposal of uncontaminated scrap from a dismantled facility by local disposal methods (scrap dealers, junk yards, and the local land fill) can result in unfavorable publicity, particularly if contaminated material is discovered in the area. To assure that the scrap is not contaminated, it will all have to be surveyed and checked. That may offset any savings realized by disposing of the clean material locally. Cost of shipping and burial at a nuclear burial ground will have to be weighed against surveillance and checking costs.

The actual control program will probably utilize an extensive tagging procedure to control movement of tools and material on the job. Inspection and tagging of all material removed from a work area will be required to control the disposal of scrap, reduce the spread of contamination, and assure that dismantling tools and equipment are utilized as effectively as possible. Contaminated tools can be transported from one work area to another by enclosing the tools in plastic bags, thereby decreasing the need for complete decontamination of tools after every work task which would increase the job costs unnecessarily.

One of the material control problems will be the storage and disposal of radiation sources for calibrating detection equipment. These calibration sources will be needed by the health physics staff until the dismantling is complete and they should be transferred to another licensee or plant when the work is complete.

In order to minimize the spread of contamination, the control of each work area must be planned and maintained by the working staff. This control will include roping off the area, marking it appropriately, confining entrance and exit to a single location and keeping the area as clean as possible. Utilization of visquene tarps and sheets to confine the material from pipe cuts, equipment dismantlings, and concrete removal will assist in limiting in contamination spread. Utilization of special paints and wall coverings in the normal plant design will assist in contamination control. These special paints (37) are designed to prevent penetration of the radioisotopes into the surfaces on which the paints are applied. Information is still being accumulated on the best material for each application, but as the operating years accumulate on present plants, the paints will be refined and their radioactivity penetration resistance will be optimized.

6.4 Waste and Work Items Disposal

Whatever option is selected by the licensee for dismantling the plant, the actual dismantling will result in the generation of large quantities of contaminated and radioactive materials. These materials must be disposed of either by on-site emtombment or by shipment for burial off-site. In a minimum storage option, the dismantling work will utilize existing facilities to process radioactive wastes and contaminated material. With each increase in the scope of work required, the load on the plant facilities will increase until they are not capable of handling the volume of material generated. If the work scope is further increased, then temporary facilities will have to be errected to handle the excess waste material generated. On a complete removal job, these temporary facilities will probably include: Demineralizers, evaporators, solidifying equipment, rad waste packaging equipment, decontamination areas and facilities, personnel decontamination facilities, contaminated clothing processing facilities, storage areas for contaminated and noncontaminated wastes, and other facilities required by the working force. These facilities are required to handle the waste materials generated by the dismantling work. Actual disposal of the radioactive materials will require collection, packaging, shipping, and ultimate burial in an AEC licensed burial ground.

Once the plant is shutdown, no additional radioactive wastes will be generated, but the dismantling must be programmed to collect, isolate, condense, and prepare the existing waste for disposal. The collection of radioactive materials during the dismantling will utilize both the plant rad waste systems, the working area environmental systems, and the decontamination fluid handling systems. The scrap materials generated by the work parties for disposal must also be collected and prepared for disposal. The initial plant decontamination operations will function to remove radioactivity from the piping and deposit it on the cleanup filters and ion exchangers. The plant rad waste systems will serve their normal functions during the dismantling cleanup, as will the plant decontamination fluid handling systems. The working area environmental systems (large blowers venting through filters) will concentrate airborn contamination on the system filters.

Except for Special Form Radioactive Materials (SFRM), the radioactive materials collected present a hazard due to radiotoxicity and they must be handled by special procedures. The SFRM (see Appendix C) are materials which if released from a package, might present some direct radiation hazard but would present no hazard from radiotoxicity or contamination. Such materials include radioactivity induced in metals or alloys which are not contaminated externally. The general rules covering the packaging and shipment of radioactive materials are found in Part 173, Subpart G "Poisonous Materials and Radioactive Materials: Definition and Preparation," and Part 178, "Shipping Container Specifications," of 49 CFR (38), and in Part 71, "Packaging of Radioactive Material for Transport," of 10 CFR. The dismantling force will be required to package the radioactive materials as covered by 49 CFR 178 and 10 CFR 71. In some cases, this will only require covering pipe ends and equipment nozzles. However, the highly radioactive core components will have to be packaged in caskets with adequate shielding to reduce the radiation levels to the point where they are not hazardous to the transportation workers. Radioactive solids produced from the cleanup purification ion exchangers, evaporator concentrator bottoms, filter cartridges, contaminated tools, rags, plastics, anti-C clothing, instruments, and equipment will be encased in concrete in 55 gallon drums. The liquid wastes will be cycled through filters, ion exchangers, evaporators, and separators until the liquid contamination is concentrated, then they will be solidified by treatment and placed in 55 gallon drums for off-site shipment. Eventually the wastes will be either stored on-site or shipped for burial. The various radioactive materials may be shipped off-site by truck or train.

6.5 Cost Effectiveness

The actual cost effectiveness of the dismantling work is affected by the requirements of working in a radiation area. Factors such as HP requirements, rad waste disposal, material control, and allowable doses result in a decreased productivity factor for individual work. The SWP's contain detailed work procedures which insure that the work is done safely, but these procedures consider cost only as a secondary requirement. The result is a 10 hour job in a contaminated area which could be done in one hour in a nonradioactive plant. Since the health physics staff is required to monitor work tasks which might result in contamination spread, an additional nonproductive worker is included in every work crew. This requirement may be relaxed as work progresses but the radiation monitors remain nonproductive personnel. The stringent material control and waste disposal requirements add additional cost. The familiarity of the worker with radiation safety procedures will decrease the cost slightly but the basic extra costs associated with radiation work can't be reduced. Depending on the task, the cost of labor for each dismantling task in a well controlled job will be from two to five times the costs of the same work in a nonradiation job. In addition, the cost of packaging and shipping radioactive material off-site for burial adds to the costs of the dismantling. Local burial rates at ERR for noncontaminated scrap were $1.75 per cubic yard while burial rates for the radioactive material were $19.50 per cubic yard to which the cost of shipping had to be added (39).

7.0 FUTURE INDUSTRY DECOMMISSIONING REQUIREMENTS

7.1 Postulated U.S. Dismantling Requirements

Forecasts of future nuclear reactor growth in the U.S. predict that 1000 nuclear power plants will be in operation in the year 2000. The AEC projection in "Nuclear Power 1973-2000," dated December 1, 1972 (4) forecast that by 2000 there would be 1200 GWe of installed nuclear power capacity as a "most likely" figure for future planning. The energy crisis, shortage of gas and oil, the proposed strip mining ban in estern states, and the increasing costs of environmental controls on fossil plants have further enhanced the competitive position of nuclear power. These factors will result in a rapid increase in the number of nuclear generating stations. Figure 19 is a graph of the projections of "Nuclear Power, 1973-2000" (4).

Nuclear plants decommissioned to date or plants that are being considered for decommissioning (CVTR, BONUS, Piqua, Pathfinder, ERR, Saxton, and Peach Bottom 1) are small prototype demonstration reactors with minimal electrical output. These reactors were decommissioned because it was not economical to operate them or it was too expensive to repair defective equipment or upgrade the plant safety systems. Experimental plants and demonstration reactors will provide the plants that need to be decommissioned until the present generation of power reactors are ready for dismantling. Assuming that a nuclear plant has an operating life of 40 years and that the plant will then be held in "possession only" status for an average of five years to allow short-lived radioisotopes to decay, the decommissioning market will come of age in 2015. The increase in the number of expected dismantlings per year is shown on Figure 20. Until that time, decommissionings will be limited to

