CONSOLIDATED EDISON CO. OF NEW YORK

PWR DECONTAMINATION FEASIBILITY STUDY

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Prepared by

Paul L. Silliman
Paul L. Silliman
Sr. Engineer

Approved by

William P. Bennett
William P. Bennett
Program Manager

Date

December 18, 1978

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

PWR DECONTAMINATION STUDY
DOE CONTRACT NO. EY-76-C-02-4008

TABLE OF CONTENTS

- 1.0 SUMMARY AND CONCLUSIONS
- 2.0 RADIATION PWR SYSTEMS
- 3.0 DECONTAMINATION EXPERIENCE
- 4.0 DECONTAMINATION METHODS SELECTION
- 5.0 CORROSION TESTING PROGRAM
- 6.0 DECONTAMINATION SYSTEM DESCRIPTION
- 7.0 DECONTAMINATION PROCEDURES
- 8.0 COSTS
- 9.0 PHASE 2-5 REQUIREMENTS
- 10.0 REFERENCES

APPENDICES

- A DECONTAMINATION SUMMARIES
- B IPL DESCRIPTION AND MATERIALS
- C IPL ESEERCO/DOW PHASE 1 TEST PROGRAM
AND SUMMARY OF RESULTS
- D DEMONSTRATION COST DATA

1.0 SUMMARY AND CONCLUSIONS

1.0 SUMMARY AND CONCLUSIONS

The purpose of improved decontamination techniques is to aid our nation's and our utility industry's efforts toward energy independence by reducing radiation levels in nuclear plants. Worker exposure to radiation continues to be of prime concern to the industry and to the Federal Government. Reducing radiation levels in areas of the steam generator would result in less exposure especially since it appears that there will be a greater amount of maintenance and repair required in that area. Planning work procedures to attain minimum worker exposure in high radiation areas, use of protective clothing and temporary shielding all would be much more effective if the source of the higher field were reduced.

While past efforts in plant decontamination have been directed to obtaining a maximum reduction in fields, it is the objective of this program to recommend methods which will minimize exposure to personnel and which will be less complex, generating less waste and therefore be more acceptable to the industry. This might be accomplished by reducing decontamination factors but increasing overall effectiveness by allowing more frequent decontamination. This approach would be implemented at reduced costs and lower radiation exposures from plant modifications and waste processing from that required by the more complex methods.

As part of the Department of Energy radiation control program, the Consolidated Edison Company of New York was contracted to perform the first part of a multi-phase decontamination program, intended to culminate in an in-plant demonstration. This report gives the results of a scoping study having the following objectives:

- o To select several processes that will likely be of use decontaminating primary systems of pressurized water reactors (PWR).
- o To make the conceptual process design and cost estimate for conducting a demonstration of each process on a PWR.
- o To prescribe the development and laboratory testing program required before a demonstration can be accomplished.

This report reviews the decontamination work which has been accomplished and concludes that it is worthwhile to investigate further four methods for decontamination for future demonstration. These are:

- Dilute chemical
- Single stage strong chemical
- Redox processes
- Redox/chemical in combination

Laboratory work is recommended to define the agents and processes for demonstration and to determine the effect of the solvents on PWR materials.

The feasibility of Indian Point 1 for decontamination demonstrations is discussed, and it is shown that the system components of Indian Point 1 are well suited for use in demonstrations.

A conceptual design of the decontamination demonstrations system is presented, and conceptual procedures based on this design are developed for the above listed methods.

An engineering cost estimate is presented for the laboratory testing phase, detailed design/cost estimate phase, the construction of the demonstration facility, and the cost of conducting each of the demonstrations. The schedule for the multi-phase decontamination program is also included.

A cost/benefit analysis is developed based on a 900 MWe unit, where cost are balanced against assigned values for radiation reduction.

In summary the report concludes:

- o There are four general processes that warrent consideration.
- o Indian Point 1 is suitable and economical for decontamination development.
- o Developments required to enable use of processes are laboratory testing of promising solvents, corrosion testing on PWR materials, and determination of decontamination waste treatment methods.
- o Decontaminations are likely to be economical to apply.
- o The estimated cost of worker exposure to radiation is \$5,300 per manrem.
- o Costs of developing processes and demonstrations on Indian Point 1 are \$2.2 million for testing, \$2.8 million for design and construction, and \$2.0 million for demonstrations. The program would cover a six year period.

2.0 RADIATION IN PWR SYSTEMS

2.0 RADIATION IN PWR SYSTEMS

Pressurized water reactor primary circuits are considered relatively free of corrosion products in comparison to most industrial processes. This clean condition obtains because components and piping, in contact with reactor cooling water, are constructed from or clad with corrosion resistant materials. The circulating fluid is of high purity and undergoes purification during service. Nevertheless there is a degree of corrosion taking place, starting with the protective oxide coating normally generated on the metal surfaces. Corrosion also results from unwanted agents present in the primary fluid. Unfortunately, corrosion products residing in or originating in the reactor core become radioactive as the result of neutron bombardment.

The major elements comprising the wetted surfaces of the primary circuit are nickel, chromium, zirconium, and iron. There are smaller amounts of cobalt and manganese. During the operation of modern pressurized water reactor plants, contamination occurs in the form of deposited activated corrosion products. Principal active isotopes are cobalt 58 and 60, iron 59, manganese 54, chromium 51, and zirconium 95.

The corrosion products containing the activated components are mainly compounds of nickel, iron and chromium. The major source of radiation is cobalt 58 in the first years of PWR plant life and later cobalt 60. This results from activation of cobalt and nickel erosion/corrosion product and subsequent deposition on the surfaces of the primary system components. Cobalt is present as

impurity in stainless steel and a major component of Satellite used for valve facing. Nickel is a major component in stainless steel and inconel.

Contamination of primary circuitry increases with service time. Maintenance requirements also increase with the age of the plant, and becomes more difficult due to the higher radiation levels because the allowable worker exposure time becomes shorter in order to avoid individual over-exposure to workers. This typically necessitates using a number of workers on a task which in the absence of radiation might take an individual worker to accomplish. Table 2-1 gives the exposure limit for the Code of Federal Regulation (CFR) Part 20, and for a typical operating utility. Figure 2-1 graphically equates exposure time limits to radiation build up in typical PWR's. It can be seen that maximum exposure time limits become very restrictive as radiation fields increase. Radiation levels in the channel heads of PWR steam generators are typically in the range of 5R/hr to 20R/hr. Access or exposure time is therefore only 36 minutes to less than 12 minutes for workers per quarter. Therefore as radiation build up occurs it is desirable to reduce it at intervals during operation or before an extensive maintenance or inspection is started.

Figures 2-2 and 2-3 show activities on primary system piping and steam generator tube sheets for various reactors.⁽¹⁾ Dose rates on tube sheets can reach very high levels within two effective full power years. Referring to Figure 2-1, it is noted that the average of the radiation levels on tube sheets (about 18R/hr) could

allow worker access for only about 10 minutes per quarter. Concentration of activity at the steam generator tube sheet is much in excess of readings on the primary piping. This can be expected since the steam generator offers more surface for radioactive deposit.

Table 2-2 gives radiation levels at various locations for several reactors at periods in their operating history. Table 2-3 gives radiation level measurements in the primary system at Indian Point 1 for various times; radiation levels determined in August 1976 at various locations in the primary and purification loops are shown in Table 2-4. These are external contact readings and serve to compare the activities at various locations throughout the plant.

Consideration of methods and means for conducting decontamination will focus on the steam generator surfaces of the primary circuit including the water boxes and tubes. In addition to this area's being the source of the highest radioactivity levels during off-line periods, it is likely to be the location for maximum over-exposure.

RATE OF CONTAMINATION BUILDUP R/H
VS.
EFFECTIVE FULL POWER DAYS (EFPD) AND
LIMITS OF EXPOSURE TIME AT VARIOUS
CONTAMINATION LEVELS

PWR PRIMARY PIPING

Figure 2-2

Shutdown Radiation Levels for PWR's
Primary Piping
Source - EPRI

11

11

PWR TUBE SHEETS

Figure 2-3

Shutdown Radiation Levels for PWR's
Tube Sheets
Source - EPRI

TABLE 2-1

WORKER RADIATION EXPOSURE TIME LIMITS

Radiation R/hr	Exposure Time Limits			
	10CFR Part 20		Typical Utility Policy	
	<u>Hrs/quarter</u>	<u>Hrs/yr</u>	<u>Hrs/quarter</u>	<u>Hrs/yr</u>
0.1	30	50	12.5	50
0.5	6	10	2.5	10
1.0	3	5	1.25	5
5.0	0.6	1	0.25	1
10.0	0.3	0.5	0.125	0.5
15.0	0.2	0.33	0.083	0.33

TABLE 2-2

Sources - EPRI report 404-2, and references (2) (3) (6)Heat exchanger tubes sheets or tubes

<u>Plant</u>	<u>EPPY</u>	<u>mR/h</u>	<u>Location</u>
Indian Point 1	5.4	18000	No. 12 Boiler Outlet
	5.4	19100	13 " "
	5.5	7500	14 " "
Robinson 2	1.8	15000	Tube Sheet
Ginna	0.63	6000	Cold leg (bottom)
	1.35	2000	"
PRTR	1.4	4500	Tubes (max. rdg.)
Turkey Pt.	0.02	15	Regen. ht. exchgr.
	0.57	1200	"
	1.2	2000	"
Sena	0.27	1100	Tube sheet (hot leg)
		850	Partition plate (hot leg)
Shippingport	3.2	232	Pipe walls
	3.2	301	Steam generator
<u>Other Areas</u>			
NPR	1	1100	Data not available
	2	2000	"
	2	2500	"
San Onofre 1	0.3	300	Let down regen. ht. exchgr.
	0.8	1000	"
	1.0	900	"
	1.5	1200	"
Maine Yankee	1.35	1000	Coolant pump suction
Conn Yankee	0.16	18	Cold leg isolation valve
	1.2	75	"
	2.3	175	"
	3.4	250	"
	4.1	175	"

TABLE 2-3

Radiation Level Measurements in the
Primary System at Indian Point 1
 (Contact)

<u>Location/Description</u>	<u>Date</u>	<u>EFPY</u>	<u>Days After Shutdown</u>	<u>Mr/h</u>
#12 Nuclear Boiler				
Inlet	5/22/70	3.8	1	33600
Outlet	5/22/70	3.8	1	16200
	11/6/74	5.4	7	18000
#13 Nuclear Boiler				
Inlet	4/13/73	4.8	104	30000
	7/24/73	4.8	205	25000
Outlet	6/3/71	4.0		15600
	4/13/73	4.8	104	25000
	11/8/74	5.4	9	19100
#14 Nuclear Boiler				
Inlet	11/11/67	1.6		18000
Outlet	11/11/67	1.6		9600
	12/3/74	5.5	34	7500
Outlet Tube Sheet	12/9/65	1.2		2140
	11/8/74	5.4	9	2400
#13 Reactor Outlet Valve				
Effluent 1 ft. from weld joint	1/6/74	4.9	373	1000
Bottom	1/8/74	4.9	375	300
Side	1/8/74	4.9	375	500-700
#12 Reactor Outlet Line				
Upstream valve	1/3/73	4.8	4	2000
2 ft. before weld joint				
#11 Reactor Inlet Valve				
Side	12/7/73	4.8	343	400
General Field	12/7/73	4.8	343	500
Downcomer Secondary Loop, at elbows				
#12	7/19/73	4.8	200	1000-1100
#13	7/19/73	4.8	200	13000-15000
#14	7/19/73	4.8	200	8000-11000
Reactor Top Flange	3/22/73	4.8	82	500-8000
Reactor Water Level	3/22/73	4.8	82	2000-4000

TABLE 2-4

Field Survey of Indian Point 1, Primary and Purification LoopsPerformed in August 1976*

(all readings in mR/h)

<u>Description</u>	<u>Loop No.</u>			
	<u>#11</u>	<u>#12</u>	<u>#13</u>	<u>#14</u>
Boiler Inlet	150	150	180	130
Boiler Outlet	75	75	80	60
Primary Coolant Pump	300	350	350	300
Primary Coolant Pump	300	300	400	10
Check Valve	110	100	200	130
Check Valve	110	100	200	130
Reactor Inlet Valve	30	30	60	180
Reactor Outlet Valve	65	50	160	75
Reactor Inlet	180	210	100	160
Reactor Outlet	220	180	200	150
Pressurizer	90			
Purification Inlet Filter #11	**			
Purification Inlet Filter #12	**			
Purification Inlet Cooler #11	60			
Purification Inlet Cooler #12	30			
Cation Exchange Bed	**			
Mixed Bed	**			
Anion Exchanged Bed	**			
Purification Outlet Filter #11	700			
Purification Outlet Filter #12	800			
Clean Water Storage Tank #11	1500			
Clean Water Storage Tank #12	800			
Clean Water Storage Tank #13	1000			
Seal Water & Primary Make-up Pump #11	400			
Seal Water & Primary Make-up Pump #12	500			
Seal Water & Primary Make-up Pump #13	500			
Purification Transfer Pump #11	400			
Purification Transfer Pump #12	200			
Purification Transfer Pump #13	400			
Blowdown Storage Tank #11	450			
Blowdown Storage Tank #12	400			
Blowdown Cooler #11	100			
Blowdown Cooler #12	120			

* - Unit out of service since 1974

** - Area not accessible

3.0 DECONTAMINATION EXPERIENCE

3.0 DECONTAMINATION EXPERIENCE

This section briefly reviews reported decontamination experience. The review forms the basis for selecting decontamination methods for further work which is described in the next section.

Much experimentation on decontamination was accomplished in the past. The results of the many tests narrowed the field to a relatively few chemical agents which have been used in commercial nuclear facilities. In the main, thorough decontaminations were attempted so the reagents selected were those whose use resulted in a high decontamination factor. Multiple stages were required to accomplish this. More recently decontaminations were performed in Canada using more dilute reagents achieving partial effectiveness but imposing less of a problem in waste treatment and disposal, and outage time. The reports of decontamination experience are examined with the view of striking a practical balance between effectiveness and complexity.

Discussion of chemical decontamination experience is divided into three categories - strong chemical, mild chemical, and "pure water". Mechanical methods are touched on for possible application in this program, in particular for steam generator channel heads in preparation for tube inspection and/or maintenance.

Strong Chemical for thorough Decontamination

A full scale primary loop decontamination of the Plutonium Recycle Test Reactor (PRTR) was performed in 1962.⁽²⁾ An eight stage method was used to reduce the fields in the 304 stainless steel components.

The effectiveness was only about 50% and some of the later stages were required to dissolve reprecipitated deposit. The major stages were alkaline permanganate followed by oxalic acid. Ammonia citrate - EDTA was used for dissolving oxalate precipitates.

A second decontamination performed in 1965 utilized the alkaline permanganate citrox method ⁽²⁾ and was much more successful with fewer stages than the 1962 effort. Effectiveness was about 95%. The cleaning consisted of an alkaline permanganate stage followed by an Ammonium citrate - oxalic acid stage.

In 1964 the Shippingport Plant was decontaminated using alkaline permanganate followed by diammonium citrate. ⁽³⁾ These solutions were circulated through the 304 stainless steel piping at a strength of about 1.3%, less than half the strength of the 1965 PRTR solutions. Decontamination was generally 95% effective, but there were local accumulations of undissolved material. This resulted in long flush periods to reduce the fields. Solvent and rinse drains were directed to ion exchangers for waste treatment. Ion exchange resin requirements were higher than anticipated. As a result this phase was time consuming and expensive.

THE NPR, or N-reactor, at Hanford, Washington was decontaminated in 1965 using a two stage process, an alkaline permanganate circulation period followed by 7.5% inhibited sulfamic acid. ⁽⁴⁾ The cleaning was only about 50% effective on 304 stainless steel surfaces, 90% on carbon steel. Corrosion was indicated on carbon steel and 400 series stainless parts.

In 1967, N-reactor steam generators containing stainless steel and Inconel were decontaminated with a 4 step alkaline permanganate - Citrox process. This was 97% effective. The 4 step method resulted in high volume of wastes, and results showed that more dilute solutions may have been just as effective.

Also in 1967, and again in 1968, carbon steel sections of the N-reactor were decontaminated.⁽⁵⁾ Phosphoric acid was used at concentrations of 10.5% and 8% for 16 and 22 minute injection. Cleaning was fairly effective but localized pitting of carbon steel at welds to stainless occurred.

The steam generators of the SENA pressurized water reactor was decontaminated in 1968 using essentially the alkaline permanganate, Citrox process.⁽⁶⁾ This decontamination was 97% effective but there were 4 stages which generated a high volume of radioactive wastes.

The SGHWR at Winfrith, England utilized the Citrox step only, omitting alkaline permanganate because of the presence of aluminum in the system.⁽⁷⁾ Although not as effective as AP-Citrox, the results were favorable, ranging from 50% to 95% effective.

In 1977 United Nuclear Industries performed beaker tests to determine a suitable solvent for Indian Point 1 secondary side chemical cleaning. Carbon steel spacer specimens from "N" reactor were used and results included determination of radiation field reduction efficiencies. The results are listed in Table 3-1, and indicated that there are possible alternatives to Citrox for single stage strong chemical method.

Mild Chemical

The Candecon process involves the use of a dilute mixture of organic acids⁽⁸⁾ added to the coolant. Oxide films are dissolved and exfoliated from the primary circuit surfaces. The chelated metals are removed in the resins of the purification system, and filters remove particulates. Finally the cleaning acids are removed by anion exchange resins.

The Candecon process was demonstrated successfully on NPD, which contains Inconel-600 tubing.⁽⁹⁾ Special experimental filters were used simultaneously. The DF* of the combined treatment was about 3.

This method although not yielding the dramatic decon factor that can be obtained with AP-Citrox (20-30) may result in equivalent performance by more frequent application.⁽¹⁰⁾ The benefits of a mild chemical method appear to warrant pursuing this type of decontamination for PWR's.

A full scale Candecon decontamination of Douglas Point was conducted in 1975.^{(8) (11)} The radiation field at feeder cabinets were reduced from 2-3 R/hr. to 0.3-0.5 R/hr. with an average DF of 6. This was accomplished in 72 hours. Surfaces treated were mainly carbon steel and Inconel.

The Gentilly 1 unit was decontaminated in 1974. The major material is carbon steel with some stainless. The average decon factor was about 4.5.

*DF = $\frac{\text{radiation field before decon}}{\text{radiation field after decon}}$, also called decon factor

The Candecon method for corrosion product removal and solvent regeneration is proprietary and few details are available. However, estimates of treatment costs sufficient for comparison purposes are made in a later section and are based on published ion exchange capacities and solvent effectiveness.

Pure Water*

A procedure which can be accomplished during relatively short outages utilizing existing components of the primary circuitry is a "redox" type involving the variation of hydrogen, oxygen, pH values and flow in the primary loop while high purification flow is maintained. Small individual decontamination factors, multiplied over several decon cycles can result in a significantly high overall decontamination factor. The procedure used for the Douglas Point redox decontamination is appropriate for further study and development. (12) Despite low decontamination ratios, this approach is worthwhile since more decontamination benefit may be achieved at lower waste disposal cost.

The Douglas Point work was done on a heavy water system containing Monel and carbon steel, but decon was confined to the Monel portion. The chemical composition of corrosion products contain large proportions of copper, which is not found in the stainless steel or Inconel material of PWR's. The radioactive products were similar to those of PWR's, and Douglas Point varied system conditions which can be varied on all PWR's: temperature, flow rate, oxygen and hydrogen. French work (13) indicates that redox cycling will work for PWR's.

*Maybe better described as "chemistry adjustments and operational techniques."

The fact that there are various operational techniques which result in a range in levels of contamination, and therefore the need for decontamination, indicates that there are variables in plant operation which influence the degree of contamination which occurs. (14)

The EPRI Report (1) on primary system shutdown radiation levels also mentions variations possibly related to differences in operating practice and shutdown operations. These differences may be useful if applied to some degree in decon method development. Examples given are differences in use of hydrazine during refuelings, and use of peroxide during shutdowns. During the steam generator inspection outage of Indian Point 2 in November 1976, reduction of primary loop activity was accomplished by introducing oxygen after cold shutdown. About 2 ppm hydrogen peroxide was maintained during a four day cleanup period utilizing the primary blowdown purification system. Significantly lower fields were measured at several locations in the CVCS* system than had been found in previous outages (See Table 3-2). In April '77 oxygen was again used to achieve cobalt solubilization. Blowdown purification system was in use during this period and approximately 2,100 Ci were removed. A procedure was used at Point Beach (15) which differed from the Indian Point practice. Hydrogen was not reduced substantially until after cooldown and no oxygen or peroxide was added. This was for the purpose of avoiding corrosion product release from the reactor fuel cladding. The reduction in temperature and dissolved hydrogen resulted in release of Co 58 and Co 60. This illustrates the point that variations in operation techniques in regard to temperature, hydrogen and oxygen

*Chemical and Volume Control System

have an effect on cobalt release in PWR's as well as PHWR's.** Further effort to determine conditions for optimum dissolution is warranted.

Mechanical Methods

Mechanical cleaning such as brushing, wiping, scouring if found to be suitable is more applicable to specific parts and does not lend itself to whole systems. While gradually contoured pipe runs and water boxes could likely be reached, the locations inaccessible to a device or sections of complicated configurations would not be accessible for effective cleaning.

In addition, this type of cleaning is likely to remove metal along with deposit, thereby altering the surface characteristics of the base material. This increases the alloy's vulnerability for formation of local corrosion cells.

Experience with cleaning condenser tubes has shown that plugs, jets and other mechanical means effectively removed the semiadherent deposits, but the thin hard oxide coating on the tubes remains. This is the type of deposit plus the radioactive components which would be necessary to remove from nuclear primary circuitry.

When some contamination has resulted in localized deposit in addition to the thin film, a mechanical means of removing it is possible as long as the area is accessible. Techniques have been developed to accomplish this type of cleaning. It is also possible that hard film in convenient areas can be removed by extremely high velocity jetting. If such methods are introduced, a trial on a designated

**Pressurized Heavy Water Reactor

area could be made prior to a system decontamination. No costly or time consuming preparations are required. Therefore, procedures and cost development will not be necessary as part of this effort.

A summary of plant decontaminations is shown in Table 3-3 and accounts of reported decontaminations are given in more detail in Appendix A.

TABLE 3-1

UNITED NUCLEAR BEAKER TESTS

"N" REACTOR STEEL SPACER SAMPLES - 16 HOURS @ 85C

<u>SOLVENTS</u>	<u>% DECON</u>	<u>CORR. RATE MILS/DAY</u>	<u>REMARKS</u>
EDTA 10%, N_2H_4 1%	98	0.073	
EDTA 4 1/2%, NH_3 pH7	99.5	0.18	
Citrosolv, NH_3 pH 3.5	99	0.17	
EDTA 5%, N_2H_4 pH4	99	0.22	inhib. breakdown
HEEDTA 5%, N_2H_4 pH4	99	0.16	inhib. breakdown
Citrox 10%, (4521)**	99	0.49	
H PO 10%, (4521A)**	99	0.106	
EDTA 3%, Citric 2%, N_2H_4 pH7	98	0.107	
Oxalic 4 1/2%, Citric 2% NH_3 pH4	99	0.28	
EDTA 10%, NH_3 pH9	99	0.13	
DTPA 5%, N_2H_4 pH4	99	0.20	inhib. breakdown
DTPA 5%, Citric 2%, N_2H_4 pH8	98	0.15	
EDTA 10%, N_2H_4 1%	96	0.07	precip. ferrous oxalate
Amm. Oxalate 1%, NH_3 pH7			
EDTA 4.5%, Amm. Oxalate 1%	94	0.15	precip. ferrous oxalate
NH_3 pH7			
HEEDTA 10%, Amm. Oxalate 1%	96	0.25	
NH_3 pH4			

*Inhibitor - 0.1% diethylthiourea
except Turco products 4521, 4521A

**Turco solvents, Turco Products Division of Purex Corp.

TABLE 3-2

ACTIVITY SURVEY INDIAN POINT 2Nov 1976

<u>Sample Point</u>	<u>Activity mR/hr</u>	
	<u>Cleanup without Peroxide add'n</u>	<u>Cleanup with Peroxide add'n</u>
CVCS charge lines	800	-
	550	190
	-	165
	450	190
Seal water line	225	110
	150	80
	-	75
	500	200
Primary sample line		
Component cooling sample line	180	60

April 1977

<u>Sample Point</u>	<u>Activity mR/hr</u>	
	<u>March 1977</u>	<u>June 1977</u>
	<u>Before O₂ Addition</u>	<u>After O₂ Addition</u>
CVCS charge lines	550	280
CVCS seal return line	225	115
CVCS seal return line	150	120
CVCS charge line	450	200
Primary sample lines	500	200

TABLE 3-3

SUMMARY OF PLANT DECONTAMINATIONS

Facility	Yr.	Method	Stages	Major Mat'l	Comment	% Decon
PRTR	1962	Oxalic Peroxide Peracetic Gluconic AP Oxalic AC-EDTA	8	304	Too many stages req'd low DF	50%
SHPPT	1964	AP, NAOH AC	2	304	Local accumul. Expensive waste gen.	95%
PRTR	1965	AP Citrox	2	304	Success Expensive	95%
NPR ("N" reactor)	1965	AP Sulfamic Sulfamic	3	304 C-steel	Low DF waste vol.	50%
NPR ("N" reactor)	1967	Citrox AP Acid rinse Citrox	4	304 Inconel	Success but 4 stages Expensive	97%
SENA	1968	AP Citrox	2	304	Success	97%
Douglas Pt	1971	H ₂ series 02	14	Monel C-steel	Success	83%
NPR ("N" reactor)	1967	H ₃ PO ₄ NH ₃	2	C-steel	Local pitting	67-98%
SGHWR	1969	Citrox (Turco 4521)	1	C-steel Stainless	Variable Success	95%
Dresden	1968	H ₃ PO ₄	1	Stainless	Low DF	50%
Douglas Pt.	1975	Can Decon	1	C-steel	Success	84%
Gentilly 1	1974	Can Decon	1	C-steel	Success	78%

4.0 DECONTAMINATION METHODS SELECTION

The plant decontamination experience review indicates the most effective method to be the alkaline permanganate-citrox. Further demonstration of this type decontamination is not planned since it has been applied successfully in existing reactors. However, it is costly, time consuming and much waste is generated. Therefore this program effort is directed towards development of methods which are simpler and generate less waste.

Selection of possible candidates for demonstration follow:

- o The moderate success of the single stage citrox decontamination of SGHWR* indicates that trials of methods calling for one solvent may give results worthy of further development. The Dow solvent NS-3 is another possibility. Dow developed this proprietary solvent applicable to PWR's. The ESEERCO** decontamination program includes corrosion testing of Indian Point 1 materials and material interfaces. Future phases of this program leading to the demonstration decontamination using NS-3 would be dependent on the successful completion of the corrosion testing. Laboratory tests on reactor artifacts should be made to find the most effective single stage strong chemical method which would then be applied as one of the demonstrations.
- o The decontamination conducted at Gentilly-1 and Douglas Point indicate that decontamination can be accomplished with dilute solutions. Laboratory testing is required to determine the effective-

* Steam Generator Heavy Water Reactor - Winfrith, U.K.

**The Empire State Electric Energy Research Corp., Project EP-5-7

ness of mild chemical methods. Successful results of tests on PWR reactor artifacts would lead to a demonstration. The partial success of the Shippingport dilute APAC method also suggests that even more dilute solutions can possibly be used with the intent of reducing the quantity of ion exchange resin for waste treatment. A simpler and less expensive method would probably require more frequent application, but the overall expense would be less. As part of the laboratory testing for selecting the optimum dilute solvent formulation, the requirements for ion exchange resin would need to be determined.

- o The redox series conducted at Douglas Point and applications of redox during PWR shutdowns indicate its possible use as a decontamination method. Determination of the optimum parameters for this type approach should be made in the laboratory. The most successful of redox applications would be demonstrated.
- o The redox application may result in conditioning the deposit making it more amenable to removal by single stage solvent or multiple stage dilute solvent. The results of laboratory tests should define the most effective combination for demonstration.

The decontamination schemes planned for laboratory testing require additional waste tankage and waste processing facilities if applied to a PWR. Methods thus far used for plant decontaminations generally consisted of moderate concentrations of chemicals to accomplish desired results and have required more waste treatment capability than included in the nuclear station facilities. In many cases the problem was solved by contracting for the carting away of decontamination drains. But today solidification of these wastes is required.

As far as Indian Point 1 demonstrations are concerned, the methods which are considered for testing require waste processing capacity which is well within the existing capability the primary purification system and the plant radwaste complex. The pure water redox concepts with pH adjustments utilized minimal amounts of chemical; the loading on the wastes systems would then be merely the released deposits of the primary piping and equipment. The low concentration of solvents, such as that used in Canadian experience, in the volumes required for an Indian Point 1 test facility can be easily handled by the existing primary purification filters and ion exchangers. Canadian tests on higher temperature filters⁽¹⁶⁾ indicated that some waste recovery can be accomplished by use of a magnetic filter. After a Can Decon cycle at NPD reactor, a magnetic filter was effective in removing submicron corrosion products generated during the decontamination. The filter efficiency was 85%. The conceptual design of the Indian Point 1 decon test facility includes 2 filters. It could be of value to utilize a magnetic filter as one of the filter components. An in-line filter has the advantage of avoiding reprecipitation of fine particulates which might be the case if removal of particulate were left to the final drain cycle of a decontamination. A magnetic filter is considered especially since the Canadian tests indicated the magnetic filter to be more effective than a 0.45 micron cartridge filter. Parallel testing of magnetic filter and cartridge type would be of value.

5.0 CORROSION TESTING PROGRAM

5.0 CORROSION TESTING PROGRAM

The type of corrosion test program needed to support demonstration of chemical decontamination at the Indian Point 1 plant has been adequately covered in other work previously done by the Dow Chemical Co. under contract with ESEERCO (project EP-5-7). Until work is more advanced on the development of particular chemical solvents no need exists to expand on the Dow work. The Dow work relates to the use of proprietary solvent NS-3, but the prescribed test program is pertinent to test needed for other solvents. Appendix C presents the Dow-ESEERCO work plan outlining the type of test program required to insure that a solvent can be used without affecting the integrity of materials in the Indian Point 1 system. Also included in Appendix C is a summary of some of the work that has been completed on the ESEERCO project.

6.0 DECONTAMINATION SYSTEM DESCRIPTION

6.0 DECONTAMINATION SYSTEM DESCRIPTION

The conceptual decontamination system, illustrated diagrammatically in Figures 6-1 and 6-2, utilizes existing Indian Point Unit No. 1 equipment and provides the necessary new components for solvent storage, circulation, and drains storage to accomplish each of the suggested demonstrations. The layout considers only steam generator No. 11 as an example; the final design will provide connection to all four steam generators. The testing associated with each steam generator would be sequential. Refer to Appendix B for a general description of Indian Point 1 including systems to be utilized in the demonstrations.

As shown in Figure 6-1 the solvent storage tank is equipped for admitting and holding each contemplated solvent and rinse solutions. The circulating system consists of pump, heat exchanger and piping to allow heating up solution prior to admission to the steam generator and for recirculating through parallel filters to remove particulates.

The shielded solvent drain tank is sized to contain the entire solvent volume. The solvent drain may then be directed to waste or for reprocessing depending on the decontamination method being demonstrated. Strong chemical solution will be directed to the existing Indian Point 1 waste evaporation system and drumming station for concentration and finally packaging of the evaporator bottoms. Refer to Figures 6-2 through 6-4 for the system arrangement. Strong chemical rinses, redox, and dilute chemical wastes will be processed through the existing Indian Point 1 blowdown purification system. Refer to Figure 6-5 for the system arrangement.

This system consists of blowdown storage tanks followed by pumps to transfer the solution through filters. The temperature of the solution is then reduced by the purification inlet coolers before admission to the ion exchangers. The ion exchange system consists of cation units, mixed bed units and anion exchangers. These are arranged so that they can be used in a variety of sequences as the procedures dictate.

The effluent of the ion exchangers will then be directed to the solvent storage tank for recirculation or for final draining through the solvent drain tank, if required.

Rinse drains may be directed to a blowdown deionizer or direct to discharge canal if activity meets requirements. The routes are shown in Figure 6-6.

Conceptual arrangement of the new equipment and piping is shown in drawings 21995-A and 21996-A.

Decontamination System Components

The components required to be installed for the decontamination system are described below.

Solvent Storage Tank (1)

Volume - 5,000 gallons

Dimensions - 8'd x 14' l

Material - stainless steel

Connections - drain (suction to pump),

heater return, circulation return,

solvent inlet, hydrogen inlet, oxygen

inlet, nitrogen inlet, vent to waste
gas system.

Solvent Heater (1)

Capacity - 150,000 lb/hr
Heat cap. - 7,665,000 BTU/hr
Ht. Source - 8,880 lb/hr, 150 psi steam
Material - shell - steel
 tubes - stainless steel
Dimensions - 2'd x 10' 1
Connections - steam inlet, condensate
 outlet, solution inlet, solution outlet.

Recirculation Pump (1)

Capacity - 300 gpm (150,000 lb/hr)
Dimensions - 5'L, 2'W, 2'h.
Material - stainless steel

Steam Generator Heat Exchanger Connections

Six inch handholes are utilized for solvent entry and
return. Weld three inch connection on six inch plates to fit.

Filters (2)

Design pressure	150 psig
Design temperature	250 F
Flow lbs/hr	90,000 lb/hr
Shell material	stainless steel
Filter material	cellulose acetate
Effective filtration	10 micron

Waste Storage Drain Tank (1)

Volume	5,000 gallon
Dimensions	8'd x 14' l
Shielding	8" concrete
Material	stainless steel

Route for all decon processes
 Additional Route for Redox & Dilute Chem

DECON SYSTEM DIAGRAM - CIRCULATION ROUTE

Strong Chem. Drain Route

DECON SYSTEM DIAGRAM - STRONG CHEM. DRAIN ROUTE

FIGURE 6-3

IPI - WASTE EVAPORATOR SYSTEM (EXISTING)

FIGURE 6-4

IPI - WASTE DRUMMING STATION (EXISTING)

IPI BLOWDOWN PURIFICATION SYSTEM (EXISTING)

IPI - CONTAMINATED RINSE DRAINS (EXISTING)
ALTERNATE ROUTE

NOTES:

WORK THIS SHEET WITH DWG 21996.

DIMENSIONS AND SIZES OF EQUIPMENT, STRUCTURES, ETC INDICATED ARE FOR ESTIMATE ONLY.

REFERENCE DRAWINGS

141099 - SPHERE AREA STEEL FRAMING PLAN EL. 70'-0"
CONCEPT DWG. 21996 - PRIMARY DECON. SYSTEM PLAN
'AT EL. 53'-0"

[illegible]

NOTES:
 1- WORK THIS DWG WITH DWG 21996-A.
 2- INSTALLATION OF NEW PIPING SHALL BE IN ACCORDANCE WITH THE CODE OF NEW YORK CITY.
 3- THE USE OF EXISTING PIPING INDICATED MUST BE REVIEWED BY NUCLEAR POWER GENERATION FOR THEIR APPROVAL.
 4- AFTER THE DECONTAMINATION SYSTEM HAS BEEN COMPLETED, ALL CUT SECTIONS OF EXISTING PIPE ARE TO BE RESTORED TO THEIR ORIGINAL FLOW CONDITIONS.

CONCEPT DWG.		21996-A	
CONSOLIDATED EDISON CO. OF NEW YORK, INC.			
STATION LOCATION INDIAN POINT #1			
PRIMARY-DECON. SYSTEM			
ARRGT OF PUMP, FILTERS & ASSOC. PIPING FOR BLR #11			
AT ELEV. 53'-0"			
DATE 12/2/67	BY V. SHEEN	REVIEWED	
NO 6374	CHECKED E.C.	DATE	
DESIGNED	BY	DATE	
APPROVED	BY	DATE	
APPROVALS			
DESIGNED	BY	DATE	
CHECKED	BY	DATE	
APPROVED	BY	DATE	
DESIGNED	BY	DATE	
CHECKED	BY	DATE	
APPROVED	BY	DATE	
DESIGNED	BY	DATE	
CHECKED	BY	DATE	
APPROVED	BY	DATE	

7.0 DECONTAMINATION PROCEDURES

7.0 DECONTAMINATION PROCEDURES

The following conceptual procedures are for each of the methods considered for demonstration. They were prepared on the basis of the drawings and diagrams developed for the study and on information obtained from accounts of decontaminations and descriptions of methods. A list of the conceptual procedures to be discussed follows:

- o Strong Chemical

- Single stage application of a solvent intended to remove most of the contaminants.

- o NS-3 Chemical

- A particular type of "strong solvent" under development by ESEERCO and Dow.

- o Redox

- A method using alternating additions of oxygen and hydrogen combined with system disturbance (pump action) to remove small quantities of contaminants by the purification system.

- o Dilute Chemical

- Application of small quantities of chemicals utilizing the purification system to remove moderate quantities of contaminants.