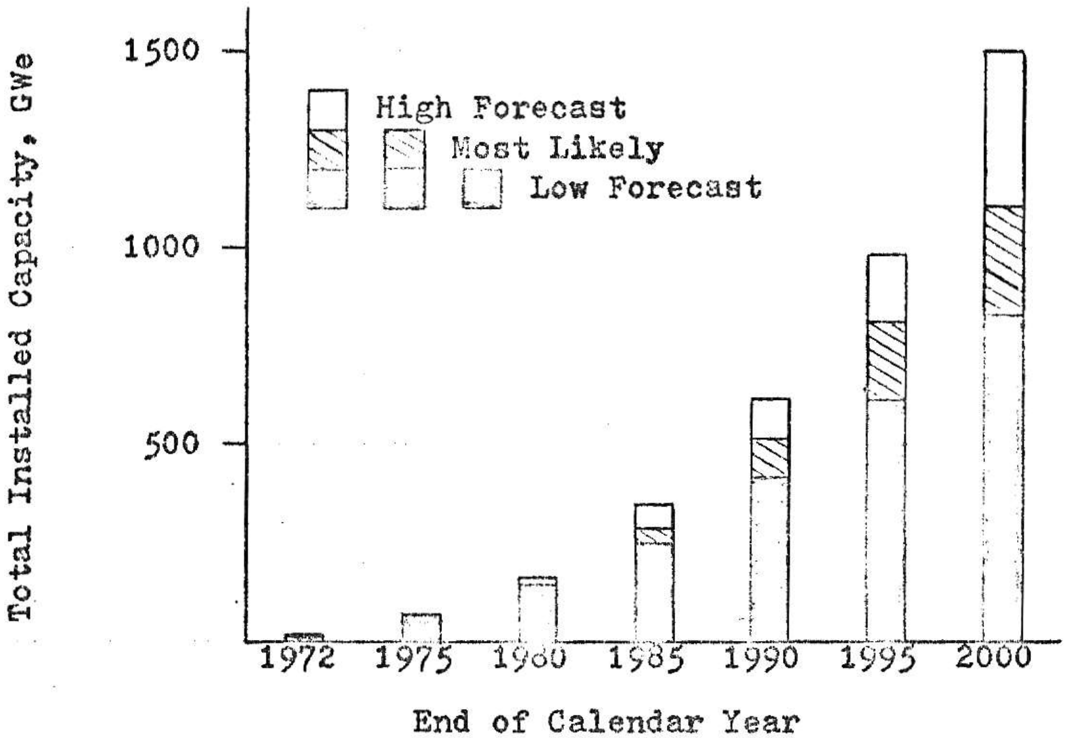

FIGURE 19 - Forecast of Installed Nuclear Capacity - U.S. December 1972

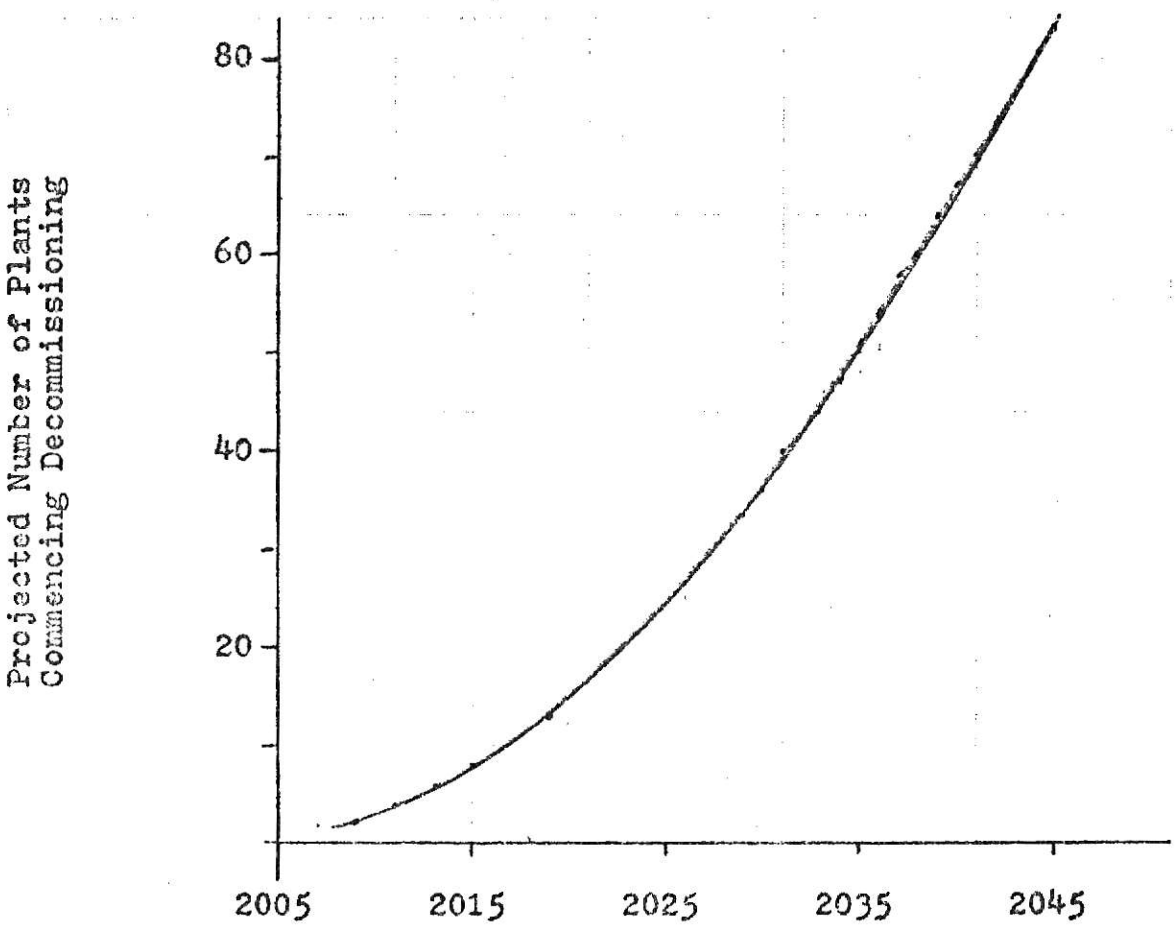

FIGURE 20 - Forecast of Number of Plants to be Decommissioned Commencing 2005

demonstration plants or to plants that are unable to meet "ratcheting" safety requirements due to early design features.

7.2 Growth of Nuclear Service Vendors

The rapid growth of nuclear power plants will result in the formation of companies which provide special services to the nuclear industry. One major work area open to new vendors is the repair, maintenance and modification of radioactive systems in nuclear power plants. This maintenance and modification work will require special remote operated tools and repair equipment. Union Carbide of ORNL and other companies are presently developing tools and procedures for providing the capability of undertaking normal maintenance functions (pipe cutting and welding) remotely (41). ORNL developed the plasma torches and procedures which were used to dismantle the ERR reactor vessel and components remotely. The cost of the specialized tools required for remote repair and maintenance would preclude their being owned by any but the largest utilities.

The heavy work load of the plant health physics staff during a large maintenance task or a dismantling would probably exceed the capability of the staffs of all except the larger utilities. Extra staff members can be hired for the duration of the work from health physics "body shops," or the entire job can be contracted to a health physics contractor who will then assume responsibility for the health physics aspects of the job. These HP contractors should develop expertise in handling contamination control and waste disposal.

The necessity of decontaminating the plant system prior to maintenance or dismantling activities will encourage the growth of decontamination contractors. At present, many contractors are offering these services including the two largest chemical cleaning contractors, Dow and Haliburton. However, it will be several years before adequate experience in the work is gained by the commercial vendors which can be applied specifically to power reactors.

The ability of a utility to develop inhouse capability to handle the activities required by a dismantling will probably depend on the utility's size. The larger utilities which are oriented toward inhouse engineering and design capability will probably develop the expertise required.

7.3 Transportation - Rail/Truck

The transportation of radioactive material will be an old industry when the shipping of appreciable amounts of radioactive material from dismantled plants becomes a significant factor in the transportation market. The packaging and shipment of radioactive materials is under the regulatory jurisdiction of both the AEC and the U.S. Department of Transportation (DOT). The Atomic Energy Act of 1954 and the Transportation of Explosives Act of 1960 provide for this dual jurisdiction. The DOT regulations can be found in Parts 173 and 178 of the Title 49 of the Code of Federal Regulations while Part 71 of 10 CFR contains the AEC rules for "Packaging or Radioactive Material for Transport." The cooperative working relationship between the AEC and the DOT (42) has been exemplified by the AEC-DOT Memorandum of Understanding, which was issued to "enhance the development of consistant regulations and to avoid duplication of effort." The latest Memorandum of Understanding which was signed in March 22, 1973 dealt with the approval of shipping containers for radioactive materials, and the AEC will now be primarily responsible for licensing the containers or packages used by all shippers of fissionable materials and intermediate or large quantities of other radioactive materials. DOT individual approval for packages used for large quantities of radioactive materials will no longer be required. The AEC has issued ammendments to 10 CFR 71 to cover new regulations regarding the containers used for the shipping of radioactive materials. This shift in regulatory function indicates that the AEC will be taking a much more active role in the packaging requirements of future dismantlings. In the ERR dismantling, the packaging of radioactive materials was done under 49 CFR 173 and 49 CFR 178.

At present, the transportation of radioactive materials is an infant industry. New cask designs and transporting rigs are constantly being submitted to the AEC for approval. By 2020, the rules and regulations should be explicit, and the industry should be capable of handling the low risk transportation of radioactive scrap from a dismantling.

The actual transportation of radioactive material may be by rail, barge, or truck. Since only one-half of present reactors under plan or in operation have rail facilities, the transportation of radioactive scrap from a reactor will probably be by truck. The lack of rail facilities at burial grounds and the general problems in railroad transportation will shift the transportation to truck. The transportation by truck can go by overweight truck transport (OWT) (exceeding 73,000 lbs. gross vehicle weight) or by legal weight truck (LWT). The OWT is subject to restrictions on nighttime, holiday, and weekend travel in various states and several states require special permits for the heavier trucks. Therefore, though the OWT can decrease the number of shipments required by 50%, the restrictions and problems of special permits can reduce any advantage in using OWT's (52).

While a LWT can carry a 25 ton payload and an OWT can carry a 35 ton payload, it is possible to load a 100 ton payload on a rail car. The problems associated with rail transportation limit its effectiveness in transporting radioactive materials. Trains are designed so that a single engine (or series of locomotives) will pull a large number of cars. Some railroads require separate locomotives for cars containing radioactive materials and the rates for special trains are expensive. Typical of the problem is the dialogue underway between General Electric Company and Santa Fe Railroad on GE's spent fuel railroad car (43). The comparative evaluation of costs for rail versus motor freight from ERR to Sheffield, Illinois averaged 1.5 times more for the railroad cars than the trucks. A revamping and revitalization of the rail industry in the next 40 years could improve their competitive position, but at present, truck transport appears to have an edge.

7.4 Radioactive Material Disposal

The ultimate disposal of radioactive material appears to be either in place, as in the "internment" option, or remote burial, as in the "removal" option. In any dismantling option, certain rad wastes including demineralizer resins, filter cartridges, and evaporator sludges will be solidified, packaged, and shipped for burial.

There are three commercial radioactive waste disposal companies in operation in the U.S. at present, and they maintain six burial sites. The near-surface burial utilized in these six sites is designed for handling low-level radioactive wastes and SFRM. The Nuclear Engineering Co. maintains burial grounds at Richland, Washington; Beatty, Nevada; Sheffield, Illinois; and Morehead, Kentucky. Nuclear Fuel Services maintains a burial ground at West Valley, New York; while Chem-Nuclear Services, Inc. has a burial site at Barnwell, South Carolina.

These burial grounds receive a wide variety of waste materials, and except for the prohibition on burial of liquids and limitations on radionuclide content of the waste, there is little or no restriction on what may be buried in these sites. Practically any material which can be packaged under DOT and AEC regulations for transportation can be buried at these sites.

These burial grounds are licensed by the AEC or by an Agreement State (an Agreement State has signed a contract with the AEC where it takes over certain responsibilities for regulation of source, byproduct, and limited amounts of SNM within the state), and are subject to the requirements of 10 CFR 20.302. Section 20.302 requires that a complete analysis of the site be made, that specific procedures for burial be approved by the AEC, that a safety analysis of the site be performed, and that the site be owned by either the State or the Federal government. (All six of the existing sites are owned by the State governments). The last requirement assures that the site will be maintained as the States have agreed to be responsible for perpetual care of the burial grounds if and when the burial operations are terminated (44). Particular care is given in the site investigation to the geology and hydrology of the site. The ion exchange properties of the soil, the movement of ground water, and the possibility of contamination of water supplies are examined extensively to preclude any possibility of hazard to the public health.

The disposal of high level radioactive waste is not a problem associated with the reactor decommissioning. The only high level wastes will result from the fuel reprocessing, and they will be handled at the reprocessing plant under normal operating procedures.

The cost of burial at an AEC licensed burial ground has been \$.80 per cubic foot, f.o.b. the burial site. This is based on a maximum surface radiation level of 200mR/hr and a maximum container weight of 15 tons. Heavier containers and larger radiation levels can be handled at additional cost. Packages should not contain explosives or more than specific amounts of SNM. The materials should be packaged in compliance with DOT and AEC regulations. Each shipment should be accompanied by the proper forms containing a complete description of the radioactive material contained in the shipment.

7.5 EPA, Envirnomental and Dual Site Considerations

Though EPA has been given authority for regulating environmental concerns, the AEC still have primary responsibility for licensing power reactors. In order to implement EPA regulation, the AEC has required that Environmental Reports be prepared in accordance with Regulatory Guide 4.2. The Envirnmental Report, submitted with the PSAR, is required to review the environmental impact of dismantling. Typically, the environmental impact of a shutdown facility will be less than the impact of an operating plant. In addition, an Environmental Report (ER) must be submitted for approval with the Decommissioning Plan. This ER will address the environmental aspects of the dismantled facility, will review the possible impact from transportation accidents involving radioactive material, and will cover such problems as on-site radioactive material storage. In general, the environmental impact of a dismantled facility will result in an improvement in the environmental condition of the site. For example, cooling tower plumes or hot circulating water discharges into rivers will be terminated. Since the normal maintenance of the plant will require periodic shutting down of the facility, the environmental consequences of such actions will have been analyzed in the plant ER. They should have little impact if the shutdown is made permanent.