CONCEPTUAL PROCEDURE-STRONG CHEMICAL

(1) Test Run and Circuit Heat Up

- (a) Open demin supply valve and fill solvent storage tank to desired volume.
- (b) Start solvent pump; arrange valving to recycle water to storage tank for heat-up.
- (c) Admit steam to heat exchanger, controlling steam flow to obtain solvent (water) outlet temperature in specified time.
- (d) Establish deionized flow to stop valve seals.
- (e) Establish deionized flow to RCP motor (Attach vent tubing to vent header).
- (f) Close recycle valve and open valve to admit solvent (water) to nuclear heat exchanger.
- (g) Open nuclear heat exchanger vent pipes and sight glasses; close down exit valve in order to fill nuclear heat exchanger to vents (level with impeller of RCP).
- (h) Close vents on nuclear heat exchanger and RCP motor.
- (i) Continue pumping, adjusting exit valve as required to maintain level in nuclear heat exchanger vents.
- (j) When there is no further decrease in storage tank level the system is full; recirculate for two hours monitoring temperature and activity at several locations.
- (k) Drain solvent (water) to blowdown purification system.

(2A) Prepare Solvent and Heat Up

- (a) Verify solvent concentration and inhibitor effectiveness and prepare to pump from tanker to storage tank.
- (b) Pump solvent and admit demin water to obtain required volume and concentration in solvent storage tank. Test for dilute solvent concentration and obtain liquid level.
- (c) Start circulating pump; arrange valving to recycle solvent to storage tank for heat up.
- (d) Admit steam to heat exchanger; controlling steam flow to obtain desired solvent outlet temperature in specified time.
- (e) Recirculate until heater outlet temperature reaches designated amount.

(2B) Filling Heat Exchanger with Solvent

- (a) Verify solvent concentration and temperature; insure sufficient volume of solvent to fill entire decon circuit.
- (b) Start solvent pump.
- (c) Establish deionized flow to stop valve seals.
- (d) Establish deionized flow to RCP motor (Attach vent tubing to vent header).
- (e) Throttle nuclear heat exchanger exist valve until solvent appears in vent pipe sight glass (level with impeller of RCP).
- (f) Close vents on primary heat exchanger and RCP motor.
- (g) Continue feed adjusting exist valve as required to maintain level in heat exchanger vents.

(2B) Filling Heat Exchanger with Solvent - (Cont'd)

- (h) When there is no further decrease in storage tank level the system is full, and the recirculation period commences.

(3) Recirculation

- (a) Recirculation started - item 2B h.
- (b) During recirculation observe:
 - 1. Level in nuclear heat exchanger vents.
 - 2. Water pressure on stop valve seals and RCP motor.
 - 3. Temperature of solvent entering nuclear heat exchanger.
- (c) Monitor Radioactivity at designated areas of decon circuit for:
 - 1. Build up on filters.
 - 2. Evidence of replating at turns or elbows.
 - 3. Reduction in nuclear heat exchanger.
 - 4. Buildup in solvent storage tank.
- (d) Observe corrosion monitor readings to insure inhibitor integrity.
- (e) Monitor pressure drop across filters and replace if pressure drop or activity reach cut off levels.
- (f) Obtain periodic samples before and after filters for analysis.
- (g) Continue recirculation designated time or until observations and analyses indicate completion of decon.

(4) Solvent Drain and Waste Treatment

- (a) Open all drains and vents allowing spent solvent access to suction of circulation pump for transfer.

(4) Solvent Drain and Waste Treatment (Cont'd)

- (b) Start pump, transferring spent solvent to waste tank.
- (c) Open WST drain of waste evaporator feed pump.
- (d) Start up evaporator feed pump and initiate evaporator operation.
- (e) Monitor evaporator concentrates frequently to determine suitable concentration/activity for transfer to packaging system.
- (f) Monitor evaporator overhead purity/activity to determine suitability for recovery or discharge.
- (g) When evaporator concentrate concentration/activity reaches appropriate level, transfer to solidification system.

(5&6) Rinses

- (a) Fill solvent tank with deionized water.
- (b) Start solvent pump and fill as in 2B c through h.
- (c) Recirculate for ten minutes.
- (d) Monitor radio activity at designated areas of decon circuit as in 3c.
- (e) Monitor pressure drop across filters.
- (f) Open for transfer all vents and drains allowing rinse water access to suction of circulation pump to waste solvent tank.
- (g) Start pump.
- (h) Test for activity, TDS and pH.
- (i) Dependent on above readings drain through blowdown purification system for rinse water recovery or route to waste evaporator for processing.

(5&6) Rinses (Cont'd)

- (j) A second rinse will probably be required. Drains should be suitable for recovery through blowdown purification system.

CONCEPTUAL PROCEDURE-NS-3 CHEMICAL

(1) Test Run and Circuit Heat Up

- (a) Open demin supply valve and fill solvent storage tank to desired volume.
- (b) Start solvent pump; arrange valving to recycle water to storage tank for heat-up.
- (c) Admit steam to heat exchanger, controlling steam flow to obtain solvent (water) outlet temperature in specified time.
- (d) Establish deionized flow to stop valve seals.
- (e) Establish deionized flow to RCP motor (Attach vent tubing to vent header).
- (f) Close recycle valve and open valve to admit solvent (water) to nuclear heat exchanger.
- (g) Open nuclear heat exchanger vent pipes and sight glasses; close down exit valve in order to fill nuclear heat exchanger to vents (level with impeller or RCP).
- (h) Close vents on nuclear heat exchanger and RCP motor.
- (i) Continue pumping, adjusting exit valve as required to maintain level in nuclear heat exchanger vents.
- (j) Where there is no further decrease in storage tank level the system is full; recirculate for two hours monitoring temperature and activity at several locations.
- (k) Drain solvent (water) to blowdown purification system.

(2A) Prepare Solvent and Heat Up

- (a) Verify solvent concentration and inhibitor effectiveness and prepare to pump from tanker to storage tank.
- (b) Pump solvent and admit demin water to obtain required volume and concentration in solvent storage tank. Test for dilute solvent concentration and obtain liquid level.
- (c) Start circulating pump; arrange valving to recycle solvent to storage tank for heat up.
- (d) Admit steam to heat exchanger; controlling steam flow to obtain desired solvent outlet temperature in specified time.
- (e) Recirculate until heater outlet temperature reaches designated amount.

(2B) Filling Heat Exchanger with Solvent

- (a) Verify solvent concentration and temperature; insure sufficient volume of solvent to fill entire decon circuit.
- (b) Start solvent pump.
- (c) Establish deionized flow to stop valve seals.
- (d) Establish deionized flow to RCP motor (Attach vent tubing to vent header).
- (e) Throttle nuclear heat exchanger exit valve until solvent appears in vent pipe sight glass (level with impeller of RCP).
- (f) Close vents on primary heat exchanger and RCP motor.
- (g) Continue feed adjusting exit valve as required to maintain level in heat exchanger vents.

(2B) Filling Heat Exchanger with Solvent - (Cont'd)

- (h) When there is no further decrease in storage tank level the system is full, and the recirculation period commences.

(3) 1st Stage Recirculation

- (a) Recirculation started - item 2B h.
- (b) During recirculation observe:
 - 1. Level in nuclear heat exchanger vents.
 - 2. Water pressure on stop valve seals and RCP motor.
 - 3. Temperature of solvent entering nuclear heat exchanger.
- (c) Monitor Radioactivity at designated areas of decon circuit for:
 - 1. Build up on filters.
 - 2. Evidence of replating at turns or elbows.
 - 3. Reduction in nuclear heat exchanger.
 - 4. Buildup in solvent storage tank.
- (d) Observe corrosion monitor readings to insure inhibitor integrity.
- (e) Monitor pressure drop across filters and replace if pressure drop or activity reach cut off levels.
- (f) Obtain periodic samples before and after filters for analysis.
- (g) Continue recirculation designated time or until observations and analyses indicate completion of 1st stage decon.

(4) Second Stage Fill & Recirculation

- (a) While continuing to recirculate 1st stage solution, commence pumping second stage solvent to solvent storage tank at a rate not exceeding the recirculation rate.
- (b) Reset heater controls to obtain temperature desired.
- (c) When second stage solvent fill is complete, continue recirculation, observing:
 - 1. Level in nuclear heat exchanger vents.
 - 2. Water pressure on stop valve seals and RCP motor.
 - 3. Temperature of solvent entering nuclear heat exchanger.
- (d) Monitor radioactivity at designated areas of decon circuit for:
 - 1. Build up on filters.
 - 2. Evidence of replating at turns or elbows.
 - 3. Reduction in nuclear heat exchanger.
 - 4. Build up in solvent storage tank.
- (e) Observe corrosion monitor readings to insure inhibitor integrity.
- (f) Monitor pressure drop across filters and replace if pressure drop or activity reach cut off levels.
- (g) Obtain periodic samples before and after filters for analysis.
- (h) Continue recirculation for designated time or until observations and analyses indicate completion of decon.

(5) Solvent Drain and Waste Treatment

- (a) Open all drains and vents allowing spent solvent access to suction of circulation pump for transfer to waste solvent tank. (WST)
- (b) Start pump, transferring spent solvent to waste tank.
- (c) Open WST drain of waste evaporator feed pump.
- (d) Start up evaporator feed pump and initiate evaporator operation.
- (e) Monitor evaporator concentrates frequently to determine suitable concentration/activity for transfer to packaging system.
- (f) Monitor evaporator overhead purity/activity to determine suitability for recovery or discharge.
- (g) When evaporator concentrate concentration/activity reaches appropriate level, transfer to solidification system.

(6&7) Rinses

- (a) Fill solvent tank with deionized water.
- (b) Start solvent pump and fill as in 2B c through h.
- (c) Recirculate for ten minutes.
- (d) Monitor radio activity at designated areas of decon circuit as in 3c.
- (e) Monitor pressure drop across filters.
- (f) Open all vents and drains allowing rinse water access to suction of circulation pump for transfer to waste solvent tank.
- (g) Start pump.

(6&7) Rinses - (Cont'd)

- (h) Test for activity, TDS and pH.
- (i) Dependent on above readings drain through blowdown purification system for rinse water recovery or route to waste evaporator for processing.
- (j) A second rinse will probably be required. Drains should be suitable for recovery through blowdown purification system.

REDOX CONCEPTUAL PROCEDURE

HYDROGEN CYCLE

- (1) Fill decon circuit with deionized water as in Strong Chemical Procedure #1a through #1i.
- (2) Open blowdown purification circuit, fill with deionized water, admitting make-up as required to maintain level in solvent storage tank.
- (3) Admit deionized water to raise water level in solvent storage tank to vent.
- (4) Reduce level to operating level by draining under hydrogen. Start to pressurize system to designated hydrogen pressure. Check potential leakage areas for hydrogen:
 - (a) Circulation pump seals.
 - (b) RCP seals and motor vent.
 - (c) Check valve.
 - (d) Stop valve seals.
- (5) Pressurize to designated hydrogen pressure and start circulating pump, diverting designated portion to purification system utilizing mixed bed ion exchanger.
- (6) Circulate, maintaining temperature, and monitor for control and observation purposes:
 - (a) Pressure.
 - (b) Temperature.
 - (c) Dissolved Hydrogen.
 - (d) pH.
 - (e) Activity in designated areas.

REDOX PROCEDURE - (Cont'd)

- (f) Hydrogen adjacent to pumps, seals, vents and other valved or capped connections.
- (g) Corrosivity.
- (7) During circulation record activity at filters and deionizer and sample fluid for testing of Fe, Cr, Co, Mn and radionuclides.
- (8) When activity on filters, deionizer, and in fluid no longer increases, circulation period is ended.
- (9) If temperature change is required, heat or cool as required making sure to record activity readings just prior to any controlled change.
- (10) Upon completing hydrogen cycle, raise level in storage tank venting hydrogen to primary vent header; reduce pressure.
- (11) Drain under nitrogen to operating level in solvent storage tank. Take samples of drains for determination of Co, Mn, Fe, etc.
- (12) Measure fields in area of steam generator to determine reduction in fields as a result of hydrogen cycle.
- (13) Measure activity build up in filters and ion exchangers of blowdown purification system.

OXYGEN CYCLE

- (1) Prepare for oxygen cycle raising level to storage tank vent.
- (2) Reduce level to operating level by draining under oxygen. Start pressurizing system to designated oxygen pressure. Check potential leakage areas for oxygen:
 - (a) Circulation pump seals.
 - (b) RCP seals and motor vent.

REDOX PROCEDURE - (Cont'd)

- (c) Check valve.
- (d) Stop valve seals.
- (3) Pressure to designated oxygen pressure and start circulating pump, diverting designated portion to purification system.
- (4) Continue as in steps 5 to 12 (substituting oxygen for hydrogen in all steps).
- (5) If second cycle of redox method is required, repeat step 1 and continue.
- (6) Complete decontamination with hydrogen cycle.

CONCEPTUAL PROCEDURE - DILUTE CHEMICAL

(1) Test Run & Circuit Heat Up

- (a) Open demin supply valve and fill solvent storage tank to desired volume.
- (b) Start solvent pump; arrange valving to recycle water to storage tank for heat-up.
- (c) Admit steam to heat exchanger, controlling steam flow to obtain solvent (water) outlet temperature in specified time.
- (d) Establish deionized water flow to stop valve seals.
- (e) Establish deionized water flow to RCP motor (Attach vent tubing to vent header).
- (f) Close recycle valve and open valve to admit solvent (water) to nuclear heat exchanger.
- (g) Open nuclear heat exchanger vent pipes and sight glasses; close down exit valve in order to fill nuclear heat exchanger to vents (level with impeller of RCP).
- (h) Close vents on nuclear heat exchanger and RCP motor.
- (i) Continue pumping, adjusting exit valve as required to maintain level in nuclear heat exchanger vents.
- (j) When there is no further decrease in storage tank level the system is full; recirculate for designated time monitoring temperature and activity at several locations.

(1) Test Run & Circuit Heat Up (Cont'd)

- (k) Divert designated portion of solvent (water) to blowdown purification system. (bps)
- (l) Continue recirculating for designated time utilizing mixed bed of bps, monitoring activity at appropriate locations.

(2A) Prepare Solvent and Heat Up

- (a) Verify solvent concentration and inhibitor effectiveness and prepare to pump solvent to storage tank.
- (b) Pump solvent and admit demin water to obtain required volume and concentration in solvent storage tank. Test for dilute solvent concentration and obtain liquid level.
- (c) Start circulating pump; arrange valving to recycle tank for heat up.
- (d) Admit steam to heat exchanger; controlling steam flow to obtain desired solvent outlet temperature in specified time.
- (e) Recirculate until heater outlet temperature reaches designated amount.

(2B) Filling Heat Exchanger with Solvent

- (a) Verify solvent concentration and temperature; insure sufficient volume of solvent to fill entire decon circuit.
- (b) Start solvent pump.
- (c) Establish deionized water flow to stop valve seals.

(2B) Filling Heat Exchanger with Solvent - (Cont'd)

- (d) Establish deionized water flow to RCP motor (Attach vent tubing to vent header).
- (e) Throttle nuclear heat exchanger exit valve until solvent appears in vent pipe sight glass (level with impeller of RCP).
- (f) Close vents on primary heat exchanger and RCP motor.
- (g) Continue feed adjusting exit valve as required to maintain level in heat exchanger vents.
- (h) When there is no further decrease in storage tank level the system is full, and the recirculation period commences.
- (i) Divert designated portion to bps utilizing cation exchanger.

(3) Recirculation of Solvent

- (a) Recirculation started - items 2Bh & 2Bi.
- (b) During recirculation observe:
 - 1. Level in nuclear heat exchanger vents.
 - 2. Water pressure on stop valve seals and RCP motor.
 - 3. Temperature of solvent entering nuclear heat exchanger.
- (c) Monitor radioactivity at designated areas of decon circuit for:
 - 1. Build up on filters and cation exchanger.
 - 2. Evidence of replating at turns or elbows.
 - 3. Reduction in nuclear heat exchanger.
 - 4. Buildup in solvent storage tank.
- (d) Monitor dissolved hydrogen and pH.

(3) Recirculation of Solvent (Cont'd)

- (e) Observe corrosion monitor readings to insure inhibitor integrity.
- (f) Monitor pressure drop across filters and replace if pressure drop or activity reach cut off levels.
- (g) Obtain periodic samples before and after filters and cation exchanger for analysis.
- (h) Continue recirculation for designated time or until observations and analyses indicate completion of cycle.

(4) Removal of the Solvent

- (a) When control indicates cleaning is complete cut out cation exchanger diverting flow to mixed bed ion exchanger. Utilizing mixed bed exchanger in place of cation results in solvent removal.
- (b) Continue recirculation until observations and analyses (as in 3) indicate solvent is removed.

8.0 COSTS

8.0 COST OF PROGRAM

The costs of the remaining phases of the decontamination project are estimated in this section. The cost is estimated to total \$7,000,000. A summary of the expenditures for each phase of the project is included. Cost development data is shown in Appendix D.

Laboratory Tests, concepts review and final design

The estimate of the costs of Phases 2 and 3 of the program is summarized as follows:

Laboratory Testing	\$1,470,000
Review & refine design concepts	140,000
Detailed design of systems	420,000
Pre-demo inspections	<u>170,000</u>
	\$2,200,000

Construction of demonstration facility

The engineering estimate for constructing the decontamination demonstration system at Indian Point 1 is based on installation of equipment in 1983 and provides for demonstration of four processes. The estimate is based on drawings No. 21995 and 21996 (Section 6). Drawings show piping and connections for conducting decontamination of steam generator No. 11 only. The estimate has been increased to include connections to the other steam generators; it is summarized below:

Equipment	\$ 340,700
Labor	400,000
Overhead	439,800
HP & QA Support	182,000
Refurbish't	700,000
Escalation	<u>763,000</u>
	\$2,825,500

Demonstrations

An estimation of the costs of conducting the Indian Point 1 demonstration includes the cost of the chemicals, manpower, and waste treatment and disposal. The basic items required to calculate the costs are given.

Basic Data

In order to estimate requirements for the various decontamination steps the following information is needed:

1. The volume of the applicable part of the primary loop is needed to determine amount of chemical, purifications requirements and disposal costs. The volume of a single primary loop bounded by the stop valves and steam generator is estimated to be 3,750 gallons and 250 gallons are allowed for the decon circuit totaling 4,000 gallons.
2. An estimate of the amount of deposit is required for predicting the ion exchange resin needs for demonstration of a dilute chemical decontamination. Deposit obtained from the surface of the primary loop of Indian Point 1 amounted to 1.0 mg/cm^2 (0.00387 lb/ft^2). It is estimated that the thickness of the Indian Point 1 deposits is 4 microns. For an estimated 13,000 ft^2 of surface with a 4 micron deposit, one loop of Indian Point 1 contains about 50 lbs. of deposit.
3. The specific activity of deposited corrosion product in Indian Point 1 is needed to estimate the waste disposal costs. For a reference date of January 1, 1979, activity due to deposits in Indian Point 1 is estimated to be 11.5 uCi/cm^2 . Extrapolating to a single loop of Indian Point 1, the activity is 140 Ci.

4. The efficiency of each solvent is required to estimate disposal costs. The following decontamination efficiencies are based on those given for the various experiments, demonstrations, and other experience in the literature.

<u>Method</u>	<u>Efficiency %</u>
Strong Chemical	90
Dilute Chemical	75
Redox	67
NS 3	93

The estimated cost summary of the decontamination demonstrations follows:

Pre-operational Training	\$ 145,000
Labor for Demos	1,014,000
Chemicals & Resin	97,000
Waste Trtmt. & Disposal	211,000
Assoc. Corrosion Testing	178,000
Post Inspections	220,000
Gen'l & Admin. Expense (5.88%)	<u>110,000</u>
	\$1,975,000

Schedule

The schedule associated with these cost estimations is shown in Figure 8-1.

Cost Benefit

In keeping with the nation's, the industry's, and the NRC's concern with radiation exposure, the benefits to be gained by reducing

fields as a result of decontamination are obviously of great value. If the practice of decontamination results in reduced worker exposure and possible reduction in the generation of wastes, it may result in helping to improve the climate in regard to acceptance of nuclear plants. This benefit cannot be measured, but any reduction in the exposure of workers or in shortening outage time will be extremely beneficial. In terms of the economics of decontamination for an operating plant, an estimate of the cost of worker exposure to radiation is required to determine a breakeven point in justifying the cost of decontamination. It is especially important in considering the dilute chemical and redox methods that are expected to yield lower decon factors than are obtained by use of more concentrated solvents. The cost of worker exposure is estimated to be \$5,300 per manrem based on exposure of 500 manrem per year for a 900 MW unit. Calculations are given in Appendix D. Costs of the decontamination methods will be compared against this value as well as the NRC assigned figure of \$1,000 per manrem.

The estimated costs for each of the decontamination methods have been scaled up to fit a 900 MW PWR unit (see Table 8-1) in order to make a cost/benefit judgement. The annual plant costs are given below and the resultant cost or benefit is given according to the estimated cost/benefit figure or the value assigned by the NRC.

<u>Method</u>	<u>Decon*</u> <u>Cost/\$yr.</u>	<u>Net Benefit - \$/yr.</u>	
		<u>Calculated</u> (basis \$5300/manrem)	<u>NRC</u> (basis \$1000/manrem)
Strong Chemical	953,000	+ 1,420,000	-500,000
Redox	889,000	+ 880,000	-560,000
NS-3	1,052,000	+ 1,410,000	-590,000
Dil. Chem.	903,000	+ 1,080,000	-530,000

Note: Assumed decontamination not an outage critical path and therefore no outage time cost was assigned to decontaminations.

According to the NRC manrem cost, there is a net loss associated with each decon method but a substantial net gain according to the cost/benefit calculation developed in this report. There can be also a cost saving when a decontamination results in reduction in plant outage time. If maintenance and repair is accomplished in a shorter time as a result of reduction in fields by decontamination, a saving of one day means a saving of about \$500,000.** This is accomplished by generating nuclear power rather than fossil fueled. It is enough to make decon profitable under each cost/benefit estimate.

* See Table 8-1 for breakdown of decon cost per year.

**Based on Con Edison average unit generation costs for Indian Point 2 vs. fossil based load units 900 MW for 24 hour operation.

A

B

PROPOSED MASTER SCHEDULE

NOTE: All Activity Durations Are In MONTHS										CONSOLIDATED EDISON CO. of NEW YORK		INDIAN POINT 1 - DECONTAMINATION DEMONSTRATION FACILITY					
REV.	DATE	PROJ. ENG.	PROJ. MGR.	REV.	DATE	PROJ. ENG.	PROJ. MGR.	SCHEDULING ENGINEER	APPROVED BY PROJECT ENGR.	APPROVED BY PROJECT MGR.	REVISION	ISSUE NO.	STATUS DATE	REQD. SERV. DATE	PROJ. NO.	S. R. NO.	SHEET NO.
								HERSKOWITZ				1	11-20-78		6347		1051

FORM NO - PSD 1 (rev 4-78)

8-0

8-0

TABLE 8-1

ESTIMATE OF DECON COSTS FOR 900 MW PWR UNIT
THOUSANDS OF DOLLARS

	<u>Strong Chem.</u>	<u>Redox</u>	<u>NS3</u>	<u>Dil. Chem.</u>
(1) System Capital Cost	3,600	2,500	3,600	2,700
(2) Carry'g Charge (18%/yr.)	640	450	640	480
(3) Cost of a Decon (chemical, waste, labor)	627	439	825	635
(4) Decon Frequency Period-Yrs.	2	1	2	1 1/2
(5) Cost of Decon per yr. (item 3 4)	313	439	412	423
(6) Total Cost per year (items 2 + 5)	953	889	1,052	903

Note: No downtime charges assigned to decontaminations.

9.0 PHASE 2-5 REQUIREMENTS

OUTLINE OF PHASES 2 & 3 OF DECONTAMINATION PROJECT

As discussed in the Decontamination Method Selection section of this study, some laboratory testing is required to obtain the definitive solvent description for each general method that has been considered. This part of the report specifies the work which will be proposed for continuing the project. It is intended that the proposal would include Phases 2 & 3 as described in the original proposal for Phase 1 (dated 3/12/76). This calls for the conducting of laboratory tests and the detailed design and cost estimate for conducting the demonstrations. The Phases 2 and 3 effort would include (1) subcontracting for laboratory testing; (2) subcontracting for review of the conceptual designs before proceeding with detailed design; and (3) the detailed design and cost estimate. Subcontractor input would be sought for advisory purposes.

1. Laboratory testing would be proposed to:

- A) Conduct tests on performance of Candecon formulation on PWR deposit. Test other mild chemical methods including NTA. Decon factor sought would be no less than three. Primary tubing samples would be obtained from Indian Point Unit 2 sections cut out for the secondary chemical cleaning purposes. Related information to be obtained would be:
 - a. Usefulness of resin clean-up system and evaporation for solvent upgrading and waste treatment.
 - b. Rate of solvent clean-up sufficient to accomplish requirements.

- B) Conduct tests to reach optimum redox conditions for decontamination. This would include varying concentration of redox, temperature, number of cycles, and pH. The effect of redox in conjunction with single stage chemical treatment would be part of this effort.
- C) Determine effectiveness of single stage CITROX decontamination (as in SGHWR decon). Test other single stage strong chemical similar to reported decon formulations tested on N reactor carbon steel spacers.

2. It would be proposed to subcontract for:

- A) Review Phase 1 concept for the application of dilute chemical type of decontamination; offer comments or improvements.
- B) Review Phase 1 concept for the application of strong chemical (CITROX, NS-3) type of decontamination; offer comments or improvements.
- C) Review Phase 1 concept for the application of redox type decontamination, and offer comments or improvements.

3. Detailed design:

- A) On the basis of findings of parts 1 and 2, above, a detailed design would be prepared. This would include complete detailed piping and equipment drawings. Controls, monitors, and valving would be specifically sized and located. Electrical drawings would also be prepared. Development of these drawings would allow the preparation of a definitive cost estimate and schedule for use in proposing the construction phase of the project.

OUTLINE OF PHASES 4 & 5 OF DECONTAMINATION PROJECT

Following DOE approval of design and planned expenditure for construction, Phase 4 would commence. Material and equipment would be purchased and the construction of the multiple - decontamination system would be started. The construction phase of the project would be accomplished by subcontractor under close supervision by Con Edison.

Parallel with this effort the existing equipment of Indian Point 1 associated with the demonstrations would be surveyed and refurbished where found necessary.

Upon completion of construction, Phase 5 would start with the finalizing of detailed procedures and training of new operators. Equipment would be checked out and each procedure would be carried out using deionized water. Finally the demonstrations would be accomplished and a project report issued.

10.0 REFERENCES

REFERENCES

1. EPRI Report 404-2 Dec. 1975, Primary System Shutdown Radiation Levels at Nuclear Power Generating Stations.
2. J.A. Ayres, Decontamination of Nuclear Reactors and Equipment, Ronald Press, NY, 1970.
3. C.S. Abrams, E.A. Salterelli, Decontamination of the Shippingport Atomic Power Station, WAPD-299 Jan. 1966.
4. W.D. Bainard, Decontamination of N-Reactor Stainless Steel Generators, NACE Conference 1969, Houston, Texas.
5. W.K. Kratzer, Single Pass Through - Reactor Decontamination of N-Reactor Carbon Steel Piping, Proceedings NACE Conference 1969, Houston, Texas.
6. H. Couez, L.F. Picone, Decontamination of a PWR Primary System SENA Plant, Proceedings of the American Power Conference vol. 33, 1971.
7. G.J.C. Nash, Decontamination of the SGHWR Prototype, Seminar on Decontamination of Nuclear Plants 7-9 May 1975, Columbus, Ohio.
8. P.J. Pettit et al, Decontamination of the Douglas Point Reactor by the CAN-DECON Process, Paper No. 39 Corrosion 78 Conference Houston Texas March 6-10, 1978.
9. B. Montford, Techniques to Reduce Radiation Fields Canadian Nuclear Ass'n Meeting, June 1976.
10. J.E. LeSurf, Control of Radiation Exposure at Candu Nuclear Power Stations, J. Br. Nuclear Energy Soc. 1977 Jan., No. 1, 53-61.
11. London Nuclear Decontamination Limited, The Candecon Process for Decontaminating Nuclear Reactors, Presentation to NRC, February 1978.
12. B. Montford, Decontamination by Cycling Techniques at the Douglas Point Nuclear Generating Station, American Power Conf. Proceedings, 1973.
13. Ph. Berge and J.J. Camp, Decontamination of the Primary Circuits by Dissolution of the Corrosion Products in the Primary Coolant at Low Temperature, Seminar on Decontamination of Nuclear Plants, 7-9 May 1975 Columbus, Ohio, co-sponsors EPRI, ASME, Ohio State Univ.

REFERENCES - (CONT'D)

14. D.J. McCormick, Overall View of Decontamination Plans in the Nuclear Industry, Seminar on Decontamination of Nuclear Plants, 7-9 May 1975, Columbus, Ohio.
15. EPRI Report 821-1 April 1978, Effects of Hydrogen Peroxide Additions on Shutdown Chemistry Transients at Pressurized Water Reactors.
16. E.J. Moskal et al., High Flow, High Temperature Magnetic filtration of the Primary Heat Transport Coolant of the Candu Power Reactors, Conference on Water Chemistry of Nuclear Reactor System, October 24, 1977, Bournemouth, UK.
17. R.M. Fryar, Plutonium Recycle Test Reactor, Nucleonics, 1958 No. 1.
18. J.A. Ayres, L.D. Perrigo, R.D. Weed, Decontamination of a PWR, Nucleonics, 1967 No. 4.
19. S.F. Whirl, J.A. Tash, Radioactive Waste Processing Control, Shippingport Atomic Power Station, ASTM Special Technical Publication No. 235, 1958.
20. The Shippingport Pressurized Water Reactor, Addison-Wesley Publishing Co.
21. J.A. Ayres, Decontamination of Water - Cooled Nuclear Reactors Status & Future Research, BNWL-SA-2460.

Appendix A

Decontamination Summaries

Decontamination Summaries

Decontamination experience gained at the following nuclear plants is discussed in this appendix.

Douglas Point	Page A-1
PRTR 1962, 1965	A-7
Shippingport	A-17
NPR Steam Generator	A-23
N-Reactor	A-28
SENA	A-32
SGHWR	A-37

DECONTAMINATION BY CYCLING TECHNIQUES AT THE (12)
DOUGLAS POINT NUCLEAR GENERATING STATION

I. Plant Description

Power Rating - 200 MWe

Type: Station is prototype CANDU-PHWR: Canadian Deuterium Uranium, Pressurized Heavy Water Reactor.

Location: On the eastern shore of Lake Huron, near Port Elgin, Ontario.

Fuel: Natural Uranium

Moderator & Coolant: Heavy Water pressurized with hydrogen.

Cooling Loop: Cycle uses primary and secondary loops like PWR
Reactor: Uses pressure tubes

Steam Generator: Uses pressure tube heat exchanger.

II. Heat Transfer System Data and Principal Construction Material

Coolant Inventory	~60,000 kg D ₂ O (~132,000 lb.)
Total Coolant Flow	180 to 200 x 10 ³ kg/min. (23.8 to 26.4 x 10 ⁶ lb/hr)
Temp., reactor outlet	293 C (559.4 F)
Temp., reactor inlet	249 C (480.2 F)
Pressure	9.02 MN/m ² (1,325 psia)

Info from: American Power Conference, Proceedings 1971

125

by the other.

5

by the other.

by the other.

by the other.

by the other.

by the other.

by the other.

by the other.

by the other.

II. Heat Transfer System Data and Principal...Cont'd

Surface Area Materials

Boilers	Monel 400 - $6.4 \times 10^3 \text{ m}^2$ (69,000 FT ²)
Headers, Feeders and Piping	C. Steel - $1.3 \times 10^8 \text{ m}^2$ ($1.4 \times 10^9 \text{ FT}^2$)
Pressure Tube and Fuel Sheath	Zircaloy - 2 - $2.06 \times 10^8 \text{ m}^2$ ($2.22 \times 10^9 \text{ FT}^2$)

III. Experiences at Douglas Point Station

1. Criticality and first power were achieved in 1966.
2. There were frequent shutdowns for system modification.
3. Oxygen entered the primary coolant system during outages.
4. Corrosion products were formed. Some adhered to internal piping and equipment surfaces. Some products, in the form of a fine suspension, were carried through the primary loop and were exposed to the neutron flux of the fuel. The neutron flux produced radioactivity in the suspended corrosion products, which then caused radiation fields wherever they were later deposited.
5. By 1969, there were severe maintenance difficulties due to the increasing radiation fields caused by the activated corrosion products which were accumulating.

1/25

e

CONFIDENTIAL - SECURITY INFORMATION

105

105

III. Decontamination Experience...Cont'd

6. The coolant purification system capacity was increased in 1969 from 180 kg/min. (23,800 LB/HR) to 1360 kg/min (179,500 LB/HR). This is an increase from 0.1% to 0.76% of the total coolant flow of 180,000 kg/min.
7. During the period 1969 to 1971, the following items were observed:
 - A. A major contributor to the radiation field caused by corrosion products is Co^{60} ; other contributors include Co^{58} , Fe^{59} , and Mn^{54} .
 - B. The corrosion products tend to be the result of oxidation, and behave as a thin, adherent film on the inside of the pressure vessels and piping of the primary coolant system.
 - C. Oxygen tends to be present in the primary coolant due to radiolytic action of neutron flow and the oxygen produces oxidizing conditions and the formation of corrosion products. The addition of hydrogen with a residual excess of 2 to 5 cm^3/kg , produces reducing conditions in the primary coolant.

1/29

1. The first step in the process is to identify the problem or issue that needs to be addressed. This involves gathering information and understanding the context of the situation.

D. When the condition of the coolant is deliberately cycled from oxidizing to reducing, and back to oxidizing, corrosion products are loosened from the film and placed in suspension in the primary coolant; in this manner, the corrosion products can be removed by the purification system, for each redox cycle. The total decontamination will be significant after several cycles.

8. During the period from October 1971 to April 1972, there were 14 outages, seven of which were devoted entirely to decontamination and the remaining seven were primarily for maintenance. The results of decontamination are shown in Fig. 1. The percentages of the metals in the corrosion products are shown in Table 1; these metals include a large percent of nickel, copper and iron. Examination of corrosion test coupons taken from the primary coolant circuit showed that no significant corrosion occurred during decontamination. During the 14 decontamination cycles, 500 to 800 Curies of Co^{60} were removed from the primary coolant system.

FIG. 1 - Boiler room S/D radiation fields and Curies Co⁶⁰ removed at S/D.

TABLE 1 - PERCENTAGE OF TOTAL METALS IN DPGS CRUD

Plant Condition	Crud Conc., mg/kg	Fe	Cu	Ni	Co	Cr	Mn
Operating	0.005-0.020	50	25	24	0.1	0.1	0.4
Shutdown	1.50	20	40	39	0.15	0.1	0.4

IV. Procedure for Decontamination of Douglas Point Station

1. Oxidizing conditions in the coolant is established
4 to 6 hours before the reactor is to be shut down, by adding
1 kg O₂ at a time, at hourly intervals, until 50 to 100 μ g
O₂/kg persists for 30 minutes after the addition.
2. The reactor is then shut down, and purification flow increased
to the maximum.
3. The boilers are hydraulically cycled.
4. The heat-transport system is temperature-cycled.
5. When the crud release subsides below a preestimated value
(dependent on the heat-transport system radiation levels),
then hydrogen is added to give 2 to 5 cm³ H₂/kg excess, and
Steps 3 and 4 are repeated.
6. When the crud subsides again, a change to oxidizing conditions
is made.
7. Redox, temperature, and hydraulic cycling are continued
until three hours before the reactor power is to be raised.
Then sufficient hydrogen is added to give a 5 to 10 cm³/kg
excess.
8. All the released crud is removed to below 100 to 200
 μ g/kg concentration, then the reactor power is raised.

I Plant Description

Power Rating - 70 MWt

Type - Plutonium Recycle Test Reactor
Pressurized Heavy Water Reactor

Location: Richland, Washington

Fuel: Natural Uranium

Moderator and Coolant: Heavy Water (D₂O)

Cooling Loop: Cycle uses primary and secondary loops
like PWR.

Steam Generator: Pressure tube Heat Exchanger pressurized
by Helium

Reactor: Pressure tube

Info from:

Nucleonics 1958 No. 1 pg. 62

Nucleonics 1967 No. 4 pg. 58

Decontam. of Nuc. Reactors & Equip - JA Ayres

II Heat Transfer System Data and Principal Construction Material

Coolant Inventory	3800 gal (33,300 lb)
Total Coolant Flow	8400 gpm (4.2×10^6 lb/hr)
Temp. Reactor Out	530 F
Temp. Reactor In	480 F
Pressure	1050 psi

Major Materials

Heat Exchangers & piping	AISI 304 stainless steel
In core pressure tubes	Zircalloy -2
In core calandria	Aluminum

Decontamination Experience - 1962

The first PRTR decontamination followed a fuel cladding failure in August 1962. Drawing the D₂O from the primary system through ion exchanger resin effectively removed suspended and dissolved material, but most of the activity remained in the primary loop in sludge accumulations or deposit on the walls of piping and equipment. Deposit was essentially magnetic iron oxide with occluded nuclides.