Two factors which will affect the economics of a dismantling are the future use of the dismantled reactor site and the nearness of a second facility (power reactor) to the dismantled plant. When a reactor is part of a multiple plant site, the economics and selection of the option for the decommissioning will be influenced by the continued operation of the other plants on the site. Future surveillance costs of the minimum storage option would be decreased, and the dismantled plant could serve as an equipment supply source in an emergency on one of the operating units. The use of radiation detection equipment, laboratory and laundry facilities and other facilities at the operating plants would decrease the cost of the dismantling. The decreased surveillance and surplus value of the plant components would tend to promote the storage option.

The final use of the plant site should also be considered. The opposition of environmental groups to new power plant sites, the high cost of land, the expense of lengthy litigation required to use some new sites for power plants, and the general attitude of the public to save "unspoiled" sites in this time of general scarcity, are all in conflict with the rapidly expanding need for more energy, and hence, more power plants. As fossil fuels become more scarce (particularly natural gas and petroleum), and as costs increase, the shift will be toward increased use of electric power, both nuclear and coal fired. This shift will require utilization of all available sites for generating stations. As a consequence, the site of a decommissioned plant will become a viable site for a new generating facility. In addition to already owning the land, the utility should experience minimum opposition from environmentalists in building a plant on an old site. State and Federal Water Use Permits, Discharge Permits, and other regulatory permits required for each new plant should receive swift approval as they will be, for the most part, extensions of already existing conditions. The utilization of existing substations and transmission lines should also encourage reutilization of the site. Circulating water cooling facilities, be they tower, rivers, or lakes can be utilized by the new plant at considerable savings in plant construction costs.

The construction of a new power plant on the site of a decommissioned facility will require careful planning and due consideration will have to be given the effects of the construction activities on the integrity of the decommissioned plant. A nuclear power generating station will generally have enough land within the exclusion zone of the site to locate several nuclear power plants. The factors which limit the site's generation capability will be site cooling capability, environmental effects, and local electric system load. Generally, locating a new power plant "on top" of an old site will mean that the plant will be next to it. Some items which might be utilized with a minimum of cost, such as intake structures or intake canal, the cooling towers (or cooling tower basins), discharge canal, and electric switching facilities might be used by the new plant. In general, only those items which are normally passive will be considered for use with the new plant.

The AEC, in Regulatory Guide 1.86, has listed as an option, the conversion of the steam system to a new nuclear or fossil fired steam supply system. This was done for the Pathfinder Power Plant and initial plans on decommissioning the Peach Bottom Unit 1 mentioned a similiar possibility. However in each case, the plants were first generation demonstration reactors which weren't economical to run. Unless there is a complete moratorium on the operation of all nuclear plants, there should be no reason to switch from a nuclear to a fossil fueled plant unless the costs of fossil fuels drop to an extraordinary degree. The utilization of a 40 year old turbine generator in a new $600,000,000 plant (660 million less 60 million TG cost) does not appear logical. This is true of any used equipment. A power generating station is a complex plant made up of many small components. Since even a small component malfunction can shut the unit down and result in lost revenue, component quality should not be compromized unless absolutely necessary. For this reason, the dismantled plant could serve as a source of replacement parts only in an emergency when new parts were not available, and the AEC would probably "frown" on using any salvaged parts in safety related systems.

If land is not available, complete removal of the old plant will result in a 3% increase in plant cost. However, the delay in construction start and the increased cost of escavation and piling installation would probably raise that to a 6% increase in cost.

8.0 CONCLUSION

The decommissioning of a nuclear power reactor is a job of considerable magnitude. The costs and work required to obtain the required objectives of a Decommissioning Plan are done at a time in the life of a plant where there will be no "return" on the investment other then the minimization of costs for services required by the AEC regulations. These costs, in general, result from requirements which are not applicable to nonoperating facilities. As a result, the license will dismantle the plant through economic coercion, and the AEC is assured of a satisfactory dismantling job.

There seems to be a general shift toward regulations or positions which will favor minimum "storage" dismantling. A decrease in the allowable occupational radiation dose would increase the attractiveness of the storage option which would minimize radiation exposure. The proliferation of multi-unit sites will decrease the surveillance costs of a "storage" dismantling at such a site and the ability to use instruments and other items salvaged from a storage site in an emergency would prove a bonus for the "storage" dismantling. Minimization of costs, manpower requirements, and a shortened schedule will be increasingly important factors if the shift toward nuclear power results in shortages of adequately trained manpower. The effective use of available manpower may be the deciding factor in the dismantling option selected.

A review of the conflicting factors which arise when a nuclear unit is shutdown reveals that the factors which encourage rapid decommissioning are artificial factors created by the regulations of the AEC. The factors which encourage delay of the job, in particular the decay of the short-lived radionuclides, cannot be altered by regulatory changes. A licensee can decrease some of the AEC requirements by applying for a "possession only" license with Tech Specs modifications, but the licensee will still have to comply with numerous regulations which aren't necessary to assure safety. A possible solution is the creation of a short term "interim pre-dismantling" license which could be issued with a minimum amount of work. The interim license would require disabling of the reactor and surveillance of the facility for any contamination spread. Skeleton maintenance and operating crews could maintain the plant in hold status until the plant is dismantled. The interim license would require only minimum decontamination, removal of all SNM and high level radioactive wastes, and the only components modified would be the control rod drives. After a given license period, five to ten years, the licensee would be required to submit a Decommissioning Plan and all the supporting documents. The interim license would enable the short-lived radionuclides to decay, and the costs and radiation exposures resulting from the dismantling would decrease significantly. The possibility of obtaining this change in licensing philosophy is enhanced by the economic consequences of maintaining the operating license and the decrease in apprehension concerning the safety of nuclear power after forty years of reactor operating experience has been accumulated.

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APPENDIX A - METHOD OF ANALYSIS OF INDUCED ACTIVITY

As shown in Figure 6 the operation of the hypothetical PWR took place in nine periods, ranging in length from about two to eight years over the time from 1980 to 2020.

In an operating period n with average power P_n and duration T_n, the rate of production of radionuclide i with decay constant λ_i is given by

$$\frac{dN_i}{dt} = P_n \phi \Sigma_i - \lambda_i N_i$$

where ϕ is the neutron flux per megawatt and Σ_i is the macroscopic cross section for production or radionuclide i. This equation is integrated over the duration of the period to give the number of atoms of nuclide i present at the end of the period.

$$N_i = \frac{P_n \phi \Sigma_i}{\lambda_i} (1 - e^{-\lambda_i T_n}).$$

A measurement at a time τ_n after the end of period n would show the number of atoms of nuclide i present as

$$N_i = \frac{P_n \phi \Sigma_i}{\lambda_i} (1 - e^{\lambda_i T_n}) e^{-\lambda_i \tau_n}$$

The contributions from all nine operating periods must be summed.

The activity in curies of radionuclide i from neutron interaction in parent nuclide k in component j in the hypothetical PWR which would be measured in 2020 as:

$$A_{ijk} = K \phi_j \Sigma_{kji} V_i \sum_{n=1}^{9} P_n (1 - e^{-\lambda_i T_n}) e^{-\lambda_i \tau_n}$$

where $K = 1/(3.7 \times 10^{10})$ curies/(disintegration/sec).

ϕ_j = neutron flux in component j (neutrons/cm^2 - sec - MW)

Σ_{kji} = macroscopic cross section of parent nuclide k in component j to to produce radionuclide i (cm^{-1})

V_j = volume of component j in cm^3

but Σ_{kji} may be represented as:

$$\Sigma_{kji} = \frac{N_o}{A_k} I_k f_{kj} \rho_j \sigma_{ki}$$

where N_o = 0.6023×10^{24} atoms/mole

A_k = atomic weight of parent element k in g/mole

f_{kj} = weight fraction of parent element k in component j

ρ_j = density of component j in g/cm^3

I_k = abundance of parent isotope k in parent element

σ_{ki} = microscopic cross section for production of nuclide i by neutron interaction with isotope k.

Finally, with W_j as the weight in g of component j, we have

$$A_{ijk} = KN_o W_j f_{kj} \frac{I_k}{A_k} \sigma_{ki} \phi_j \sum_{n=1}^{9} P_n (1 - e^{-\lambda_i T_n}) e^{-\lambda_i \tau_n}$$

where

$$\sum_{n=1}^{9} P_n (1 - e^{-\lambda_i T_n}) e^{-\lambda_i \tau_n}$$

is the power factor.

APPENDIX B - CODE USED TO CALCULATE ACTIVITY FOR PATHFINDER DECOMMISSIONING

```
                                  80 COLUMN LISTING
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
*********************************************************************************

// JOB
// FOR
*EXTENDED PRECISION
*ONE WORD INTEGERS
*LIST ALL
*TRANSFER TRACE
*ARITHMETIC TRACE
*IOCS(CARD,1403 PRINTER)
** N-29  ACTIVITY CHECK
C     ------------------------------------------------------------
C     N29BV - ACTIVITY CHECK
C     ------------------------------------------------------------
      DIMENSION XAG(15),XC(15),XCA(15),XCO(15),XCU(15),XCR(15),XFE(15)
      DIMENSION XH(15),XK(15),XMN(15),XNA(15),XNI(15),XPB(15),XS(15)
      DIMENSION XSN(15),XZN(15),XZR(15),ITITL(35),TYPE(45)
      DIMENSION DACT1(32),DACT2(32),DACT3(14)
      DIMENSION PERCT(15),TAC(39),DACT(78)
C....
      COMMON AVPOW(4),XLVTM(4),DCTIM(4),CONS,AVGNO,ADJWT,SUMPF
C....
      DATA TYPE /'304 ','SS  ','    ','304L',' SS ','    ','303 ','SS  '
     1,'    ','316 ','SS  ','    ','316L',' SS ','    ','405 ','SS  ',
     2'    ','416 ','SS  ','    ','A212',' GR ','8   ','A105',' GR ',
     3'11  ','SPEC',' OVE','RLAY','A335',' GR ','22  ','USS ','T-1 ',
     4'    ','B167',' INC','ONEL','CONC','RETE','    ','WATE','R   ',
     5'    '/
      DATA XAG /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000/
      DATA XC  /0.0008, 0.0003, 0.0015, 0.0010, 0.0003, 0.0008, 0.0015,
     1  0.0033, 0.0035, 0.0005, 0.0015, 0.0020, 0.0015, 0.0000, 0.0000/
      DATA XCA /0.0000, 0.0000, 0.0050, 0.0000, 0.0000, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0440, 0.0000/
      DATA XCO /0.0020, 0.0020, 0.0020, 0.0020, 0.0020, 0.0020, 0.0020,
     1  0.0001, 0.0020, 0.0020, 0.0020, 0.0020, 0.0020, 0.0000, 0.0000/
      DATA XCU /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000/
      DATA XCR /0.1900, 0.1700, 0.1800, 0.1700, 0.1700, 0.1250, 0.1300,
     1  0.0000, 0.0000, 0.2100, 0.0250, 0.0070, 0.1600, 0.0000, 0.0000/
      DATA XFE /0.7700, 0.7700, 0.7900, 0.7700, 0.7800, 0.8550, 0.8500,
     1  0.9840, 0.9880, 0.6920, 0.9650, 0.9700, 0.0900, 0.0140, 0.0000/
      DATA XH  /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0100, 0.1111/
      DATA XK  /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0130, 0.0000/
      DATA XMN /0.0200, 0.0200, 0.0200, 0.0200, 0.0200, 0.0100, 0.0125,
     1  0.0100, 0.0090, 0.0250, 0.0050, 0.0100, 0.0100, 0.0000, 0.0000/
      DATA XNA /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0160, 0.0000/
      DATA XNI /0.1000, 0.1000, 0.0900, 0.1300, 0.1200, 0.0000, 0.0000,
     1  0.0000, 0.0000, 0.1000, 0.0000, 0.0100, 0.7500, 0.0000, 0.0000/
```

```
                              80 COLUMN LISTING
0    0    1    1    2    2    3    3    4    4    5    5    6    6    7    7
1    5    0    5    0    5    0    5    0    5    0    5    0    5    0    5
...............................................................................

      DATA XPB /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000/
      DATA XS  /0.0003, 0.0003, 0.0007, 0.0003, 0.0003, 0.0003, 0.0003,
     1 0.0004, 0.0005, 0.0003, 0.0003, 0.0005, 0.0007, 0.0000, 0.0000/
      DATA XSN /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000/
      DATA XZN /0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000,
     1 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000/
      DATA XZR /0.0000, 0.0000, 0.0060, 0.0000, 0.0000, 0.0000, 0.0060,
     1 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000, 0.0000/
      DATA DACT1 /'P -',' 33','CR-',' 51','AR-',' 37','CD-','115','FE-',
     1' 59','HG-','203','IN-','114','SR-',' 89','Y -',' 91','ZR-',' 95',
     2'CO-',' 58','S -',' 35','SN-','113','SN-','123','CA-',' 45','ZN-',
     3' 65'/
      DATA DACT2 /
     1        'AG-','110','SN-','119','CO-',' 57','MN-',' 54','V -',' 49',
     2'CD-','109','SB-','105','FE-',' 55','NA-',' 22','TL-','204','SN-',
     3'121','CO-',' 60','H -','  3','CD-','113','NI-',' 63','AG-','108'/
      DATA DACT3 /
     1'AR-',' 39','C -',' 14','CA-',' 41','CL-',' 36','ZR-',' 93','BE-',
     2' 10','PD-','107'/
      DO  75 I=1,32
   75 DACT(I)=DACT1(I)
      DO  76 I=33,64
   76 DACT(I)=DACT2(I-32)
      DO  77 I=65,78
   77 DACT(I)=DACT3(I-64)
C.... LOGICAL UNIT ASSIGNMENTS
      NR=2
      NW=5
C....
      CALL DATE (IM,ID,IY)
  900 READ(NR,1010) ITITL,NCASE
 1010 FORMAT(35A2,5X,I5)
      IF(NCASE)999,999, 10
   10 READ(NR,1000)(XLVTM(I),I=1,4),(DCTIM(I),I=1,4)
 1000 FORMAT(7(E8.0,1X),E8.0)
      READ(NR,1000)(AVPOW(I),I=1,4),FFLUX,SFLUX,TLWT
      READ(NR,1030) PERCT
 1030 FORMAT(8(F6.0,4X))
      DO 80 I=1,15
   80 PERCT(I)=PERCT(I)/100.
C.... AVAGADRO'S NUMBER
      AVGNO=0.6023E+24
C....
      CONS=1.0/(3.7E+10)
C.... CHANGE WEIGHT FROM LBS TO GRAMS
      ADJWT=453.59*TLWT
      EAGT =0.
      ECT  =0.
```

```
                        HO COLUMN LISTING
0    0    1    1    2    2    3    3    4    4    5    5    6    6    7    7    8
1    5    0    5    0    5    0    5    0    5    0    5    0    5    0    5    0
.................................................................................
      ECAT =0.
      ECOT =0.
      ECUT =0.
      ECRT =0.
      EFET =0.
      EHT  =0.
      EKT  =0.
      EMNT =0.
      ENAT =0.
      ENIT =0.
      EPBT =0.
      EST  =0.
      ESNT =0.
      EZNT =0.
      EZRT =0.
      DO  20 I=1,15
      EAGT = EAGT + (PERCT(I)*XAG(I))
      ECT  = ECT  + (PERCT(I)*XC (I))
      ECAT = ECAT + (PERCT(I)*XCA(I))
      ECOT = ECOT + (PERCT(I)*XCO(I))
      ECUT = ECUT + (PERCT(I)*XCU(I))
      ECRT = ECRT + (PERCT(I)*XCR(I))
      EFET = EFET + (PERCT(I)*XFE(I))
      EHT  = EHT  + (PERCT(I)*XH (I))
      EKT  = EKT  + (PERCT(I)*XK (I))
      EMNT = EMNT + (PERCT(I)*XMN(I))
      ENAT = ENAT + (PERCT(I)*XNA(I))
      ENIT = ENIT + (PERCT(I)*XNI(I))
      EPBT = EPBT + (PERCT(I)*XPB(I))
      EST  = EST  + (PERCT(I)*XS (I))
      ESNT = ESNT + (PERCT(I)*XSN(I))
      EZNT = EZNT + (PERCT(I)*XZN(I))
      EZRT = EZRT + (PERCT(I)*XZR(I))
   20 CONTINUE
C.... SUM TOTAL PERCENTAGE OF MATERIALS AVAILABLE
C.... CALCULATION OF ACTIVITY DUE TO P -33
      CALL N2901 (10.19, 0.00023,0.0013,0.0, 0.065,0.000002,0.0,
     1EST,EST,0.0, FFLUX,FFLUX,0.0, TAC(1))
C.... CALCULATION OF ACTIVITY DUE TO CR-51
      CALL N2901 (9.12, 0.001056,0.000848,0.01605, 0.00074,15.9,0.00003,
     1EFET,ECRT,ECRT,FFLUX,SFLUX,FFLUX, TAC(2))
C.... CALCULATION OF ACTIVITY DUE TO AR-37
      CALL N2901 (7.37, 0.02418,0.02381,0.0, 0.0067,0.0002,0.0,
     1ECAT,EKT,0.0, FFLUX,FFLUX,0.0, TAC(3))
C.... CALCULATION OF ACTIVITY DUE TO CO-115
      CALL N2901 (5.87, 0.002008,0.0,0.0, 0.000002,0.0,0.0,
     1ESNT,0.0,0.0, FFLUX,0.0,0.0, TAC(4))
C.... CALCULATION OF ACTIVITY DUE TO FE-59
      CALL N2901 (5.63, 0.0000591,0.0006304,0.01697, 1.01,0.00018,.0017,
     1EFET,ENIT,ECOT,SFLUX,FFLUX,FFLUX, TAC(5))
```