As a result of laboratory and pilot plant trials, the alkaline permanganate - oxalic acid stage was modified to include two components for plutonium core material removal. Corrosion tests were conducted on the various materials which could be exposed to the cleaning agents. As a result the following steps were taken:

- 1) Primary pump seals were protected by backflushing with pure water during the decon process.
- 2) Flow instrumentation, containing copper alloys and cadmium plate were valved out of the decon system.
- 3) Insure decon agents are thoroughly removed because subsequent reactor operations at high temperature would result in cracking of stainless steel.

Preparations for the decontamination included procedural instruction, comprehensive schedule and a rehearsal using deionized water instead of chemicals. Decon system schematic is shown in figure 1.

During the decontamination, precipitation and redeposition occurred during the oxalic acid phase because the allowable residence time was exceeded. The procedure was modified to redissolve the deposit by adding a permanganate step followed by ammonium citrate-EDTA. Even then there was slight reprecipitation and wool filters were utilized to collect the precipitate.

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

1125

Decontamination Experience Cont'd

The following table is an indication of the effectiveness of the cleaning. Several locations were monitored just before the decontamination and after flushing and draining.

Location	Before	After	D.F.
<u> </u>	<u>mR/hr</u>	<u>mR/hr</u>	<u> </u>
Volute Pump 1	13	3.5	3.7
Volute Pump 2	17	4.5	3.8
Volute Pump 3	32	5.5	5.8
Inlet Ht. Exchgr.	90	27	3.3
Outlet Ht. Exchgr.	35	6	5.8

FIGURE 1

FIGURE 7

Decontamination Experience - 1965

After the 1962 decontamination, radiation build up increased and the increase during 1965 reached levels which affected operations and maintenance. The increased buildup was found to be the result of primary pump seal system wear products from the Stellite surfaces. This provided cobalt for activation in the reactor core. Co^{60} was subsequently deposited throughout the primary system. By 1965, radiation had reached 80 mR/hr to 300 mR/hr at about 500 EFPD. Max. reading on steam generator tubes was 4500 m R/hr.

The decontamination of 1965 was performed during an outage resulting from rupture debris release in the Fuel Element Rupture Test Facility, an independent loop in the PRTR system.

The decontamination method used was different from the APOX used in 1962. Precipitated oxalate carried some of the activity and had to be removed by an additional cycle of dibasic ammonium citrate. Later development work showed that a satisfactory decom. could be accomplished by CITROX, a combination of citrate-oxalate reagent. Accordingly the AP-CITROX method was applied for the 1965 decom. the solutions were prepared at 120% of specified strength to allow for dilution during transfer and use based on the following specification:

NaOH	5%
KMnO ₄	3%
oxalic	2.5%
Amm. Citrate	5.0%
Fe(SO ₄) ₃	0.2%
diethylthiourea	0.1%

0445 :

100.

5

0445

0445

Prior to conducting the cleaning, the fuel elements were removed and the D₂O was drained from the primary system. Safety precautions were made as follows:

1. Flow-sensing lines, primary system valve packings, primary pump seals were protected from decon agents by supplying back pressure of demineralized water.
2. Secondary side of primary heat exchanger and the core blanket system were protected by nitrogen blanketing.
3. Gasometer and related piping and heat exchanger were pressurized with Helium. Helium pressurization system was isolated from the primary loop pressurizer.
4. Temporary sight glass installed on storage tank.
5. Instruments disconnected, lines capped. All dead legs equipped with drains.
6. Intensive training program

The decontamination procedure consisted of the following steps:

1. Alkaline permanganate circulated for two (2) hours at 105°C, drain and rinse with demin. water to pH 11.
2. Citrox circulated for two (2) hours at 80°C, drain and rinse with demineralized water to 1000 ohm-cm. Then circulate demin. water using ion exchange clean-up to specific resistance of 10⁶ ohm-cm. Adjust pH to 10-10.5 with LiOH.

301

0

8

The results of the decontamination process are shown in table 1 giving the radiation levels at various locations before and after the job.

Analysis of decon efficiencies shown in table 2 indicates the quantity of activity removed. About 30 curies of radionuclides had been removed.

Corrosion probes were utilized during the decontamination, and the results are shown in table 3.

3

1

220

THE UNIVERSITY OF CHICAGO

— १५५ —

六

PRTR-2

Table 1

**Radiation Level Changes on Primary System Resulting
from Decontamination^a**

System Location	Before Decontami- nation (mR/hr)	After Decontami- nation (mR/hr)	DF
Inlet to lower ring header	300	10	30
Volute, primary pump—2	150	5	30
Volute, primary pump—3	120	5	24
Pressurizer inlet piping	120	10	12
Inlet to degasser	100	5	20
Bottom of rupture detection heat exchanger header	190	5	38
Inlet header, injection pumps	60	5	12
Outlet header, injection pumps	100	5	20
Pressurizer outlet piping	190	5	38
Header to primary pump No. 2	220	5	44
Inlet to primary pump No. 1	210	5	42
Inlet to lower ring header	300	10	30
By-pass around primary heat exchanger and pressurizer	100	5	20
Piping from primary heat exchanger to pressurizer	110	5	22
Primary storage tank (5 points)	95	5	19
Pressurizer (3 points)	100	7	14
Secondary side of primary heat exchanger:			
6 in. inside manhole	1200	5	240
12 in. from tubes	1800	5	360
Maximum reading on tubes	4500	25	180

1/25

Table 2

Radiochemical Data in Decontamination Effluents^a

Isotopes	AP ^b	Radioactivity Level ($\mu\text{Ci/liter}$)		
		Citrox-1 ^c	Citrox-2 ^d	Citrox-3 ^e
Total β activity	5.8	530	860	880
Isotopic γ activity				
⁵⁴ Mn	0.013	--	--	--
⁶⁵ Zn	1.80	--	--	--
¹³⁷ Cs	18.4	--	--	--
¹²⁴ Sb	0.62	3.1	6.4	9.3
⁶⁰ Co	--	770	1100	1200

^a Taken from Ref. 38.^b Alkaline permanganate at pumpout (2 mR/hr).^c Citrox at start of circulation, temp., 60°C. (70 mR/hr).^d Citrox after 1 hr of circulation, temp., 70°C. (130 mR/hr).^e Citrox after 2 hr of circulation, temp., 77°C. (150 mR/hr.).

Table 3

Total Corrosion Measured by Samples and Probes During PRTR
Primary System Decontamination³⁸

Type of Material	Total Corrosion	
	(mg/cm ²)	Equivalent Penetration, (mils)
A. Coupons		
304 SS, sensitized	0.02	0.001
304 SS	0.03	0.0015
316 SS, sensitized	0.10	0.005
316 SS	0.02	0.001
416 SS, heat-treated	1.30	0.065
416 SS	0.32	0.016
440-C SS, heat-treated	8.12	0.406
440-C SS	33.4	1.67
302 SS, sensitized	0.28	0.014
302 SS	0.08	0.004
A-215 CS	30.4	1.52
A-106 Grade B CS	35.0	1.75
Zircaloy-2	"	"
304 SS with Stellite overlay	0.12	0.006
17-4 PH SS	0.4	0.02
Inconel-600	6.2	0.31
Inconel-X	3.2	0.16
B. Probes		
304 SS	"	"
1010 CS	16.0	0.8
410 SS	11.0	0.55

^a Not measurable.

1125

1125

50 75) (2)

(2)(3)(19)(20)

Decontamination of the Shippingport Atomic Power Station*

I. Plant Description

Power Rating:	231 MWt, 68MWe
Type:	Pressurized water reactor
Location:	Shippingport, Pa.
Fuel:	Uranium Oxide
Moderator & Coolant:	Water pH 9.5 - 10.5 with LiOH
Cooling Loop:	Primary and secondary loops
Steam Generator:	Pressure tube heat exchanger pressurized by hydrogen
Reactor:	Pressure Vessel

II. Heat Transfer System Data and Principal Construction Material

Coolant Inventory:	19,100 gal (2500 CF)
Total Coolant Flow:	22.86×10^6 lb/hr (3 loop)
Temperature Reactor Out:	538 F
Temperature Reactor In:	508 F
Pressure:	2000 psi

*Info. from:

WAPD 299 Decontamination of the Shippingport Atomic Power
 Station
ASTM, PT 235 - Symposium on Radioactivity in Industrial Water and
 Industrial Waste Water

The Shippingport Pressurized Water Reactor - Addison-Wesley
Publishing Co.
Decontamination of Nuclear Reactors & Equipment - J.A. Ayres

RECEIVED - [illegible] - [illegible] - [illegible]

RECEIVED - [illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

Decontamination of the Shippingport Atomic Power Station

Major Materials

Heat Exchanger & piping:	AISI 304 stnls steel
In core tubes:	Zircaloy 2

Decontamination Experience

In preparation for reactor modification work, the Shippingport PWR was decontaminated after 28,000 EFPH. The average radiation level had reached 80 mR, ranging from 14 to 625 mR. Highest values were the steam generator inlets and the hot legs between the boiler and reactor chambers. Activity was attributed to corrosion product activation within the core and subsequent redistribution throughout the main coolant system. The major contributor was Co^{60} ; minor contributors were Co^{58} , Fe^{59} , Mn^{54} , Cr^{51} , and Hf^{181} .

The decontamination method selected was the alkaline permanganate-diammonium citrate process, but more dilute solutions were used, which permitted the use of ion exchange resin for waste treatment. Prior to the decontamination, a mock run was performed simulating all steps of the process. Decontamination consisted of the following stages:

1. Recirculation of a 1.3% solution of alkaline permanganate at 250 F for 24 hours.
2. Process spent solution through deionizers after cooling, utilizing effluent for rinsing the reactor coolant system. Flush deadlegs.

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

SECRET

1/25

Decontamination Experience (Cont'd)

3. Recirculate a 1.3% solution of ammonium citrate at 250 F for 28 hours.
4. Process spent solution through deionizers after cooling, utilizing effluent for rinsing the reactor coolant system. Flush deadlegs.

Decontaminating solutions were admitted to the reactor coolant by use of a charging pump. The reactor coolant pumps operating at slow speed maintained circulation of 520 psig and 250 F. A charging pump furnished primary makeup water for continuous purging of mechanisms as well as the main coolant pump. A diagram of the decontamination system is shown in figure 1. The system contained 15 demineralizers for deionizing the waste permanganate and waste citrate solutions. The demineralizers were operated in series but did not provide the anticipated capacity. The waste treatment phase was therefore time consuming as well as requiring 850 CF of resin for the permanganate phase and 990 CF of resin for the citrate phase.

Table 1 indicates the effectiveness of the decontamination and shows an average decon. factor of 50, quite effective for the pipe walls and steam generators. But there were some areas where undissolved material collected.

Decontamination Experience (Cont'd)

These areas, as high as 17R/hr had to be decontaminated locally by flushing, removal of the component or rendered safe by shielding. This experience points out that decontamination methods should completely dissolve material and form no insoluble material that will accumulate in dead areas.

CJ

...for the purpose of ...
...the ...
...the ...
...the ...
...the ...

...the ...

...the ...

...the ...

Figure 1

Diagram of Decon System

Table 1
SHIPPINGPORT--DECONTAMINATION
SUMMARY OF AVERAGE EXTERNAL DOSE RATES

	Dose Rates Before Decontamination*				
	Loop A	Loop B	Loop C	Loop D	Average
Pipe Walls	183	308	235	195	232
Steam Generators	396	276	274	258	301

Dose Rates After Decontamination*					
Loop A	Loop B	Loop C	Loop D	Average	DF
6.2	6.2	3.6	2.8	4.7	49
4.7	9.6	6.7	3.8	6.2	49

*All radiation levels are given in mr/hr as measured with a shielded probe survey instrument.

I Plant Description

Power Rating - 3290 MW_t

Type - Plutonium Production Reactor

Location - Richland, Washington

Fuel - Natural Uranium

Moderator - Graphite

Coolant - Light Water

Cooling Loop - Primary and Secondary loops like PWR.

Steam Generator - Pressure tube heat exchanger

Reactor - Pressure tube

* Info from:

1. Decontamination of N-Reactor stainless steel generators
by W D Bainard - NACE 1969 Houston Texas, Conf.
2. Decontamination of Nuclear Reactors and Equipment - J A Ayres

II Heat Transfer System Data and Principal Construction Material

Coolant Inventory - 3700 gal (per 2 generators)

Total Coolant Flow - 200,000 gpm

Temp. Reactor Out - 540 F

Temp. Reactor In - 430 F

Pressure - 1760 psi

Major Materials

Heat exchanger piping - stainless steel 304

In core pressure tubes - Zircalloy -2

Reactor piping - carbon steel

Decontamination Experience

In order to perform maintenance on the N-Reactor it was necessary to decontaminate to allow economical use of manpower. Other steam generators of the twelve (12) which make up the N-Reactor system required decontamination in 1967 and 1968 to allow retubing operation.

The process chosen for the 1965 decontamination was a 3 step procedure : recirculating a solution of alkaline permanganate at 105⁰C for 2 hours, a dilute (.12%) inhibited sulfamic acid rinse for 30 min., and finally a 3 hour recirculation of 8% proprietary inhibited sulfamic acid at 80⁰C. There were delays and dilution of reagent which may have contributed to an unsatisfactory decon performance. Figures 1 & 2 show the decontamination recirculation diagram and the reagent feed system which were used for these decontaminations.

The 1967 decontamination method was selected on the basis of laboratory tests showing that the proprietary ammoniated citric and oxalic acids was superior to sulfamic acid. In addition a preliminary acid step was applied to remove carbon steel corrosion deposit overlay that might reduce the effectiveness of the permanganate stage if not eliminated. Again there were difficulties in applying the specified strength reagents, but despite this, the decontamination was successful.

The third decontamination, conducted in 1968, utilized the same chemicals but added a dilute acid rinse after the permanganate stage to insure an effective strong acid stage which followed.

Decontamination Experience Cont'd

The experience gained with each successive decontamination is reflected in the decon record for each effort:

<u>Date</u>	<u>Before</u> <u>mR/hr</u>	<u>After</u> <u>mR/hr</u>	<u>Method</u>	<u>Decon Factor</u>
1965	2000	1000	AP-Sulfamic	2
1967	2500	10	AP-Citrox	250
1968	1100	3	AP-Citrox _w - Citrox _s	370

Figure 1 Modified piping arrangement.

Figure 12 N Reactor decontamination system flow diagram.

Decontamination Experience

Previous extensive laboratory work had indicated that inhibited phosphoric acid would effectively decontaminate carbon steel systems with acceptable corrosion rates provided the solution was applied in a continuous single pass, rather than by recirculation. Further testing using a proprietary inhibited phosphoric acid indicated that critical reactor components which would be exposed to the solvent were not damaged. Corrosion testing showed greatest rates were on carbon steel adjacent to a carbon steel - stainless steel weld; a penetration of about 0.06 mils could be expected from a 20 minute decontamination.

The N reactor carbon steel piping was decontaminated twice in preparation for maintenance work, in October 1967 and in July 1968. On both occasions phosphoric acid was used. The first cleaning consisted of a 16 minute injection of 10.5% phosphoric acid at 160 - 180 F, followed by a demineralized water rinse until the pH increased to 6. Following this the reactor was returned to recirculating cooling with ammoniated water, pH 10, and system feed & bleed was established until system parameters returned to normal. The second decontamination was conducted utilizing inhibited phosphoric acid, but the concentration was reduced to 8%, the exposure time increased to 22 minutes at an increased temperature of 185 F. table 1 gives radiation readings

11/27

11/27

11/27

11/27

11/27

at eight locations before and after the two decontaminations. Figure 2 gives fluid analyses during the various stages of the 1967 decontamination process. It can be seen that rinsing was not complete since activities increased when coolant recirculation commenced. Corrosion coupons were utilized during the 1967 decontamination and the results are shown in Table 3.

The cleaning was fairly effective, but there was localized pitting of carbon steel at welds to stainless steel as well as redeposition of radio-nuclides in the downstream portion of the piping.

These cleaning have continued on practically a yearly basis. The 1975 decontamination was accomplished by a single pass application of a 9% solution of inhibited Phosphoric acid (TURCO 4512-A) or 173⁰F for 23 minutes.

1/25

11

11

11

11

Table 1

-N Reactor Front and Rear Elevator Radiation Readings, mr/hr

Date	Front Elevator				Rear Elevator			
	Center	Apron	Nozzles	Connectors	Center	Apron	Nozzles	Connectors
10-5-67 Before Decontamination on 10-10-67	140	200	290	1000	110	210	440	280
10-19-67 After Decontamination on 10-10-67	12	24	33	20	32	80	150	50
Decontamination Factor	12	8	9	50	3	3	3	6
7-21-68 Before Decontamination on 7-22-68	47	82	74	310	93	200	250	320
8-9-68 After Decontamination on 7-22-68	10	20	30	26	19	48	110	58
Decontamination Factor	5	4	2	12	5	4	2	5

Table 2

Isotopic Analysis of Solution Samples, Average of Four Tubes
1967 Decontamination
(micro-curies per liter)

Solution	Time from Start of Acid Injection (minutes)	Cr-51	Mn-54	Co-58	Fe-59	Co-60	ZrNb-95	Ru-103	BaLa-140	Ce-141	CePr-144
Recirculating coolant	-21	0.01	0.01	0.01	0.02	0.03	0.09	0.06	0.01	0.05	0.06
Single Pass Water	-2	0.13	0.01	—	0.05	—	0.02	0.04	0.27	0.01	0.42
Acid	2	52	41	18	370	690	25	30	0.19	150	480
Acid	6	36	16	2.6	30	750	8.7	15	3.1	3.5	740
Acid	10	17	5.3	3.3	9.3	310	2.7	0.53	15	0.77	1.3
Acid	14	15	4.4	2.9	3.2	170	0.95	0.52	0.46	0.83	0.39
Single Pass Water Rinse	26	0.02	0.20	0.01	0.36	0.31	0.03	0.01	0.27	0.01	0.05
Single Pass Water Rinse	36	0.01	0.03	—	0.07	0.04	0.01	—	—	—	—
Recirculating Coolant	54	2.3	2.3	0.51	0.36	5.1	2.1	0.91	3.1	0.94	6.6
Recirculating Coolant	64	2.1	1.3	0.20	0.74	4.7	2.8	0.54	1.9	1.2	7.3
Recirculating Coolant	69	2.5	1.2	0.35	—	4.0	2.9	1.6	0.57	0.81	8.7

Table 3

TABLE 8 - Corrosion During 1967 Through-Reactor Decontamination

Sample	Tube 2547	Tube 1147	Tube 2565
A245 Carbon Steel (CS) to	0.006	0.033	0.035
A245CS Crevice Coupons, mils	0.069	0.036	0.048
	0.033	0.027	0.045
	0.016	0.038	0.037
A245CS Non-crevice Coupons, mils	0.022	0.032	0.041
	0.041	0.034	0.042
A212CS Welded to 304 Stainless	0.014	0.0164	0.0157
Steel (SS), mils	0.023	0.0166	0.0125
AMS 5616SS, mils	0.016	0.0102	0.007
	0.011	0.0085	0.010
304 SS, mils	0.014	0.002	0.0022
	0.009	0.0024	0.004
Zr-2, Autoclaved, Crevice, mg/dm ²	-2.7	+1.3	+2.7
	-7.7	-0.9	-1.8
Zr-2, Etched, Crevice, mg/dm ²	-2.9	+5.4	+0.9
	-5.0	+2.3	+0.9
Zr-2, Autoclaved, mg/dm ²	-4.1	+0.4	+2.7
	-7.2	+1.8	-0.9
Zr-2, Etched, mg/dm ²	-3.2	0.0	+0.4
	-0.4	+4.1	-0.9
Zr-2 - A245CS Coupled ⁽¹⁾			
a) Zr-2, Autoclaved, mg/dm ²	-4.1	-1.3	+2.7
b) A245, mils	0.021	0.033	0.043
c) Zr-2, Autoclaved, mg/dm ²	-5.9	-0.9	-2.3
d) A245, mils	0.029	0.043	0.038
e) Zr-2, Etched, mg/dm ²	-3.2	+0.9	+5.4
f) A245, mils	0.0201	0.049	0.040
g) Zr-2, Etched, mg/dm ²	-3.6	+2.3	+3.2
h) A245, mils	0.0397	0.0323	0.061

⁽¹⁾ Couples are: a to b, c to d, e to f, g to h.