80 COLUMN LISTING

```
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
********************************************************************************

C.... CALCULATION OF ACTIVITY DUE TO HG-203
      CALL N2901 (5.37, 0.01267,0.0,0.0, 0.0000001,0.0,0.0,
     1EPBT,0.0,0.0,FFLUX,0.0,0.0, TAC(6))
C.... CALCULATION OF ACTIVITY DUE TO IN-114
      CALL N2901(5.06, 0.0000573,0.0000295,0.0, 0.0003,0.00003,0.0,
     1ESNT,ESNT,0.0,FFLUX,FFLUX,0.0, TAC(7))
C.... CALCULATION OF ACTIVITY DUE TO SR-89
      CALL N2901 (5.02, 0.001876,0.0,0.0, 0.000074,0.0,0.0,
     1EZRT,0.0,0.0, FFLUX,0.0,0.0, TAC(8))
C.... CALCULATION OF ACTIVITY DUE TO Y -91
      CALL N2901 (4.41, 0.01231,0.001907,0.0, 0.00028,0.000019,0.0,
     1EZRT,EZRT,0.0,FFLUX,FFLUX,0.0, TAC(9))
C.... CALCULATION OF ACTIVITY DUE TO ZR-95
      CALL N2901 (3.89, 0.001907,0.000307,0.0, 0.076,0.004,0.0,
     1EZRT,EZRT,0.0, SFLUX,FFLUX,0.0, TAC(10))
C.... CALCULATION OF ACTIVITY DUE TO CO-58
      CALL N2901 (3.57, 0.01697,0.01155,0.0, 0.0002,0.073,0.0,
     1ECOT,ENIT,0.0, FFLUX,FFLUX,0.0, TAC(11))
C.... CALCULATION OF ACTIVITY DUE TO S -35
      CALL N2901 (2.92, 0.0000044,0.0013,0.0, 0.00017,0.270,0.0,
     1EST,EST,0.0,FFLUX,SFLUX,0.0, TAC(12))
C.... CALCULATION OF ACTIVITY DUE TO SN-113
      CALL N2901 (2.15, 0.0000851,0.0000573,0.0, 0.9,0.0003,0.0,
     1ESNT,ESNT,0.0, SFLUX,FFLUX,0.0, TAC(13))
C.... CALCULATION OF ACTIVITY DUE TO SN-123
      CALL N2901 (2.03, 0.0003993,0.0005063,0.0, 0.001,0.0024,0.0,
     1ESNT,ESNT,0.0, SFLUX,FFLUX,0.0, TAC(14))
C.... CALCULATION OF ACTIVITY DUE TO CA-45
      CALL N2901 (1.53, 0.0005339,0.0000025,0.0, 0.67,0.00028,0.0,
     1ECAT,ECAT,0.0, SFLUX,FFLUX,0.0, TAC(15))
C.... CALCULATION OF ACTIVITY DUE TO ZN-65
      CALL N2901 (1.04, 0.007639,0.0,0.0, 0.407,0.0,0.0,
     1EZNT,0.0,0.0, SFLUX,0.0,0.0, TAC(16))
C.... CALCULATION OF ACTIVITY DUE TO AG-110
      CALL N2901 (1.02,0.004420,0.0,0.0, 2.65,0.0,0.0,
     1EAGT,0.0,0.0, SFLUX,0.0,0.0, TAC(17))
C.... CALCULATION OF ACTIVITY DUE TO SN-119
      CALL N2901 (1.01, 0.002008,0.002759,0.0,.01,0.001,0.0,
     1ESNT,ESNT,0.0, SFLUX,FFLUX,0.0, TAC(18))
C.... CALCULATION OF ACTIVITY DUE TO CO-57
      CALL N2901 (.938, 0.01155,0.01155,0.0, 0.073,0.00003,0.0,
     1ENIT,ENIT,0.0, FFLUX,FFLUX,0.0, TAC(19))
C.... CALCULATION OF ACTIVITY DUE TO MN-54
      CALL N2901 (.870, 0.0182,0.001056,0.0, 0.0002,0.051,0.0,
     1EMNT,EFET,0.0, FFLUX,FFLUX,0.0, TAC(20))
C.... CALCULATION OF ACTIVITY DUE TO V -49
      CALL N2901 (.767, 0.000848,0.000848,0.0, 0.000007,0.0149,0.0,
     1ECRT,ECRT,0.0, FFLUX,FFLUX,0.0, TAC(21))
C.... CALCULATION OF ACTIVITY DUE TO CD-109
      CALL N2901 (.537, 0.0000851,0.0,0.0, 0.000065,0.0,0.0,
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                                80 COLUMN LISTING
0    0    1    1    2    2    3    3    4    4    5    5    6    6    7    7    8
1    5    0    5    0    5    0    5    0    5    0    5    0    5    0    5    0
*******************************************************************************

     1ESNT,0.0,0.0, FFLUX,0.0,0.0, TAC(22))
C.... CALCULATION OF ACTIVITY DUE TO SB-125
      CALL N2901 (.347, 0.0005063,0.0,0.0, 0.14,0.0,0.0,
     1ESNT,0.0,0.0, SFLUX,0.0,0.0, TAC(23))
C.... CALCULATION OF ACTIVITY DUE TO FE-55
      CALL N2901 (.257, 0.001056,0.01639,0.01155, 2.8,0.00007,0.0034,
     1EFET,EFET,ENIT, SFLUX,FFLUX,FFLUX, TAC(24))
C.... CALCULATION OF ACTIVITY DUE TO NA-22
      CALL N2901 (.244, 0.04348,0.0,0.0, 0.000007,0.0,0.0,
     1ENAT,0.0,0.0, FFLUX,0.0,0.0, TAC(25))
C.... CALCULATION OF ACTIVITY DUE TO TL-204
      CALL N2901 (.178, 0.0000661,0.001267,0.0, 0.0003,0.0000001,0.0,
     1EPBT,EPBT,0.0, FFLUX,FFLUX,0.0, TAC(26))
C.... CALCULATION OF ACTIVITY DUE TO SN-121
      CALL N2901 (.139, 0.002759,0.0003993,0.0, 0.001,0.0019,0.0,
     1ESNT,ESNT,0.0, SFLUX,FFLUX,0.0, TAC(27))
C.... CALCULATION OF ACTIVITY DUE TO CO-60
      CALL N2901 (.131, 0.01697,0.004464,0.0002045, 39.1,0.002,0.0007,
     1ECOT,ENIT,ENIT, SFLUX,FFLUX,FFLUX, TAC(28))
      TAC(28)=TAC(28)+(CONS*AVGNO*ADJWT*ECUT*0.01087*0.00076*FFLUX*
     1SUMPF)/1.0E+24
C.... CALCULATION OF ACTIVITY DUE TO H -3
      CALL N2901 (.0565, 0.0001984,0.0,0.0, 0.0005,0.0,0.0,
     1EHT,0.0,0.0, SFLUX,0.0,0.0, TAC(29))
C.... CALCULATION OF ACTIVITY DUE TO CD-113
      CALL N2901 (.0495, 0.001203,0.0,0.0, 0.0000021,0.0,0.0,
     1ESNT,0.0,0.0, FFLUX,0.0,0.0, TAC(30))
C.... CALCULATION OF ACTIVITY DUE TO NI-63
      CALL N2901 (.00753, 0.0001674,0.0006304,0.01087, .00045,15.0,.009,
     1ENIT,ENIT,ECUT, FFLUX,SFLUX,FFLUX, TAC(31))
C.... CALCULATION OF ACTIVITY DUE TO AG-108
      CALL N2901 (.00693, 0.004843,0.00442,0.0, 1.24,0.0011,0.0,
     1EAGT,EAGT,0.0, SFLUX,FFLUX,0.0, TAC(32))
C.... CALCULATION OF ACTIVITY DUE TO AR-39
      CALL N2901 (.00267, 0.0001597,0.02381,0.0, 0.0024,0.024,0.0,
     1ECAT,EKT,0.0, FFLUX,FFLUX,0.0, TAC(33))
C.... CALCULATION OF ACTIVITY DUE TO C -14
      CALL N2901 (.000121, 0.000854,0.000025,0.0, 0.000798,0.208,0.0,
     1ECT,ECT,0.0, SFLUX,SFLUX,0.0, TAC(34))
C.... CALCULATION OF ACTIVITY DUE TO CA-41
      CALL N2901 (6.3E-5, 0.0243,0.000153,0.0, 0.00381,0.00004,0.0,
     1ECAT,ECAT,0.0, SFLUX,FFLUX,0.0, TAC(35))
C.... CALCULATION OF ACTIVITY DUE TO CL-36
      CALL N2901 (.231E-5, 0.02381,0.02381,0.0, 0.006,0.00381,0.0,
     1EKT,EKT,0.0, FFLUX,SFLUX,0.0, TAC(36))
C.... CALCULATION OF ACTIVITY DUE TO ZR-93
      CALL N2901 (.729E-6, 0.001862,0.001853,0.0, 0.222,0.0033,0.0,
     1EZRT,EZRT,0.0, SFLUX,FFLUX,0.0, TAC(37))
C.... CALCULATION OF ACTIVITY DUE TO BE-10
      CALL N2901 (.257E-6, 0.000854,0.0,0.0, 0.0026,0.0,0.0,
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                                    80 COLUMN LISTING
 0    1    1    2    2    3    3    4    4    5    5    6    6    7    7    8
 5    0    5    0    5    0    5    0    5    0    5    0    5    0    5    0
********************************************************************************

     1ECT,0.0,0.0, FFLUX,0.0,0.0, TAC(38))
**** CALCULATION OF ACTIVITY DUE TO PD-107
     CALL N2901 (.99E-7, 0.004843,0.0,0.0, 0.0009,0.0,0.0,
     1EAGT,0.0,0.0, FFLUX,0.0,0.0, TAC(39))
**** SUMMATION OF TOTAL ACTIVITIES
     TOTAL=0.0
     DO  30 I=1,39
     TOTAL=TOTAL+TAC(I)
  30 CONTINUE
**** WRITE INPUT
     WRITE(NW,2000) NCASE,IM,ID,IY,ITITL
2000 FORMAT('1',T9,'BLACK AND VEATCH CONSULTING ENGINEERS',T74,'CASE ',
     1I3,/,T9,'ACTIVITY LEVEL CALCULATIONS',T74,A2,'/',A2,'/',A2,//
     2T9,35A2,/)
     WRITE(NW,2600)
2600 FORMAT(///,T9,'INPUT PARAMETERS',/ T9,16('-'))
     WRITE(NW,2610)FFLUX
2610 FORMAT(' ',T9,'FAST FLUX (MW)',T70,E12.4)
     WRITE(NW,2620)SFLUX
2620 FORMAT(' ',T9,'SLOW FLUX (MW)',T70,E12.4)
     WRITE(NW,2630)(XLVTM(I),I=1,4)
2630 FORMAT(' ',T9,'LIVE TIME (YR)',     T38,4(2X,E9.3))
     WRITE(NW,2640)(DCTIM(I),I=1,4)
2640 FORMAT(' ',T9,'DECAY TIME (YR)',T38,4(2X,E9.3))
     WRITE(NW,2650)(AVPOW(I),I=1,4)
2650 FORMAT(' ',T9,'AVERAGE POWER GENERATED (MW)',T38,4(3X,F8.2))
     WRITE(NW,2660)TLWT
2660 FORMAT(' ',T9,'TOTAL WEIGHT (LB)',T70,F12.3)
     WRITE(NW,2100)
2100 FORMAT('0',T9,'MATERIAL',T24,'PERCENTAGE',/)
     I1=-2
     DO  50 I=1,15
     I1=I1+3
     I2=I1+2
     PERCT(I)=PERCT(I)*100.
     IF(PERCT(I)) 50, 50, 40
  40 WRITE(NW,2110)(TYPE(J),J=I1,I2),PERCT(I)
2110 FORMAT(T9,3A4,T25,F7.3)
  50 CONTINUE
     WRITE(NW,2700)
2700 FORMAT('0',T9,'CALCULATED PARAMETERS',/,T9,21('-'))
     WRITE(NW,3000)
3000 FORMAT(1H0,T9,'ELEMENT',T24,'LEVEL'/)
     I1=-1
     DO 300 I=1,39
     I1=I1+2
     I2=I1+1
     IF(TAC(I))310,300,310
 310 WRITE(NW,3010)(DACT (J),J=I1,I2),TAC(I)
3010 FORMAT(1H ,T9,2A3,T21,E11.4)
```

80 COLUMN LISTING

```
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
*****************************************************************

  300 CONTINUE
      WRITE(NW,2670)TOTAL
 2670 FORMAT(///,T9,'TOTAL ACTIVITY',T67,E15.4)
      GO TO 900
  999 CALL EXIT
      END
// DUP
*DELETE             N29BV
*STORE        WS  UA N29BV
// JOB
// FOR
*EXTENDED PRECISION
*ARITHMETIC TRACE
*TRANSFER TRACE
*ONE WORD INTEGERS
*LIST ALL
**    ACTIVITY CHECK SUBROUTINE
C     ---------------------------------------------------------------
      SUBROUTINE N2901 (XLAMB,RFWTA,RFWTB,RFWTC,FLUXA,FLUXB,FLUXC,ELMA,
     1ELMB,ELMC,FLXA,FLXB,FLXC,ACTIV)
C     ---------------------------------------------------------------
C.... ACTIVITY CHECK SUBROUTINE
      DIMENSION APFC(4)
      COMMON AVPOW(4),XLVTM(4),OCTIM(4),CONS,AVGNO,ADJWT,SUMPF
C....
      I=7777
      SUMPF=0.0
      DO 100 I=1,4
      EXP3= 1.0/(EXP(XLAMB*OCTIM(I)))
      APFC(I) = AVPOW(I) * (1.0-1.0/(EXP(XLAMB*XLVTM(I))))*EXP3
      SUMPF = SUMPF + APFC(I)
  100 CONTINUE
C
      ACTVA = CONS*AVGNO*ADJWT*ELMA*RFWTA*FLUXA*FLXA*SUMPF
      ACTVB = CONS*AVGNO*ADJWT*ELMB*RFWTB*FLUXB*FLXB*SUMPF
      ACTVC = CONS*AVGNO*ADJWT*ELMC*RFWTC*FLUXC*FLXC*SUMPF
C
      ACTIV =(ACTVA + ACTVB + ACTVC)/1.0E+24
      RETURN
      END
// JOB
// FOR
*EXTENDED PRECISION
*ONE WORD INTEGERS
*LIST ALL
*TRANSFER TRACE
*ARITHMETIC TRACE
*IOCS(1403 PRINTER,CARD)
**    N-29 ACTIVITY CHECK
C     ---------------------------------------------------------------
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                                     80 COLUMN LISTING
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
*******************************************************************************
C
      DIMENSION ITITL(35),AVPOW(4),XLVTM(4),DCTIM(4)
      DIMENSION PERCT(20)
      DIMENSION TYPE(20),ELEM(100,20),ID(100,4),XIKAK(100),FRCS(100)
      DIMENSION TRCS(100),ICODE(100),ELTOT( 50),XLAMB(100)
C
      DATA IEOD /'***'/, IBLNK /'   '/
      DATA TYPE /'304 SS','304LSS','303 SS','316 SS','316LSS','405 SS',
     1'416 SS','A212-B','A10511','OVRLAY','A33522','USS T1','B-167 ',
     2'CONCRE','WATER ','AGGREG'/
C
C.... LOGICAL UNIT ASSIGNMENTS
      NR=2
      NW=5
C....
      CALL DATE (IM,ID,IY)
C.... NUMBER OF MATERIALS.
      NMATL=16
C.... NUMBER OF ELEMENTS.
      NELEM=30
C.... READ MATERIALS INVESTIGATED BY REACTOR AND TRACE MATERIAL PERCENTS
      DO  75 I=1,NELEM
      READ(NR,1020)  (ELEM(I,J),J=1,NMATL)
   75 WRITE(NW,2120) (ELEM(I,J),J=1,NMATL)
C.... READ NUCLIDES,IK/AK,FAST AND THERMAL REACTION CROSS SECTIONS, AND
C           ELEMENT CODE NUMBERS.
      DO  76 I=1,200
      READ(NR,1040) (ID(I,J),J=1,4),XLAMB(I),XIKAK(I),FRCS(I),TRCS(I),
     1ICODE(I)
      IF(ID(I,1)-IEOD) 74,77,74
   74 WRITE(NW,4000)(ID(I,J),J=1,4),XLAMB(I),XIKAK(I),FRCS(I),TRCS(I),
     1ICODE(I)
   76 CONTINUE
      WRITE(NW,2130)
      CALL EXIT
   77 NID=I-1
  900 READ(NR,1010) ITITL,NCASE
      IF(NCASE)999,999, 10
   10 READ(NR,1000)(XLVTM(I),I=1,4),(DCTIM(I),I=1,4)
      READ(NR,1000)(AVPOW(I),I=1,4),FFLUX,SFLUX,TLWT
      READ(NR,1030) (PERCT(I),I=1,NMATL)
      DO  80 I=1,NMATL
   80 PERCT(I)=PERCT(I)/100.
C.... AVAGADRO'S NUMBER
      AVGNO=0.6023E+24
C....
      CONS=1.0/(3.7E+10)
C.... CHANGE WEIGHT FROM LBS TO GRAMS
      ADJWT=453.59*TLWT
C.... CALCULATE TOTAL PERCENT WEIGHT FOR EACH ELEMENT IN EACH MATERIAL.
```

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                                80 COLUMN LISTING
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
.................................................................

      DO  90 I=1,NELEM
   90 ELTOT(I)=0.0
      DO 100 I=1,NELEM
      DO 100 J=1,NMATL
  100 ELTOT(I)=ELTOT(I)+ PERCT(J)*ELEM(I,J)
C     ----------------------------------------------------------------
C.... ACTIVITY CALCULATIONS
C     ----------------------------------------------------------------
      TACT=0.0
      DO 190 I=1,NID
      IF(ID(1,I)-IBLNK)110,175,110
  110 WRITE(NW,3010) (ID(I,J),J=1,4)
      IF(I-1)120,120,150
  120 WRITE(NW,2000) NCASE,IM,ID,IY,ITITL
      DO 140 J=1,NMATL
      IF(PERCT(J))140,140,130
  130 PRCT=PERCT(J)*100.0
      WRITE (NW,2110) TYPE(J),PRCT
  140 CONTINUE
      WRITE(NW,2700)
      WRITE(NW,3000)
      GO TO 170
  150 IF(STACT)160,190,160
  160 STACT=STACT/1.0E+24
      WRITE(NW,3020) STACT
      TACT=TACT+STACT
  170 STACT=0.0
  175 APFC=0.0
      DO 180 J=1,4
      EXP1= 1.0/EXP(XLAMB(I)*DCTIM(J))
  180 APFC=AVPOW(J)*(1.-1./EXP(XLAMB(I)*XLVTM(J)))*EXP1 + APFC
      K=ICODE(I)
      STACT=STACT+CONS*AVGNO*ADJWT*ELTOT(K)*XIKAK(I)*FRCS(I)*FFLUX*APFC
      STACT=STACT+CONS*AVGNO*ADJWT*ELTOT(K)*XIKAK(I)*TRCS(I)*SFLUX*APFC
  190 CONTINUE
      STACT=STACT/1.0E+24
      WRITE(NW,3020) STACT
      TACT=TACT+STACT
      WRITE(NW,2670) TACT
      GO TO 900
  999 CALL EXIT
 1000 FORMAT(7(E8.0,1X),E8.0)
 1010 FORMAT(35A2,5X,I5)
 1020 FORMAT(10E8.0)
 1030 FORMAT(8(F6.0,4X))
 1040 FORMAT(4A2,F7.0,3F10.0)
 2000 FORMAT('1',T9,'BLACK AND VEATCH CONSULTING ENGINEERS',T74,'CASE ',
     1I3,/,T9,'ACTIVITY LEVEL CALCULATIONS',T74,A2,'/',A2,'/',A2,//
     2T9,35A2,/)
 2110 FORMAT(T12,A6,T25,F7.3)
```

```
                                 80 COLUMN LISTING
0    0    1    1    2    2    3    3    4    4    5    5    6    6    7    7
1    5    0    5    0    5    0    5    0    5    0    5    0    5    0    5
********************************************************************************

  2120 FORMAT(' ',8E11.4/)
  2130 FORMAT(' ','MORE THAN 200 NUCLIDE CARDS')
  2670 FORMAT(///,T9,'TOTAL ACTIVITY',T67,E15.4)
  2700 FORMAT('0',T9,'CALCULATED PARAMETERS',/,T9,21('-'))
  3000 FORMAT(1H0,T9,'ELEMENT',T24,'LEVEL'/)
  3010 FORMAT(' ',T9,4A2)
  3020 FORMAT('+',T21,E11.4)
  4000 FORMAT(' ',4A2,4(2X,E11.4),2X,I3)
       END
// DUP
*DELETE             N29BV
*STORE       WS  UA  N29BV
// JOB
// XEQ N29BV


 19.0E-2 19.0E-2 18.0E-2 17.0E-2 17.0E-2 12.5E-2 13.0E-2                 21.0E-
  2.5E-2  0.7E-2 16.0E-2                  6.1E-5
 10.0E-2 10.0E-2  9.0E-2 13.0E-2 12.0E-2                                 10.0E-
          1.0E-2 72.0E-2
  0.8E-3  0.3E-3  1.5E-3  0.1E-2  0.3E-3  0.8E-3  1.5E-3  3.3E-3  3.5E-3  0.5E-
  1.5E-3  0.2E-2  1.5E-3
  2.0E-2  2.0E-2  2.0E-2  2.0E-2  2.0E-2  1.0E-2 12.5E-3  1.0E-2  0.9E-2  2.5E-
  0.5E-2  1.0E-2  1.0E-2                  0.8E-3
  0.3E-3  0.3E-3  0.7E-3  0.3E-3  0.3E-3  0.3E-3  0.3E-3  0.4E-3  0.5E-3  0.3E-
  0.3E-3  0.5E-3  1.5E-4
  4.5E-4  4.5E-4  0.7E-3  4.5E-4  4.5E-4  0.4E-3  0.7E-3  3.5E-4  0.5E-3  0.3E-
  0.3E-3  0.4E-4                          1.1E-3
                  0.6E-2                          0.6E-2

  0.2E-3
                                          0.3E-3


                          4.4E-2         38.5E-3

                          1.3E-2         17.4E-3
  0.2E-2  0.2E-2  0.2E-2                          0.1E-3
          0.2E-2                          1.5E-5

                          1.6E-2          2.5E-2
 35.9E-4
          3.5E-3  0.5E-2

                                          4.8E-8

                         52.9E-2

                          1.0E-2 11.1E-2
```

80 COLUMN LISTING

```
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
.................................................................................

          0.4E-4                          8.5E-6

                                          0.3E-4

                                          2.9E-4

                                          1.5E-5

                                          0.1E-6

                                          0.5E-9

                                          3.7E-6

                                          1.1E-6

                                          4.3E-6

                                          2.6E-6

                                          0.4E-2
```

SM-153	129.2	.001777	.4	210.	25
PM-149	114.6	.000397	.045	4.	24
CD-115	110.0	.00257		.3	23
SC-47	74.4	.001520	.064		29
TB-161	36.7	.001393		.768	27
SN-125	26.9	.0005005		.004	9
SR-91	26.0	.001907	.000014		8
ND-147	22.8	.001194		2.	24
P -32	17.69	.0296	.225		6
	17.69	.0323	.007	190.	7
	17.69	.0213	.1	.00008	21
P -33	10.12	.000237	.065	.002	6
	10.12	.001316	.000002		6
CR-51	9.10	.000829		17.	2
	9.10	.001042	.00074		1
	9.10	.01611	.00003		2
AR-37	7.21	.0241	.0067	.0025	11
	7.21	.0238	.002		12
CD-115M	5.89	.00257		.36	23
	5.89	.00202	.000002		9
FE-59	5.62	.0000591		1.2	1
	5.62	.01697	.0017		13
	5.62	.000623	.00018		3
HG-203	5.43	.001139	.000001		30
IN-114M	5.06	.0000556	.0003		9
	5.06	.0000295	.00003		9

80 COLUMN LISTING

```
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7   8
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5   0
.................................................................................
```

SR-89	4.97	.001876	.000074		8
Y -91	4.29	.001231	.00028		8
-	4.29	.001907	.000014		8
ZR-95	3.89	.001907		.076	8
	3.89	.000307	.004		8
CO-58	3.56	.01156	.073		3
	3.56	.01697	.0002		13
SC-46	3.04	.0222	.010	25.	22
	3.04	.001656	.300		29
S -35	2.92	.0213		.4	21
	2.92	.001316		.27	6
	2.92	.00000437	.00017		6
SN-113	2.14	.0000809		1.3	9
	2.14	.0000556	.0003		9
SN-123M	1.961	.000500	.0024		9
	1.96	.000398		.001	9
DY-159	1.757	.00000554		96.	28
CA-45	1.552	.000514		.67	11
	1.552	.000000823	.00028		11
GD-153	1.054	.00001272		125.	27
	1.054	.00000320		.0009	28
ZN-65	1.041	.00748		.407	10
SN-119M	1.012	.00202		.010	9
	1.012	.00277	.001		9
AG-110M	.973	.00447		3.5	16
CO-57	.930	.01156	.073		3
	.930	.01156	.00003		3
MN-54	.811	.001041	.051		1
	.811	.01820	.0002		5
V -49	.767	.000829	.0144		2
	.767	.000829	.000007		2
SM-145	.744	.000206	.119	.7	25
CD-109	.558	.0000783		3.	23
	.558	.00214	.000065		9
EU-155	.385	.001510	.1	5.5	25
FE-55	.289	.001041		2.8	1
	.289	.01156	.0034		3
	.289	.01641	.00007		1
SB-125	.256	.000500		.144	9
NA-22	.244	.0435	.000007		14
TI-204	.1824	.0000714	.0003		30
	.1824	.001139	.000001		30
CO-60	.1323	.01697		37.2	13
	.1323	.00447	.002		3
	.1323	.01087	.000760		15
	.1323	.000203	.0007		3
H -3	.0566	.01069	950.		20
	.0566	.01069	3.	476.	20
	.0566	.1334	.34		20
	.0566	.0742	.015		19

```
                                  80 COLUMN LISTING
0    0    1    1    2    2    3    3    4    4    5    5    6    6    7    7
1    5    0    5    0    5    0    5    0    5    0    5    0    5    0    5
................................................................................