The weight changes on the Zr-2 samples were very small and the negative changes probably represent slight mechanical damage to the sample caused by the sample spacers. Slight surface etching occurred on some of the carbon steel samples and numerous, tiny (1 mil dia) pits occurred on the carbon steel next to the weld on some of the A212-304 weld coupons. All the other samples were in excellent condition.

Decontamination of SENA PWR Primary System* (6)

I. Plant Description

Power Rating - 825 mwt

Type - Electric Production

Location - Chooze, France

Fuel - Uranium Oxide

Moderator & Coolant - Light water

Cooling Loop - Primary & Secondary loops

Steam Generator - Pressure tube heat exchanger pressurized
by hydrogen

Reactor - Pressure vessel

II. Heat Transfer System Data and Principal Construction Material

Coolant Inventory - 350 CF/loop isolated (2620 gallons)

Total Coolant Flow - 11×10^6 lbs/hr.

Temp. Reactor Out - 577°F

Temp. Reactor In - 508°F

Pressure - 2200 psi

Major Materials

Heat Exchanger & Piping - 300 series Stainless

Fuel Cladding - 304 Stainless Steel

* Info from: American Power Conference Proceedings 1971

125

Decontamination Experience

In 1968, after 2,400 EFPH, core support fragments caused damage to the steam generators and decontamination was required for access. Radiation levels were found to be 1800 mR/hr. max. The decontamination method chosen for the SENA plant was essentially the permanganate/oxalic-citrate process.

The initial plant design provided a bypass line in each loop to allow circulation through each isolated loop. Thus, solution inventory requirements were reduced, allowing utilization of plant equipment for filling and circulation of decontamination solutions. The required solution temperature was obtained by operation of the main coolant pump which was used to circulate the decon solution.

Review of plant materials and corrosion tests indicated four vulnerable areas:

1. Pressurization by nitrogen of the secondary side of the steam generators was not required because a hydrostatic test indicated leak tight heat exchangers.
2. The bearings of the main coolant pumps are composed primarily of graphitar which is attacked by the decontamination solutions. A purge flow of demineralized water was provided to avoid this exposure.
3. The gaskets of the charging pumps are mainly carbon and would be affected by the decon solutions, but

Decontamination Experience (continued)

since they are easy to replace they were not protected.

4. The loop isolation valves required establishing water pressure inside the body of the valve to insure leak-free seating of the valve discs.

Figure one is a diagram of the primary system and the equipment utilized for preparing and admitting decontamination agents to the system. Also shown are the facilities for maintaining purge flows and back pressure on the isolation valves. Preliminary tests were performed to establish that equipment to be used for decontamination was operating properly, to confirm the validity of the procedure, and to train operators.

The procedure consisted of a pre-heat stage followed by injection of the first decon agent and circulation and finally three rinses. The 3 steps were repeated using the second decon agent. The total time required was 78 hours for each loop (minimum). Twelve men were required to effectively conduct the decontamination.

At SENA the major activities in the steam generators was associated with activated corrosion products, consisting of oxides of iron, chromium, and nickel. The following table indicates the results of the decontamination on the four loops:

Steam Gen.	<u>Hot leg radiation mR/hr</u>			
	Tube Sheet		Partition Plate	
	<u>Before Decon.</u>	<u>After</u>	<u>Before Decon.</u>	<u>After</u>
1	700	50	350	15
2	750	50	400	25
3	450	5	300	3
4	450	3	320	3

<u>Cold leg radiation mR/hr</u>				
1	600	3	300	7
2	650	4	300	8
3	400	3	300	3
4	400	3	300	3

The low DF's for the numbers 1 & 2 generator's inlets are attributed to the presence of core barrel bolt metal.

1/25

R

100

SPR. ON BIR

T

E

FLOW DIAGRAM USED FOR DECONTAMINATION OF A REACTOR COOLANT LOOP

Decontamination of SGHWR*

Plant Description

Power Rating - 100 MWe

Type - Pressure tubed boiling water reactor

Location - United Kingdom, Dorchester, Dorset

Fuel - Enriched Uranium, Zircaloy 2 Cladding

Moderator - Separate low pressure heavy water system

Coolant - Light water, neutral (no additives)

Materials - AISI 321 stainless steel, carbon steel

Coolant Flow - 1.2×10^6 lb/hr

Decon Experience

There have been eight decontaminations performed on the SGHWR. This account concerns those performed in 1969 and 1971. The major component of the deposits to be removed was iron oxide with minor amounts of copper, nickel, chromium & zinc. Major nuclides were cobalt 60, zinc 65, and chromium 51.

The alkaline permanganate step of the AP-CITROX method was omitted because it was feared that there would be channel seal leakage allowing exposure of the aluminum calandria to the solvent. Tests on primary riser piping sections indicated that Turco 4521, CITROX, was adequate by itself to accomplish decontamination. It was effective only in the temperature range of 75-85 C. The materials most sensitive to corrosion were AISI stainless iron and mild steel, but an exposure rate of 12 hours was tolerable for a corrosion allowance of 50 microns.

*Info from Ohio State 1975 Seminar and private communication UKAEA

10-10-68

*Info from Ohio as of 12/22/52. Every effort will be made to obtain information from other sources.

STUDYING WITH ME : C.T.

10267-10271

Decontamination of SGHWR - 1969/1971 (cont'd)

Despite procedural difficulties, which necessitated two cycles of CITROX, the south circuit was decontaminated effectively in August 1969. There was evidence of redeposition in the lower parts of the circuit, possibly as a result of overheating. In addition, the corrosion allowance was exhausted earlier than expected (4 hours) during the second cycle and the decontamination was terminated earlier than planned.

The north circuit was decontaminated in April 1971 and the cycle was limited to two hours because of the reduction in the corrosion allowance. A second cycle was started but was stopped at one and one half hours when purge flow to a circulator was lost. Despite the difficulties, the decontaminations were effective. The decontamination improvement factors are given below:

<u>Location</u>	<u>Improvement Factor</u>	
	<u>North</u>	<u>South</u>
Exterior of steam drums		
outer side	17	15
inner side	21	13
Exterior of level measuring vessel	11	15
Centre of charge face	2.0	1.3
Between charge face & drum	4.4	4.4
Outside upper lagging box	3.5-9.0	4.0-16
Inside upper lagging box	11-19	11-39

TO: SAC, NEW YORK
FROM: SAC, NEW YORK

RE:

NY 100-100000

NY 100-100000

CC-

100

Decontamination of SGHWR - 1969/1971 (cont'd)

Corrosion coupon results are listed below:

	<u>Microns</u>	
	<u>N</u>	<u>S</u>
AISI 410 (rolled joint mat'l)	25.50	75-125
Mild steel	25	50
Stainless steel	2	2
Stainless steel/mildsteel welds	-	125
Stainless steel/stainless steel welds	-	2
Stainless steel/AISI 410 welds	-	100
Mild steel/mild steel welds	-	75-125

Subsequent inspection showed the fuel elements to be cleaned to the zinc oxide layer. The stainless steel parts were clean. The steam drums were clean. Stainless steel pipe work was found to be free from oxide film and dose rates were less by a factor of 20.

APPENDIX B

IP1 DESCRIPTION AND MATERIALS

APPENDIX B

INDIAN POINT 1 DESCRIPTION AND MATERIALS

This appendix consists of a process description of Indian Point 1, a physical description of the existing components which would be utilized during decontamination, and a complete list of materials for each component.

Process Description

Indian Point No. 1 is a 615 MWt (265 MWe) Babcock & Wilcox PWR in which the nuclear fuel is cooled by a stream of hot water kept under pressure sufficient to prevent boiling. This hot primary water is used to generate steam from a secondary water circuit in an indirect heat exchanger or boiler. The steam is then used to drive a conventional steam turbine-generator.

Figure 1 shows a flow diagram of Indian Point 1. Water at the rate of about 53 million pounds per hour is pumped by means of eight primary coolant pumps through the core. These pumps are the canned type in which the pump impeller and driving electric motor rotor are enclosed in a stainless steel can. Bearings are within the can and are lubricated by the water being pumped. Stator windings are outside of the can. Primary water is heated about 30 F by the reactor heat generated. The heat is transferred from the primary water to the secondary water circuit in four heat exchangers.

The water in the primary system is kept at a pressure of 1500 psi by means of a steam bubble generated by electric heaters in an elevated pressurizer. This pressure insures that the primary water will not boil within the core.

The steam generated in the secondary circuit is 405 psi saturated steam with a temperature of about 450 F. The steam is superheated in twin oil fired superheaters to a temperature of 1000 F.

Figure 2 shows the flow in one the four boiler loops. About 32,000 gpm of hot primary water is pumped through each of the steam generators and returned to the bottom of the reactor by two primary coolant pumps in each loop. Each loop has a pair of stainless steel stop and check valves to permit isolation of a loop.

Figure 3 is a cross section of the nuclear portion of Indian Point No. 1. The pressure vessel surrounding the core, the four boilers, primary pumps and valves are enclosed in a carbon steel sphere 160 feet in diameter with 1 1/2 inch thick walls. The sphere is large enough to contain the entire water inventory in the form of steam. The sphere is shielded by a concret enclosure with six foot walls. Its purpose is to absorb gamma radiation during operation.

Figure 4 is a sectional elevation of the entire plant, and figure 5, a plan of the station including the new waste evaporators.

The reactor pressure vessel is shown in Figure 6. It contains a core of 120 fuel bundles, each having 195 stainless steel clad rods. To increase generation, rods are pushed upward. The hafnium control rods emerge from the core, the space being filled by follower rods of zircaloy.

The primary blowdown purification system is connected to loop #12. Figure 7 is a diagram showing its main components.

The purification process consists of pumping primary coolant blow-down water by means of purification transfer pumps through the purification inlet filters. The water is then cooled to approximately 120 F using purification inlet coolers and then passed through cation and mixed bed ion exchangers. Finally, the water passes through a purification outlet filter to a clean water storage tank.

Discharge from the inlet filters is normally directed to the coolers as outlined above, however, up to 40,000 lbs/hr may be processed through a gas stripper which operates in a side stream. The stripper removes dissolved gases such as air, fission gases, and hydrogen, and should be useful during the redox demonstrations. The stripper gas discharge goes to the Waste Gas System for further treatment. The stripped effluent is pumped by the purification stripper discharge pumps No. 11 and No. 12 through the purification inlet coolers and then to the ion exchangers. Water is processed through the gas stripper only when air is present, fission gases require removal, or when hydrogen must be removed because of shutdown for major repair of for refueling.

The three transfer pumps which take their suction from the blowdown storage tanks and pump the water through the purification system are arranged in parallel, each having a capacity of 110 gpm at 240 feet TDH. The normal purification rate of 36,000 lb/hr will be satisfied with one pump. These pumps may also be used to pump to the waste collection tanks of the Liquid Waste Disposal System.

The eight inlet filters operated in parallel receive water from the transfer pumps and effectively filter 5 micron particles. Each new filter passes 50 gpm with a 2 psi pressure drop and two filters are normally in operation. A bypass around the filters is provided. Fouled filters may be removed and replaced while the purification system is in service.

Before the water is processed through the ion exchangers it is cooled to approximately 120 F in one of two inlet coolers because temperatures about 140 F cause the ion exchange resins to break down. The coolers are shell and tube heat exchangers with in-stream water flowing through the tubes. Cooling water comes from the Fresh Water Cooling System. The cooling capacity of each heat exchanger is 80,000 lbs/hr cooling the water from 200 F to 120 F.

After passing through the purification inlet coolers, the water is normally passed through one of three cation exchangers and one of three mixed bed exchangers. The cation exchangers are arranged in parallel as are the mixed bed exchangers and should it be required, all may be operated simultaneously, or either group may be bypassed.

When boric acid is present in the water and has to be removed the mixed bed exchangers are bypassed and after being processed through the cation exchangers the water is passed through two boric acid clean-up exchangers where the boron is removed. These anion exchangers may be operated singly, in parallel or in series when one of the exchangers is approaching exhaustion.

Following the ion exchange processing of the water, the flow is directed through one of two parallel connected purification outlet filters. Each filter will pass 160 gpm at a 2 psi pressure drop when clean. The effective filtration through each filter is 10 microns. These filters remove the small particles of resin and any other insoluble particles which carry over from the ion exchangers. The service hours obtained from these outlet filters should be considerable because of the small amount of carried over particles (fine content) expected due to the special grade resin used in the beds. The outlet filters if fouled are backwashed with water from the clean water storage tanks to the spent resin storage tanks.

The purified water will by-pass the clean water storage tanks after going through the outlet filters. Two clean water transfer pumps will be available for transferring decon solutions to the solvent storage tank for recirculation or to the Waste Disposal System.

Figure 8 is a diagram of the waste processing system. Plant liquid wastes from various sources are directed to waste collection tanks, total capacity 300,000 gallons.

Liquid waste is fed from hold up tanks to the waste evaporator through a preheater to increase the temperature to about 215 F. The heated feed then flows to the stripping column where radioactive gases and other volatiles are stripped off by steam counter flow. After stripping, feed is introduced into the evaporator shell. Feed rate is controlled by level in the evaporator. Heat for boiling

the feed solution in the evaporator shell is provided by steam condensing in a submerged tube bundle. The vapor produced in the evaporator is passed through a series of demisters and flexi trays in the absorption tower before entering the shell side of a condenser. Distillate from the condenser is then pumped through a distillate cooler to storage. The concentration level in the evaporator shell is maintained at a uniform level by continuous recirculation and when the concentrate reaches the prescribed strength, recycling ceases and the concentrate is pumped out to a holding tank.

Evaporator concentrates are directed to a waste packaging system.

The process used for solidification of liquid radioactive waste is a three part system controlled from one location by a control and monitoring system. Liquid waste from the evaporator concentrate holding tank is transferred at a controlled rate to the polymer mix chamber where polymer is introduced at a controlled rate. The waste-polymer steam is then directed to a mixing chamber where catalyst is introduced at a controlled rate. This occurs just prior to let down into the disposable container. The feed rates can be controlled as follows:

<u>Process feed</u>	<u>Range</u>
Liquid Waste	1 - 10 gpm
Polymer	1/2 - 5 gpm
Catalyst	5 - 50 gpm

Physical Description

The physical description of the equipment constituting the main components of the primary system follows:

Nuclear Reactor

The nuclear reactor vessel is a cylindrical shell of carbon steel with an i.d. of 9'9". The inside wall is lined with 0.109 inches of stainless steel. The general arrangement of the vessel and its internal structure is shown in Figure 1.

Each of the four inlet connections in the bottom hemispherical head and each of the four outlet connections in the upper part of the shell has a minimum i.d. of 20 1/16 inches. There are 21 penetrations in the bottom head for control rod nozzles. The top closure head is secured by fifty 5 1/2" diameter studs. The closure head and vessel are seal welded to a stainless steel membrane to prevent leakage.

Three alternating layers of stainless steel and water are arranged between the core and the pressure vessel wall to serve as thermal shields.

The internal components of the reactor consist of the following assemblies:

1. Control rod nozzle seal
2. Lower grid plate and plenum chamber
3. Core shroud
4. Hold-down columns
5. Upper grid plate and plenum chamber

Most of the internal components are fabricated from type 304 stainless steel. Inconel is used for special bolting and springs. Weld and nozzle claddings are stainless steel type 308.

Steam Generators

The steam boiler in each coolant loop consists of a U-tube lower heat exchanger drum, straight upper separator drum and connecting risers and downcomers. The tubes are 1" o.d. type 304 stainless steel and are rolled and welded into 308 stainless steel clad tube sheets. The inlet head contains a single 24" diameter primary inlet connection, 4 - 6" diameter hand holes, a 16" diameter manway. The outlet head contains 2 - 18" diameter primary outlet nozzles, 4 - 6" diameter hand holds, and a 16" diameter manway. The outlet head contains 2 - 18" diameter primary outlet nozzles, 4 - 6" diameter hand holes, and a 16" diameter manway. One inch diameter vent and drain connections are provided. Most of the major components are 304 stainless steel or 304 clad. Joints, seams and manways are 308 stainless clad.

Main Stop Valves

Two motor operated 24" gate valves in each loop serve to isolate the boiler, pumps and check valves of each loop from the reactor and the other loops of the primary coolant system. Valves are equipped with by-pass lines and glove valves for equalizing pressure. The gate valves and by-pass assemblies are mainly 304 stainless steel.

Check Valves

Each loop has an 18" stainless steel swing check valve downstream of each pump to prevent gross recirculation of flow through an idle pump. To permit natural circulation through the loop, the check valves allow a small forward flow with the disc in the vertical position. The check valves permit reverse flow to

provide for warming up of a cold loop. A 1/8 inch hole in the body or disc of the valve permits a small reverse flow.

Primary Pumps

These pumps are vertical, single-stage centrifugal pumps. Most of the pump internals and the pump casing are type 304 stainless steel. The main impeller is attached to the rotor shaft of the driving motor, whose stator and rotor are encased in inconel cans. The motor bearings are graphitar 14. (carbon & graphits)

Pressurizer

The pressurizer consists primarily of a pressurizer tank, heaters, surge line, spray line and spray head. The tank is stainless clad and the internals are stainless steel.

Primary Piping

The primary piping is type 304 stainless steel and the weld joints are type 308 stainless.

Blowdown Purification System

The equipment materials and sizing are described in their order in the purification train.

Blowdown Coolers

There are two blowdown coolers rated for 80,000 lb/hr each cooling the blowdown from 485 F to 200 F. The heat rating is 22.8×10^6 BTU/hr. Both tube and shell materials are 304 stainless steel since the coolant is primary makeup water.

Components - Purification System

The components of the primary purification system are described below:

Blowdown Storage Tanks

Two tanks with a capacity of 3000 CF each are constructed of 304 stainless steel.

Purification Transfer Pumpe (3)

Type - canned rotor, horizontal centrifugal

No. stages - one

Design capacity - 110 gpm, 240 ft. TDH

Material - stainless steel

Horsepower - 15

RPM - 3450

Purification Inlet Filters (8)

Casing design pressure - 150 psig

Casing design temperature - 250 F

Casing material - carbon steel

Capacity & pressure drop - 50 gpm, 2 psi (clean)

Effective filtration - 5 micron

Filtering medium - cellulose

Cleaning provision - dispose entire unit

Purification Inlet Coolers (2)

<u>Parameter</u>	<u>Tube Side</u>	<u>Shell Side</u>
Design pressure (psig)	150	200
Design temperature (F)	400	200
Flow (lbs/hr)	80,000	160,000
Inlet temperature (F)	200	105
Outlet temperature (F)	120	145
Material - stainless steel		
Duty - 6.4×10^6 BTU/hr.		
Effective surface - (ft. ²)	1310	

Purification Gas Stripper

Type - vertical, 3 stage, tray
Design pressure - 50 psig (internal)
14.7 psig
Design temperature - 300 F
Operating pressure - 10 psig
Operating temperature - 240 F
Flow rate - 40,000 lb/hr.
No. tray sections - 2
Total trays - 144
Material - stainless steel

Purification Stripper Discharge Pumps (2)

Type - Horizontal, centrifugal
No. of stages - one
Design capacity - 85 gpm, 245 ft. TDH
Material - stainless steel
Horsepower - 15
RPM - 3450

Purification Ion Exchangers (6)

Type - 3 cation, 3 mixed bed
Shell design pressure - 140 psig
Shell design temperature - 120 F
Shell material - stainless steel
Resin bed depth - 48"
Maximum flow rate - 53,000 lb/hr
Pressure drop (clean) - 10 psi
Specific rate - 6 gpm per ft.²

Resin type - Cation - Rohm & Haas XE77

Anion - Rohm & Haas XE150

Resin volume/unit - 63 2/3 ft.³

Maximum pressure drop - 35 psi

Anion Exchangers (2)

Shell design pressure - 130 psig

Shell design temperature - 120 F

Shell material - stainless steel

Resin Bed depth - 60 inches

Maximum flow rate - 40,000 lb/hr

Pressure drop (clean) - 10 psi

Specific rate - 6gpm/ft.²

Resin type - Rohm & Haas XE78

Resin volume/unit - .60 ft.³

Maximum Pressure drop - 45 psi

Purification Outlet Filters

Casing design pressure - 150 psig

Casing design temperature - 250 F

Casing material - stainless steel

Capacity & pressure drop - 160 gpm at 2 psi (clean)

Effective filtration - 10 micron

Filtering medium - sintered stainless steel

Cleaning provision - backwash to spend resin storage tanks.

Components - Waste Evaporator

Shell & Tube Bundle

Shell material - nickel - chrome alloy

Tube material - inconel 600

Tube surface area - 784 ft.²

Number & size of tube - 190, 3/4" o.d. - 20 BWG

Absorption Tower

Shell material - stainless steel

Demisters - stainless steel

"Bubble caps & trays - stainless steel

Evaporator Condenser

Shell material - stainless steel

Tube material - stainless steel

Tube surface area - 312 ft.²

Number & size of tubes - 140, 3/4" o.d. - 20 BWG

Vent Condenser

Shell material - stainless steel

Tube material - stainless steel

Tube surface area - 15 ft.²

Number & size tubes - 16, 1/2" o.d. - 20 BWG

Distillate Cooler

Shell material - carbon steel

Tube material - stainless steel

Tube surface area - 84 ft.²

Number & size tubes - 30, 1/2" o.d. - 20 BWG

Feed Preheater

Shell - carbon steel

Tube material - stainless steel

Tube surface area - 43 ft.²

Number & Size tubes 16, 1/2" o.d. - 20 BWG

Components - Packaging System

Polymer system

Storage tanks (5)

Volume - 450 gallons

Material - aluminum

Feed pump (1)

Capacity - 5 gpm

TDH - 92 ft.

Catalyst system

Storage tanks (1)

Volume - 100 gallons

Material - polypropylene

Feed pump (1)

Capacity - 60 gph

TDH - 312 ft.

Catalyst

Material - sodium bisulfate

Liquid waste pump (2)

Capacity - 10 gpm

TDH - 92 ft.

Material & Interface List

A complete list of materials is given on the following pages as prepared by Dow for ESEERCO.

1.3 Materials Lists

1.3.1 Primary Coolant System

1.3.1.1 REACTOR VESSEL & INTERNALS

Material Listing

SA-105 Gr II

SA-212 Gr B

A-213-55 Type 304

SA-240 Gr S

A-240-54 Gr S Fin 1(A)

A-240-55 Gr S Fin 1(A)

A-276-55 TP 304 Cond A Fin 2

AISI 304 SS

Type 308L

Inconel "X"

Stellite Alloy Haynes #25

1.3.1.2 PRESSURIZER MATERIAL LISTING

SA-106 GR C

AISI Type 304 SS

SA-182 GR F 304

MIL-S-18170 Type 304

SA-212 GR B

Type 308L

SA-213 Type 304

Type 316

SA-240 GR S

SA-312 Type 304

1.3.1.3 STEAM GENERATOR

Material Listing

SA-105 Grade II
SA-182 Type F 304
SA-212 Grade B
SA-213 Type 304
SA-240
Type 304 SS
Type 308L
Type 309 SS

1.3.1.4 PRIMARY COOLANT PUMP

Material Listing

SA 182 Gr F 304
SA 212 Type 304
SA 351 CF 8
AISI 304

1.3.1.5 PRIMARY STOP VALVE

Material Listing

A 182 Grade F 304
A 216 Grade WCB
A 276 Type 304
A 276 Type 304 Stellite 1-D
A 351 Grade CF 8
A 351 Grade CF 8 C
Type 304 SS
308L Type A SS weldment
Flexitallic Style R3-15

1.3.1.5 PRIMARY STOP VALVE (Continued)

Haynes Alloy #25 (AMS 5537)

Haynes Alloy #25 (AMS 5759)

Graphic Impregnated Asbestos (John Crane 187-1 Diagonal Cut)

Stellite #6

1.3.1.6 PRIMARY CHECK VALVE

Material Listing

A 182 Grade F 6

A 182 Grade 304L gold plated

A 182 Grade F 316

A 351 Grade CF 8

Type 308L

Type 416 SS

Inconel X-(Jan-W-562A Class 2)

Stellite #25

Stellite #6

1.3.2 PRIMARY RELIEF SYSTEM

Material Listing

SA 105 Gr 1

SA-106 Gr A

SA 212 Gr B

Type 304 SS

Type 308L

Type 316 SS

Carbon Steel

1.3.3 PRIMARY VENT SYSTEM

Material Listing

Type 304 SS

Type 308L SS

1.3.4 PRIMARY COOLANT BLOWDOWN SYSTEM

Material Listing

SA-167 Grade 3 Type 304

SA-376 Grade Type 304

Type 303 SS

Type 304 SS

Type 308L

Type 315 SS stellited

Type 316 SS

Stellited 316 SS

Type 416 SS

Chevron Packing

Dupont Teflon

Flexitallic Gasket

1.3.4.1 PRIMARY COOLANT BLOWDOWN VALVE

Material Listing

Type 303 SS

Chevron Packing

Type 304 SS

Flexitallic Gasket

Type 316 SS (stellited)

Teflon

Type 319 SS

Type 329 SS

Type 416 SS

1.3.4.2 PRIMARY COOLANT EMERGENCY BLOWDOWN ISOLATION VALVE

Material Listing

SA-212 Grade B

SA-212 Grade B (stellited Face)

Asbestos Packing

Carbon Steel

Exelloy

Parkersized Steel

Siliconized Steel

Soft Iron

1.3.5 COMPONENT DRAIN SYSTEM

Material Listing

A 182 Grade F316

AISI Type 316 SS

Chrome Plate MS 600

Stellite MS 481

1.3.6 DECAY HEAT SYSTEM

Material Listing

A-182 Grade F 316

AISI Type 316 SS

Chrome Plate MS 600

Stellite MS 481

1.3.7 PURIFICATION SYSTEM

Material Listing

A-105 Gr 2

SA-167 Grade 3 Type 304

SA 212 Gr B

1.3.7 PURIFICATION (Continued)

A 216 Grade WCB

A 216 Grade WCB (stellited MS-481)

SA-312 Gr TP 304

SA-367 Grade Type 304

SA-376 Grade Type 304

Type 303 SS

Type 304 SS

Type 308L

Type 316 SS

Armco Iron BHN 90 maximum

Carbon Steel

Dupont Viton Compound B

Evolloy 300 BHN (AISI 410)

Stellite AISI-C1120

Stellited NS-481

1.4 Material Interfaces

1.4.1 Primary Coolant System Interfaces

1.4.1.1 Pressurizer Material Interfaces

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
SA 106 Gr C	SA 182 GR F 304	Cladding
	Type 308L SS	Weldment
	Carbon Steel	Weldment
SA 182 GR F 304	SA 213 Type 304	Weldment
	AISI 304 SS	Weldment
	SA 312 Type 304	Weldment
	MIL-S-18170 Type 304	Weldment
	Type 308L SS	Weldment
	Carbon Steel	Weldment

1.4.1.1 Pressurizer Material Interfaces (Continued)

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
SA 212 GR B	SA 240 GR S	Cladding,
	AISI 304 SS	Compression
	Type 308L SS	Weldment
		Weldment
SA 213 Type 304	Type 308L SS	Weldment
SA 240 GR S	Type 303 SS	Weldment
	AISI Type 304 SS	Weldment
	ASME 312 Type 304 SS	Welded
	SA 312 Type 304 SS	Weldment
	Type 308L SS	Weldment
	Type 316 SS	Weldment
Type 303 SS	Type 308L SS	Weldment
Type 304 SS	AISI 304 SS	Weldment
	ASME 312 Type 304	Weldment
	Type 308L SS	Weldment
	ASME 312 Type 304	Welded
	Type 308L SS	Weldment
ASME 312 Type 304 SS	Type 308L SS	Weldment
SA 312 Type 304	Type 308L SS	Weldment
MIL-S-18170 Type 304	Type 308L SS	Weldment
Type 308L SS	Type 316 SS	Weldment
Type 316 SS		

1.4.1.2 Steam Generator Material Interfaces

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
SA 105 GR II	SA 213 Type 304 Type 308L SS	Weld Weldment
SA 182 Type F 304	SA 212 GR B SA 213 Type 304 Type 308L SS Type 309 MO SS	Weldment Expanded in Contac Weldment Weldment
SA 212 GR B	SA 240 GR S Type 304 SS Type 308L SS Type 309 MO SS	Cladding Weldment Weldment Weldment
SA 213 Type 304	Type 308L SS	Weldment
SA 240 GR S	Type 308L SS Type 309 MO SS	Weldment Weldment
Type 304 SS	Type 308L SS Type 309 MO SS	Weldment Weldment
Type 308L SS	Type 309 MO SS	Weldment
Type 309 MO SS		

1.4.1.3 Primary Stop Valve - Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
A 182 GR F 304	A 351 GR CF8 A 351 GR CF8 C Stellite #6	Weldment Seating Weldment
A 216 GR WCB	A 276 Type 304 SS	Compression
A 276 Type 304	A 276 Type 304 Stellited 1-D A 351 GR CF8 Graphite Impreg. Asbestos (John Crane 187-1 Diagonal Cut) Haynes Alloy #25 (AMS-5759) Type 304 SS	Compression Close Tight Fit; welded Compression Compression; close tolerance; stem bushing Weld
A 276 Type 304	Graphite Impreg. Asbestos Stellited 1-D (John Crane 187-1 Diagonal Cut) Haynes Alloy #25 (AMS-5759)	Compression Compression and close tolerance
A 351 GR CF8	A 351 GR CF8 C Flexitallic Style R3-15 Graphite Impregnated Asbestos (John Crane 187-1 Diagonal Cut) Type 304 SS Haynes Alloy #25 (AMS 5537) Haynes Alloy #25 (AMS 5759) Stellite #6 Type 308L SS	Compression Compression Compression Compression Compression Weldment Weldment
A 351 GR CF8 C	Haynes Alloy #25 (AMS 5759) Stellite #6	Compression Compression

1.4.1.3 Primary Stop Valve - Material Interface Listing (Continued)

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
Flexitallic Style R3-15		
Graphite Impreg. Asbestos (John Crane 187-1 Diagonal Cut)	Haynes Alloy #25 (AMS 5537)	Compression
Haynes Alloy #25 (AMS-5537)	Haynes Alloy #25 (AMS 5759)	Compression
Haynes Alloy #25 (AMS-5759)	Type 304 SS	Compression
Stellite #6		
Type 304 SS		
Type 308L SS		

1.4.1.4 Primary Check Valve - Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
A 182 GR 304 L (Gold Plated)	A 182 GR F 6	Compression
	A 182 GR F 316 M	Compression
	A 351 GR CF8 M	Compression
	Inconel X-(Jan-W- 562A Class 2)	Compression
A 182 GR F 6	A 182 GR F 316 M	Compression
	A 351 GR CF8 M	Compression
A 182 GR F 316	A 351 GR CF8 M	Compression;welded
	Inconel X-(Jan-W- 562A Class 2)	Contact(Spring loa
	Stellite #6	Compression
	Stellite #25	Compression
	Type 416 SS	Compression; close tolerance
A 351 GR CF8	Type 308L	Weldment
	Stellite #6	Compression
Type 308L		
Inconel X-(Jan-W 562A Class 2)		
Stellite #6		
Stellite #25	Type 416 SS	Bolted
Type 416 SS		

1.4.2

Primary Relief System - Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
SA 105 GR 1	SA 106 GR A	Welded
	Type 304 SS	Weldment
	Type 308L	Weldment
	Carbon Steel	Weldment
SA 106 GR A	Type 304 SS	Weldment
	Type 308L	Weldment
	Type 316 SS	Weldment
	Carbon Steel	Weldment
SA 212 GR B	Type 316 SS	Weldment
	Carbon Steel	Weldment
Type 304 SS		
Type 308L		
Type 316 SS		
Carbon Steel		

1.4.3 Primary Vent System - Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
Type 304 SS	Type 308L SS	Weldment

1.4.4 Primary Coolant Blowdown System Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
SA 167 GR 3 Type 304	SA 376 GR Type 304	Weldment
SA 376 GR Type 304	Type 308 L	Weldment
Type 303 SS	Type 304 SS Chevron Packing Dupont Teflon	Compression;bolted Compression Compression
Type 304 SS	Type 308L Type 316 SS Type 416 SS Flexitallic Gasket Stellited 316 SS Chevron Packing Dupont Teflon Type 315 Stellited	Weldment Compression Bolted Compression Compression Compression Compression Compression
Type 308L		
Type 315 SS Stellited		
Type 316 SS	Type 416 SS Stellited 316	Compression Compression
Type 416 SS		
Chevron Packing		
Dupont Teflon		
Flexitallic Gasket		
Stellited 316 SS		

1.4.4.1 Primary Coolant Blowdown Valve Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
Type 303 SS	Type 304 SS	Compression
	Teflon	Compression
	Chevron Packing	Compression
Type 304 SS	Type 316 SS (Stellited)	Compression
	Type 329 SS	Compression
	Type 416 SS	Bolted
	Teflon	Compression
	Chevron Packing	Compression
	Flexitallic Gasket	Compression
Type 316 SS (Stellited)	Type 329 SS	Compression
Type 319 SS		
Type 329 SS	Type 416 SS	Compression
Type 416 SS		
Chevron Packing		
Flexitallic Gasket		
Teflon		

1.4.4.2 Primary Coolant Emergency Blowdown Isolation Valve
Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
SA 212 GR B	Asbestos Packing	Compression
	Carbon Steel	Weldment
	Exelloy	Weldment
	Parkerized Steel	Compression
	Siliconized Steel	Compression
	Soft Iron	Compression
SA 212 GR B (Stellited Face)	Exelloy	Compression
Asbestos Packing	Exelloy	Compression
	Parkerized Steel	Compression
	Siliconized Steel	Compression
Carbon Steel	Parkerized Steel	Compression
Exelloy	Parkerized Steel	Compression
	Siliconized Steel	Compression
Parkerized Steel		
Siliconized Steel		
Soft Iron		

1.4.5 Component Drain System Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
A 182 GR F 316	AISI Type 316 SS	Bolted
	Chrome Plate MS 600	Bolted
	Stellite MS 481	Compression
AISI Type 316 SS	Stellite MS 481	Compression
	Chrome Plate MS 600	Bolted
Chrome Plate MS 600		
Stellite MS 481		

1.4.6 Decay Heat System Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
A 182 GR F 316	Stellite MS 481	Compression;welded
	AISI 316 SS	Compression;bolted
	Chrome Plate MS 600	Compression
AISI Type 316 SS	Stellite MS 481	Compression
	Chrome Plate MS 600	Bolted;compression
Chrome Plate MS 600		
Stellite MS 481		

1.4.7

Purification System Material Interface Listing

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
A 105 GR 2	Evolloy 300 BHN (AISI 410)	Weldment
	A 216 GR WCB Stellited MS-481	Welded
A 216 GR WCB	Armco Iron BHN 90 Maximum	Compression
	Stellite NS-481	Compression
	Type 308L SS	Weldment
	Carbon Steel	Weldment
	Evolloy 300 BHN (AISI 410)	Compression
	Stellite AISI-C1120	Compression
A 216 GR WCB Stellited MS-481	Evolloy 300 BHN (AISI 410)	Compression
SA 167 GR 3 Type 304	SA 376 GR TP 304	Welded
	SA 312 GR TP 304	Welded
	SA 367 GR Type 304	Welded
	Type 308L	Weldment
SA 312 GR TP 304	Type 304 SS	Welded;bolted
SA 367 GR Type 304	Type 304 SS	Weldment
	Type 308L	Weldment
A 376 GR Type 304	Type 304 SS	Weldment
	Type 308L	Weldment
Armco Iron BHN 90 Maximum		
Type 303 SS	Dupont Viton Compound B Type 316 SS	Compression Weldment
Type 304 SS	SA 212 GR B SA 167 GR 3 Type 304	Bolted Weldment
Type 308L		
Type 316 SS		
Carbon Steel	Evolloy 300 BHN (AISI 410)	Weldment

1.4.7

Purification System (Continued)

<u>Material</u>	<u>Interface Material</u>	<u>Description of Interface</u>
Dupont Viton Compound B		
Evolloy 300 BHN (AISI 410)	Stellite	Compression
SA 212 GR B		
Stellite AISI-C1120		
Stellite NS-481

INDIAN POINT HEAT BALANCE 275 MW NUCLEAR PLANT

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

Figure 1

**NUCLEAR PLANT SCHEMATIC DIAGRAM
SHOWING ONE OF FOUR LOOPS**

Figure 2

SECTIONAL ELEVATION - NUCLEAR STEAM GENERATOR

Figure 3

SECTIONAL ELEVATION - ENTIRE PLANT

Figure 4

PLAN OF STATION

Figure 5

REACTOR VESSEL

Figure 6

PRIMARY BLOWDOWN PURIFICATION SYSTEM

Figure 7

FIGURE 8

INDIAN POINT — LIQUID WASTE SYSTEM

11921033
124
781
855
255

APPENDIX C

IPl ESEERCO/DOW PHASE 1 SUMMARY

Section 1 - Test Program

Section 2 - Summary of Results

APPENIDX C - Section 1

IP 1 ESEERCO/DOW PHASE 1 SUMMARY

Corrosion Test Program

This section of appendix C is a description of a corrosion test program for determining the compatibility of Dow NS-3 solvent with the materials of Indian Point 1.