        .0566  .0001488 .000029   .0005          18
        .0566  .0238    .0002                    12
        .0566  .001134  .000001                  30
EU-152  .0533  .00315             5930.          26
CD-113M .0495  .001205  .000002                   9
EU-154  .0433  .00343   1.        450.           26
        .0433  .00214             .06            23
PM-145  .0385  .000206  .119      .7             25
SN-121M .00915 .000398  .0019                     9
        .00915 .00277             .001            9
SM-151  .00769 .0001367           .0006          27
        .00769 .000495  .37       102.           25
        .00769 .000390  .06       1.8            24
NI-63   .00756 .000623            15.             3
        .00756 .01087   .009                     15
        .00756 .0001840 .00045                    3
AG-108M .0691  .00480             1.24           16
        .0691  .00447   .0011                    16
AR-39   .00257 .0238    .024                     12
        .00257 .0001597 .0024                    11
C -14   .000121.000924  .0001     .0009           4
        .000121.0000231           .235           17
..
PATHFINDER DECOMMISSIONING    ACTIVITY CHECK
   .042      .1033     .1692    .2115     2.42      2.08      1.67      1.25
   15.0      50.9       93.8    112.4   1.24E+11  1.00E+11  4800.
       100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK BOILER SHROUD
   .042      .1033     .1692    .2115     2.42      2.08      1.67      1.25
   15.0      50.9       93.8    112.4   0.68E+11  1.36E+11  1440.
       100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK STEAM SEPARATORS
   .042      .1033     .1692    .2115     2.42      2.08      1.67      1.25
   15.0      50.9       93.8    112.4   2.55E+08  2.87E+08   452.
       100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK BOILER CONTROL RODS
   .042      .1033     .1692    .2115     2.42      2.08      1.67      1.25
   15.0      50.9       93.8    112.4   2.87E+11  2.15E+11   169.
       100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK HOLDDOWN
   .042      .1033     .1692    .2115     2.42      2.08      1.67      1.25
   15.0      50.9       93.8    112.4   5.95E+10  1.04E+11  7000.
       100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK GRID PLATE
   .042      .1033     .1692    .2115     2.42      2.08      1.67      1.25
```

```
                              80 COLUMN LISTING
0   0   1   1   2   2   3   3   4   4   5   5   6   6   7   7
1   5   0   5   0   5   0   5   0   5   0   5   0   5   0   5
................................................................................

   15.0     50.9       93.8     112.4  5.95E+10 1.04E+11 5600.
                   100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK SEPERATOR SUPPORT SHELF
   .042     .1033     .1692     .2115     2.42      2.08       1.67     1.25
   15.0     50.9       93.8     112.4  8.50E+07 0.96E+08  770.
                   100.

PATHFINDER DECOMMISSIONING    ACTIVITY CHECK VESSEL WALLS
   .042     .1033     .1692     .2115     2.42      2.08       1.67     1.25
   15.0     50.9       93.8     112.4  1.12E+06 4.79E+06 146425.

          100.
PATHFINDER DECOMMISSIONING    ACTIVITY CHECK CONCRETE
   .042     .1033     .1692     .2115     2.42      2.08       1.67     1.25
   15.0     50.9       93.8     112.4  0.43E+05 0.43E+06 200000.

                                                              100.

 @FIN
```

APPENDIX C - CRITERIA FOR SPECIAL FORM RADIOACTIVE MATERIAL

To be considered special form radioactive material, the following requirements must be met:

1. The radioactive material must either be in massive solid form or encapsulated.

2. Each item in massive solid form or each capsule must have no overall dimension less than 0.5 mm, or must have at least one dimension greater than 5 mm.

3. Each item, or the capsule material, must not dissolve or convert into dispersible form to the extent of more than 0.005 percent, by weight, by immersion for one week in water at pH 6-8 and 68°F, and a maximum conductivity of 10 micro-mhos/cm., and by immersion in air at 86°F.

4. If in massive solid form, the radioactive material must not break, crumble or shatter if subjected to the percussion test described below, and must not melt, sublime or ignite at temperatures below 1,000°F.

   a. Free drop - A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in such a position as to suffer maximum damage.

   b. Percussion - Impact of the flat circular end of a one-inch diameter steel rod weighing three pounds, dropped through a distance of 40-inches. The capsule or material shall be placed on a sheet of lead, of hardness number 3.5 to 4.5 on the Vicker's scale, and not more than one-inch thick, supported by a smooth, essentially unyielding surface.

   c. Heating - Heating in air to a temperature of 1,475°F and remaining at that temperature for a period of ten minutes.

   d. Immersion - Immersion for twenty-four hours in water at room temperature. The water shall be at pH 6-8 with a maximum conductivity of 10 micro-mhos/cm.

APPENDIX D - TYPICAL SAFE WORK PROCEDURES
SWP. P-1. PIPE AND METALS CUTTING

Introduction

All existing piping and metals in this plant are considered to be potentially contaminated internally and/or externally. This contamination must not be spread in an uncontrolled manner.

This specification describes the methods and protective measures to be employed during the cutting of pipe and other metals which are to be altered during dismantling of this plant. This specification supplements the requirements set forth in the Radiation Safety Safe Work Procedure. The preparation of new materials is exempted from the requirements of this specification.

This specification may be revised to suit special situations with the prior approval of the Project Manager. The Contract Supervisor will obtain his and the safety committee's concurrence before authorizing any changes. Each approved change will be entered onto the master copy of this specification with specific description of the situations where the change is applicable, and the change will be signed and dated by the Project Manager.

Prerequisites

1. A work permit has been prepared and approved.
2. The work area has been surveyed and defined by Radiation Monitoring in accordance with the Radiation Safety SWP and a radiation work procedure (RWP) has been posted at the entrance to the work area.
3. The work item of interest has been identified with a white tag bearing the serial number of the work permit and the Contract Supervisor's signature.
4. The Contract Supervisor with the plant staff to insure that any holds required on operating equipment are being maintained.

Equipment Requirements, Furnished by the Utility Plant Staff

1. Personal monitoring devices as specified on the RWP.

2. Continuous air monitor (CAM) with sniffer hose.

3. Other air and area monitoring equipment as specified on the RWP.

Equipment Requirements, Furnished by Contractor (or Provided by the Utility's Dismantling Crew

The contractor will provide all other tools, equipment and materials, including:

1. Respirators, goggles, anti-C clothing, etc. as shown on RWP.

2. Industrial-type vacuum cleaner with filter and hose.

3. Waste catch and storage materials (polyethylene sheets--6 mils minimum thickness, drums, duct tape, blotter paper, hand rags, masking tape, etc.)

4. Cutters. Roller cutters, chain-type cutters, hack saw, sabre saws, tin snips, powered hand-held saws and guillotine saws, according to the need. Flame and arc cutting devices and abrasive cutting wheels are not acceptable since they promote the spread of contamination.

5. Scaffolding, chain falls, etc.

Work Method

1. Insure that the foregoing prerequisites have been met and that a Radiation Monitoring Technician is present at the start of the work. The degree of his continued presence will depend on the particular case. This need will be recorded on the RWP.

2. Install scaffolding and chain falls according to the need. Install restraints to prevent significant movement when the cut is made.

3. For pipe cutting, check adjacent valves to insure the line is properly isolated, drained, and vented. If draining cannot be clearly established, a hole will be drilled at the low point in the pipe immediately prior to the start of cutting operations. When only a small amount of drainage is anticipated the hole may be made with only a handheld drill motor and a bucket to catch the drainage. When substantially large quantities are anticipated, drilling and draining shall be accomplished in a manner similar to that shown on the attachment to the SWP. Rubber gloves and a face shield are the minimum wet protection requirements when drilling pipe. Radiation Monitoring will specify other items as needed.

4. Install a catch basin of polyethylene sheet or other suitable device under the work piece to catch falling insulation, pipe cuttings, libricants, etc.

5. Install a hood-type airborne collection system, with vacuum cleaner and filter, directly over the work piece. Turn on vacuum cleaner.

6. Locate CAM sniffer hose inlet to monitor the cut. Turn on CAM. If the CAM reaches the alarm point at any time during the work, all work is to be stopped, the work piece openings are to covered, the area is to be evacuated and the health physics staff is to be notified. The HP staff will direct the remedial effort and advise when normal work can be resumed.

7. Remove insulation as necessary to the work and deposit in storage drum.

8. Wipe the work piece thoroughly with a damp rag and deposit rag in radioactive material disposal drum.

9. Smear the work piece to a point one foot either side of the planned cut. Decontaminate same as needed by scrubbing or as otherwise instructed by Radiation Monitoring Technician. Repeat this step until smear counts are acceptable to Radiation Monitoring (HP staff member).

10. Make the cut with the chosen tool.

11. Cap the open ends of piping and equipment cavities with two separate applications of polyethylene sheet and duct tape, immediately after completion of the cut. If the space between the cut ends is not sufficient for capping, apply a double layer of polyethylene in a sleevetype wrapping.

12. Make required additional cuts in accordance with all of the foregoing.

13. When the pipe or other device is to be removed, all open ends shall be capped as soon as possible during the removal in accordance with item 11 above.

14. The removed pieces will be surveyed by Radiation Monitoring who will specify any additional protection required prior to removal to the place of disposition.

15. Wrap all tools, power equipment, chain fall, etc, in polyethylene sheet or bags and leave inside the work area until Radiation Monitoring has surveyed and affixed a noncontaminated clearance tag to same. If the tools and equipment are contaminated, they will be decontaminated by the Contractor's personnel and cleared for removal by Radiation Monitoring before they can be taken from the work area.

16. Uncleared tools and equipment used in the dismantling may be transferred from one work area to another when they are contained within a polyethylene bag which has been cleared by Radiation Monitoring.

17. Clean and decontaminate the work area as specified by Radiation Monitoring.

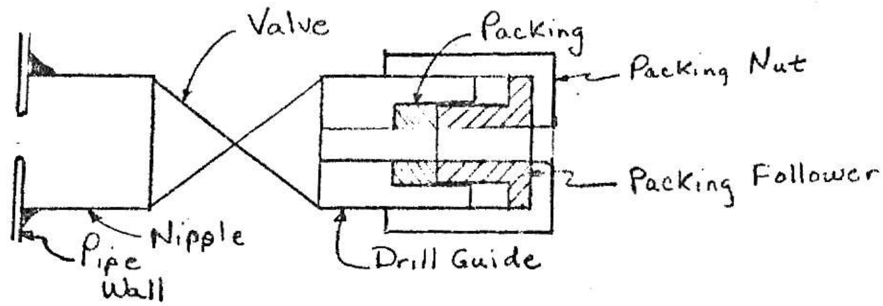
18. Advise the contract supervisor when the work is completed and return work permit and white tag.

Approved by Project Manager...

Approved for Safety Committee...

ATTACHMENT TO S.W.P. No. P-1

To obtain and demonstrate draining when a low point is not available.



1. Prepare drill guide.

2. Prepare a 3/4" x ½" pipe nipple, T.O.E., add pipe cap to project threads and shape opposite end to fit O.D. of pipe to be drained.

3. Clean work area on pipe to be drained and obtain Radiation Monitoring approval for welding.

4. Weld nipple to pipe using electrode compatible to pipe, remove cap and install gate valve.

5. Screw drill guide into valve (to prevent valve seat damage).

6. Open valve wide, insert special short flute long shank 3/8" drill bit through guide and hand-tighten packing nut.

7. Place catch basin under work.

8. Vent system to be drained.

9. Put on face shield and rubber gloves.

10. Attach drill motor to bit, assume a position so packing gland leakage cannot enter face shield and drill hole.

11. Withdraw drill sufficient to permit valve closure but still retain packing seal.

12. Close valve, remove drill and guide and dispose of both per Radiation Monitoring instructions.

13. Attach hose or bucket to valve and drain system.

14. After draining is completed, Contract Supervisor representative will confirm draining, tag nipple and note the location.

15. When nipple is tagged, remove tag and valve and cap nipple.

16. Dispose of valve, bucket, hose and tools per Radiation Monitoring instruction.

17. Deliver tag to Contract Supervisor.

APPENDIX E - DETERMINATION OF HYPOTHETICAL PLANT ACTIVITY INVENTORY

The hypothetical plant inventory shown on Table I was derived in the following manner. The reactor inventory was calculated using the load model of Figure 6. A 1000 MW(e) reactor vessel and internals was approximated using information in the Westinghouse and CE standard plant reference PSAR's (46,47). Table E-1 provides the base data used for the calculation. The activities are based on one year cooldown time to allow decay of fuel activities prior to shipping fuel off-site. The primary coolant loop and spent fuel storage pool activities were calculated based on the total corrosion product equilibrium crud film thickness (47). Table E-2 provides the data used in the calculations. Estimation of the concrete activity is from information contained in the ERR dismantling documents (12,13). Since the concrete composition is subject to variations due to composition of the aggregate and cement, this figure is a rough approximation only.

The inventory for the rest of the plant systems is based on information contained in the CE standard plant PSAR (47). Using maximum activities for shielding, the inventory was based on Co-60, Fe-59, Co-58, Mn-54, Cr-51, and ZR-25 activities. No decay was assumed nor was compensation for fission product inventory included.

TABLE E-1 - DATA FOR CORE ACTIVITY CHECK

	Weight (lbs.)	Material	S-Flux	F-Flux
Lower Support Plate	7,500	304ss	1.46(13)*	9.8 (13)*
Upper Support Plate	6,040	304ss	1.63(13)	6.05(13)
Core Shroud	18,759	304ss	8.60(12)	1.36(14)
Core Barrel	44,349	304ss	8.4 (11)	8.2 (12)
Thermal Shield	58,175	304ss	3.2 (11)	3.6 (12)
Reactor Vessel	213,470	A-533-B	3.38(10)	1.46(11)

*( ) denote powers of ten

TABLE E-2 - DATA FOR PRIMARY LOOP AND SPENT FUEL POOL ACTIVITY CHECK

Primary Loop Component	Approximate Area ft^2	Crud Film Thickness (47) (mg/cm^2)	Decay Time Year	DF
Steam Generators	3,080	1.0	1	10
Steam Generator Tubes	200,000	0.1	1	10
Pressurizer	1,053	0.4	1	10
Reactor	9,960	1.0	1	10
Pumps and Piping	1,796	1.0	1	10
Fuel Storage Pool	16,000	0.001	1	20

APPENDIX F - COST ESTIMATE OF DISMANTLING OPTIONS

The cost estimate contained in Table VI was derived from the data contained in WASH-1230 (Vol. 1) (45). A breakdown accounting follows the account outline of NUS-531 "Guide for Economic Evaluation of Nuclear Reactor Plant Designs," (48). The relative large cost of the internment option is due to complete removal of all piping and electrical equipment in the turbine plant. The material and equipment cost item of Table VI includes shipment and burial costs. While the engineering costs for the storage option is almost 40% of the Construction cost, the engineering costs drops to 5% for the removal option. Health physics costs are contained in the overhead cost item. No cost for decontamination of the radioactive systems in the plant prior to commencement of the dismantling activities are included. It is assumed that the system decontamination and fuel removal costs will be handled by the plant operations budget.

TABLE F-1 - BREAKDOWN OF LABOR COST ITEM IN DISMANTLING COST OPTIONS

Account No.	Description	Storage	Costs ($1000) Internment	Removal
21.	STRUCTURES AND IMPROVEMENTS			
211.	Yard Work	--	40	70
212.	Containment	57	853	1493
213.	Turbine and Heater Bldg.	14	24	383
214.	Intake and Discharge	--	32	288
215.	Primary Aux. Bldg.	72	196	556
217.	Fuel Handling Bldg.	16	68	214
218.	Remaining Bldgs.	10	30	244
22.	REACTOR PLANT			
221.	Reactor Equipment	24	51	302
222.	Reactor Coolant System	20	414	650
223.	Safeguard Cooling System	10	145	285
224.	Rad Waste and Disposal	17	60	135
225.	Fuel Handling and Storage	17	78	144
226.	Reactor Aux. Equipment	85	362	485
227.	Instruments and Controls	20	80	170
23.	TURBINE PLANT			
231.	Turbine Generator Equipment	10	117	379
232.	Circulating Water System	--	70	502
233.	Condensing Systems	--	30	280
234.	Feedwater System	--	395	435
235.	Other Plant Equipment	--	426	527
236.	Instruments and Controls	--	60	85
24.	ELECTRIC PLANT EQUIPMENT	20	450	650
25.	MISC. PLANT EQUIPMENT	10	40	213
	TOTAL	402	4021	8490

*Based on Cost of demolition at 10% of construction cost. (1971 costs)

APPENDIX G - COMPARISON OF CALCULATED AND MEASURED DOSE RATES RESULTING FROM ACTIVATION OF THE ERR REACTOR

The ERR induced activities were calculated by the method contained in Appendix A. Table G-1 shows an example of how the inner thermal shield was divided into four distinct sections. Table G-2 shows a comparison between the resulting calculated dose rates and the actual measured gamma dose rates from components of the ERR. The only large variation is found on the measurement of dose rates from the concrete sample located on the inner edge of the biological shield.

TABLE G-1 - TYPICAL COMPONENT DATA FOR CALCULATING ACTIVITY FOR THE ERR

Component	Weight, g	Average Flux, n/cm^2-sec-MW Fast	Thermal	Element	w/o
Inner thermal shield-ASTM-A240-63				C	0.08
				Cr	20.
+30. to -34.25	2.18(+6)	4.09(+9)	2.62(+9)	Mn	2.
+30. to +85	1.86(+6)	3.46(+8)	2.21(+8)	Fe	64.645
-34.25 to -59.	8.29(+5)	5.10(+8)	3.26(+8)	Co	0.2
-59 to -70.25	3.15(+5)	9.54(+7)	6.11(+7)	Ni	12.
				P*	0.045
				S*	0.030
				Si*	1.0

*These elements produced no radionuclides with half-lives in the range of interest.

TABLE G-2 - COMPARISON OF CALCULATED AND MEASURED GAMMA RAY DOSE RATES

Dose Point	Components Included	Dose Rate (6/2/71) Calculated R/hr	Measured R/hr
Outside Pressure Vessel (913 ft-4 in. elev.)	Inner Th. Shield, Pressure Vessel, Insulation, Outer Th. Shield	33.68	33.5
Inside Inner Th. Shield (913 ft-4 in. elev.)	Inner Th. Shield, Pressure Vessel	940.7	828.
Core & Shroud Plates	Core and Shroud Plates	8115.	6000.
Upper Baffle Plate (or Shadow Shield) (916 ft elev.)	Upper Baffle Plate	3050.	2400.
Instrument Tube (Upper)	Inner Th. Shield, Pressure Vessel, Insulation, Outer Th. Shield, Instrument Tube	24.93* 14.19**	24.
Instrument Tube (lower)	Core & Shroud Plates, Inner Th. Shield, Pressure Vessel, Insulations, Outer Th. Shield, Instrument Tube	36.11* 25.37**	30.
Core Sample Hole 9 ft from outer edge (assume inner edge)	Concrete with Contribution from Pressure Vessel of .27 R/hr	.41	3.15

*Tube calculated as 4 infinite slabs.
**Tube calculated as thin cylindrical shell.

ECONOMIC CONSIDERATIONS IN THE DECOMMISSIONING OF LIGHT WATER REACTORS

by

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ABSTRACT

The decommissioning or dismantling of a nuclear power generating station entails a considerable amount of work. An initial review of the demonstration power reactors decommissioned to date reveals that the actual dismantling may vary from storage of radioactive material and components in the plant buildings to complete removal of all radioactive material from a site. The expense of complying with AEC regulations has been and will continue to be the primary cause for most reactor decommissionings. A general review of the AEC regulations which apply specifically to dismantling reveal that three major options and one minor one are available for dismantling a reactor.

The range of acceptable dismantlings include minimum on-site "Storage" or mothballing, "Internment," or complete "Removal." Modification of the turbine for accepting steam from another source is also mentioned as an option by the AEC. A general review of the problems encountered in dismantling work discloses that plant radiation levels, contamination control, and waste disposal problems result in complexity and increase the cost of the dismantling activities. The documents that must be submitted to the AEC include a decommissioning plan, a safety analysis, an environmental report, and a final status report. The factors which affect the cost effectiveness of dismantling work are safe work procedures, material control, waste disposal, and health physics procedures and large labor costs result due to work in a radiation field.

The future requirements for dismantling work will be minimal for the immediate future except for decommissioning of experimental reactors.

Appreciable work on decommissioning of present nuclear power plants should commence in 2015 with increased dismantling requirements from then on. In order to economically handle the dismantling and to minimize the radiation exposure to the occupational worker, the AEC, along with assistance from the licensees, should generate minimum requirements for a "pre-dismantling" interim license which will allow the licensee to place the plant in hold from when operation has ceased until the short-lived radionuclides decay. This will minimize the cost to the licensee and radiation exposure to the worker without being detrimental to the public health and safety.