PWR Decontamination
Solvent
Materials Testing
Program
Report 1
Addendum 1
for ESEERCO Project EP-5-7

Presented to the
Empire State Electric Energy Research Corporation
1271 Avenue of the Americas
New York, New York

by

Dow Industrial Service-Nuclear
Division of the Dow Chemical Company, U.S.A.
Midland, Michigan

November 26, 1976

6/1/77
12/17/76

CONTENTS-Addendum 1

	Page
1. INTRODUCTION	1
2. IMPORTANT ISSUES IN CORROSION TESTING	1
3. GENERAL CORROSION	4
3.1 Testing Approach	4
3.2 Phase 1 General Corrosion Testing	5
4. GALVANIC CORROSION	5
4.1 Overview of Galvanic Testing	5
4.2 Phase 1 Galvanic Corrosion Testing	6
5. CREVICE & PITTING CORROSION	7
5.1 Corrosion Mechanisms and Testing Method	7
5.1.1 Crevice Initiation Using Artificial Multi-Crevice	7
5.1.2 Electrochemical Pitting Initiation Studies	7
5.2 Solvent Sequestered in a Crevice	9
5.2.1 Propagation in Subsequent Service	9
5.2.2 No Crevice Evaluations will be Made in Phase 1 Corrosion Testing	10
6. STRESS CORROSION TESTING	11
6.1 Stress Corrosion Testing Methods	11
6.2 Phase 1 Stress Corrosion Testing	11
7. PHASE II SPECIAL TESTS	13
7.1 Graphitar 14	13
7.2 Irradiated Materials	13
8. SUMMARY	14

TABLES

1. Significant Wetted Alloys to be used in Phase 1 Corrosion Testing	15
---	----

LEGAL NOTICE

This report was prepared as an account of work performed by the Dow Chemical Company ("Dow") and sponsored by the Empire State Electric Energy Research Corporation ("ESEERCO"). Neither ESEERCO, members of ESEERCO nor Dow nor any person acting on behalf of either:

- a. Makes any warranty or representation, expressed or implied, with respect to accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately-owned rights; or
- b. Assumes any liability with respect to the use of, or for damages resulting from the use of, any information, apparatus, method or process disclosed in this report.

1. INTRODUCTION

The planned ESEERCO PWR radiation level reduction program will involve the use of a proprietary solvent developed by The Dow Chemical Company. But before this solvent can be used, a corrosion study will be made to insure the compatibility of the solvent with the materials of construction of a PWR. What follows is a summary of the comprehensive corrosion program broken down according to corrosion mechanism. Within the proposed outline for each corrosion mechanism a brief summary of the work to be done in Phase 1 of the corrosion program is listed.

The corrosion program is divided into two phases. The first phase being an abbreviated program designed to uncover any obvious negative characteristics of the solvent, while Phase II will involve testing the more subtle aspects of solvent behavior such as decomposition in a crevice, etc.

2. IMPORTANT ISSUES IN CORROSION TESTING

The corrosion testing program should be organized in such a way that all possibilities of corrosion damage resulting from decontamination can be evaluated. The important issues that must be addressed by a comprehensive testing are as follows:

1. All alloys and heat treatments which the decontaminating solution will wet should be tested.
2. All special geometrical circumstances of bi-metallic joints, crevices, welds, safe ends, etc., should be tested. Special consideration should be given to those areas where plant integrity is particularly important during decontamination.
3. Stress corrosion is the only corrosion mechanism that operates at such a rate that gross failure through this mechanism is possible during the time of the decontaminating process. Therefore, emphasis must be placed on this corrosion mechanism and specimens should be tested in both the open and crevice circumstances.
4. The decontamination process will occur at elevated temperatures and, therefore, temperature transients will occur. The maximum temperature transient must be screened since chemical reaction rates are so strongly influenced by temperature.
5. In the event the cleaning process does not occur as originally planned, time of exposure should be expanded in the test program. This is in line with the conservative philosophy that should be adopted in the corrosion program. Always test on a "worst possible case" basis (so that all bets are covered).

6. As the corrosion product impurities are transferred from metal surfaces to the solvent, its composition is shifted to a "used" composition. Most corrosion testing should be done on a "simulated used" version of the solvent.
7. All air cannot be excluded from the decontaminating solvent before it is injected in the plant nor can all air be removed from the plant prior to decontaminating. Therefore, the solvent must be tested over some range of aeration with the most conservative state being an air saturated condition.
8. The surface to volume ratio throughout the plant will vary over a wide range. To insure that adequate inhibitor levels are present in the solvent, a range of inhibitor concentrations must be tested.
9. A conservative approach to corrosion testing dictates that one assume pre-existing cracks and crevices in the plant. How these cracks react after decontamination must be answered. In addition, do the thermal decomposition products of solvent affect cracks and crevices?
10. Materials in the core region of the reactor have experienced a significant integrated neutron flux. The question of how these highly irradiated materials behave in the solvent should be addressed.

11. The behavior of the cleaned surfaces to normal reactor operation must be evaluated so that if there is a "pickling effect" from the cleaning this can be studied before the actual cleaning.

All of the above issues need to be addressed in the corrosion program. How these issues will be addressed according to each corrosion mechanism is listed.

3. GENERAL CORROSION

3.1 Testing Approach

All wetted alloys should be evaluated according to their general corrosion behavior. Table 1 lists the important alloys in Indian Point 1. These alloys need to be screened under a wide range of conditions. Tests should be made regarding inhibitor concentration, temperature transients, pH transients, flow rate, alloy surface finish, heat treatment, degree of aeration, vapor phase corrosion, long term storage tests, welds, potential impurities, alternate wetting and drying, and some test should be made on the solvent with some of the more important components out of spec. Of course, not all the variables should be tested on each alloy. However, it is proposed that all variables be screened using type 304SS. If significant material behavior results from any of the variables, then additional tests should be developed to understand the nature of the change.

Alloys other than 304SS should be tested as a function of those variables that are relevant to that particular alloy class.

3.2 Phase 1 General Corrosion Testing

All alloys that will be wetted during the cleaning will have their general corrosion rates measured in both "new" and "simulated used" solvent for times and temperatures approximating the actual cleaning operation. In addition, type 304SS will be tested as a function of: heat treatment, inhibitor concentration, temperature, pH and surface finish.

4. GALVANIC CORROSION

4.1 Overview of Galvanic Testing

The following galvanic couples should be evaluated:

304 to SA 212 Gr. B and to SA 105 GR. II, Graphitar 14
to 304; 410 to 304, and 1020 to 304.

These alloy combinations should be screened by coupling them in the same fashion as they are electrically connected in the plant. Some couples will be welded, others bolted or samples pressed together.

The most important couples, such as safe-ends, clad areas, should be mocked-up and tested under dynamic conditions. This will not need to be done for the majority

of galvanic couples, only the critical ones. The less important couples should be tested by using flat-plate like coupons where the area ratios between the anodic and ~~cathodic~~ member of the couple can be easily varied. The parameters most important in screening galvanic corrosion are: anodic cathode area ratio, degree of aeration, solution velocity, temperature, inhibitor concentration, and the flow of electrical contact -- welded or mechanical.

4.2 Phase 1 Galvanic Corrosion Testing

This testing should center around the following metal combinations:

- 1) Graphitar to 304
- 2) 1020 to 304

Couple 1 will be tested to measure in mechanical property change in the Graphitar 14. Since this is a pump bearing material, it is important to know if the solvent is compatible with this material early in the test program.

Couple 2 represents galvanic couples that will exist in valve internals and also some auxillary loops where carbon steel is a structural material. The galvanic couple between 304 stainless and carbon steel is potentially the most extreme galvanic couple that will exist in Indian Point 1 and should provide data regarding the maximum corrosion rate to be expected from galvanic

coupling during the cleaning. (High area ratios will be tested to approximate a cladding crack).

Although the other couples that are listed are important, they will need to be covered in greater detail than Phase I testing will permit. They should be included under Phase II testing.

5. CREVICE AND PITTING CORROSION

5.1 Corrosion Mechanisms and Testing Method

Crevice or pitting corrosion occurs in two stages. The first phase being crevice or pit initiation followed by phase two - propagation. Separate specimen designs are necessary to study both steps as they relate to a cleaning solvent.

5.1.1 Crevice Initiation Using Artificial Multi-Crevice

A teflon multi-crevice can be bolted to a plate type specimen. The teflon crevice is machined so that many pie shaped crevices are exposed to the alloy under test. This multi-crevice provides for a statistical prediction of an alloy susceptibility to crevice attack, induced by either oxygen or metal ion concentration cells.

5.1.2 Electrochemical Pitting Initiation Studies

The multi-crevice test allows for crevice initiation susceptibility to be determined under only one set of environmental condition per test. By using potentiodynamic scans an alloy can be tested over a wide range

of both oxidizing and reducing environments in a single test. Measurements can be made regarding the propagation of an already initiated crevice. By using both polarization studies and the artificial crevice test, conclusions can be drawn regarding both susceptibility to attack and severity of crevice propagation.

Members from all wetted alloy classes should be tested electrochemically for crevice attack. Those alloys that indicate susceptibility to pitting or crevice attack from electrochemical testing should be tested using artificial multi-crevices. Enough data should be gathered on any alloy susceptible to pitting or crevice attack so that a probability can be attached to the depth of attack that will occur during a cleaning cycle.

Another very important question must be addressed in these crevice tests. How does the solvent behave when it is trapped in a crevice and remains there after flushing? That is, will solvent sequester in a crack under stress cause the crack to grow and if the solvent thermally decomposes in the crevice, will its decomposition products adversely affect the crack? To gather data in this area will require complex specimen geometrics and the use of fracture mechanism type tests.

5.2 Solvent Sequestered in a Crevice

The concerns that:

- (1) the solvent may cause propagation of existing cracks during the cleaning cycle; and
- (2) decomposition products of NS-3 may cause extension of existing cracks during subsequent operation. It is proposed that these questions should be addressed in Phase II of corrosion testing in the following manner.

Specimens will consist of wedge open loaded fracture mechanic specimens that are self stressed. The specimen is to be made of SA 105 Gr. II and clad with stainless in the notch area. The specimens will be covered with a stainless steel sheet by bolting to the specimen and gasketing with silicone rubber. Specimens will be loaded to a stress intensity value just below that for mechanical crack extension in air, and then exposed to a cleaning cycle. After the cleaning cycle, the stainless steel covering will be removed, and crack extension -- if any -- measured. A set of control specimens will be tested in an identical manner, except their exposure will be to demineralized water for the time and temperature specified for the cleaning cycle.

5.2.1 Propagation in Subsequent Service

The specimens for this portion of the program will be identified to those for the first part. However, two options exist for the introduction of NS-3 into the

pre-crack. One can run the specimen through a full simulated cleaning with a relatively low stress intensity to assure solvent trapped in the crack, and then load them to a desired stress intensity level after removal of the gasketing material and replacement of the stainless steel cover plates prior to exposure of 550°F water. This cycle would result in maximum blunting of the crack from corrosion during the cleaning cycle, but would most nearly represent the situation in the reactor vessel nozzles. The second option would be to minimize the crack blunting by exposing the slightly loaded specimens to ambient temperature solvent for some fixed period of time, with the gasketed stainless steel covers in place, and then proceeding as in the first example with the subsequent simulated reactor service tests. This would represent a worst case situation from the standpoint of stress intensity at the tip of the crack, since crack blunting would be minimal. The control specimens for these tests would be the same specimens used for the cleaning cycle tests.

5.2.2 No Crevice Evaluations will be Made in Phase 1 Corrosion Testing

It is proposed that all wetted alloys be screened electrochemically for pit and crevice initiation. In addition, stainless type 304 should be tested using artificial multi-crevices. No other alloys will be studied using the multi-crevice technique unless electrochemical testing indicates additional tests are warranted.

6. STRESS CORROSION TESTING

6.1 Stress Corrosion Testing Methods

Specimen designs for testing stress corrosion vary in their design from laboratory to laboratory, but generally they can be divided into two classifications.

This division can be made depending on the method used in stressing the specimens. In one method, a constant load is applied that will produce tensile stresses over the cross section in axially or in the outer fibers of a bend specimen. In the second method, the stresses are induced by subjecting the specimen to a predetermined and constant deflection. Both types are represented in the proposed corrosion program.

6.2 Phase 1 Stress Corrosion Testing

In Phase 1 studies a double U-bend configuration is recommended for screening all wetted alloys for susceptibility to stress attack. Since the U-bend contains large amounts of elastic and plastic strain, it provides one of the most severe tests available for smooth corrosion test specimens. A double U-bend configuration is desirable since it incorporates a crevice condition in the specimen and makes initiation times shorter when stress cracking is possible.

In addition to using U-bend tests, it is proposed that a slow straining technique be used for testing 304.

Type 304 is a ductile material that will relax under a static load and by using a continuous straining technique, will compensate for this tendency. This method has several inherent advantages over conventional U-bend type tests. Continuous straining of the specimen causes the passive film to break repeatedly; thus, fresh metal is constantly exposed to the test environment. The chances of both crack initiation and crack propagation are considerably increased. The strain on the specimen at any time is proportional to the displacement of the cross head and elapsed time and can be adjusted to suit the test environment and alloy. The load on the specimen is monitored continuously by a load cell. A sudden drop immediately identifies a crack initiation or advance, which can be verified by a metallographic examination. Since stress corrosion cracking in many alloy environment systems occurs preferentially at certain electrochemical potentials, it would be advantageous to do selected slow strain rate tests under controlled potential conditions. Controlled potential experiments are analogous to changing the oxygen concentration in solution. Changes in the positive direction correspond to adding oxygen; changes in the negative direction to removing oxygen.

7. PHASE II SPECIAL TESTS

7.1 Graphitar 14

To evaluate the behavior of Graphitar 14 as a bearing material requires that it be tested under actual dynamic conditions. To do this requires a small test loop be built that will incorporate a small canned motor pump using Graphitar #14 pressed bearings. This loop should be built so that actual cleaning conditions can be simulated. As a scale experiment, electrochemical corrosion monitoring devices also can be evaluated in this loop. If the electrochemical monitoring equipment proves accurate, it could be used during the actual cleaning as a safeguard.

7.2 Irradiated Materials

One unanswered question might be addressed as a special test. That is, do materials exposed to a large integrated neutron flux behave differently than unirradiated materials? The corrosion test program will involve unexposed metal. To correlate the findings on unirradiated materials to radiation damaged alloys it will be necessary to test an irradiated alloy -- preferably Type 304 stainless steel. If there are Charpy V-Notch containers in the core that are available for testing, they could be cut into tensile specimens and used to answer this question.

8.0

SUMMARY

In developing effective solutions for decontamination it is important that they do not compromise the materials of construction during the decontaminating process nor lead to aggravation of corrosion at a later time during normal operation or lay-up. The corrosion program as outlined will point out any potential corrosion problems before the solvent is used. Further, it is assumed that "worst case" conditions used during testing insure no reasonable environmental conditions will be overlooked. The Phase I portion of The Corrosion Study is rather ambitious in its scope. If it is not possible to finish Phase I in the allotted time, the unfinished portion should be included in Phase II testing.

Table 1 describes the important alloys to be screened in Phase I corrosion testing. The corrosion mechanisms to be studied on each alloy are listed in this table. Additional will be made to the table as necessary for Phase II work.

TABLE 1

Significant Wetted Alloys
To Be Used
In Phase I Corrosion Testing

<u>Alloy</u>	<u>Location</u>	<u>Type of Cor. Tests</u>
304	Primary Piping & Cladding	All Tests
304L	"	General, galvanic
316	"	General, galvanic
CF-8	Valve body; impeller in Main Coolant Purge	General, dynamic, " crevice
416	Bolting Mat'l in valves	General, stress, galvanic
410	"	"
SA 105 Gr II	Structural Mat'l in Nozzles	Crevice, galvanic, Stress
SA 212 Gr B	Structural Mat'l in Reactor Vessel	Crevice, galvanic, Stress
1020	Valve Internals	Galvanic, general
Haynes Alloy 25	Valve Internals	Stress
Stellite #6	Hard Facing In Valves	Stress
Graphitar 14	Main Coolant Pump Bearings	Dynamic Loop Test

APPENDIX C - Section 2

IP 1 ESEERCO/DOW PHASE 1 SUMMARY

Test Results

This section is a summary of the findings of the corrosion testing performed by Dow on Indian Point 1 materials. Included are:

	<u>Page</u>
Conclusions	C-1
Materials for Test	C-3
Test Basis	C-4
Digest of Results	C-10

Conclusions - Phase 1 Static Corrosion Tests

1. Single metal testing revealed no corrosion rates greater than 10 mils per year, with the exception of Graphitar 14, a pump bearing material.
2. The proposed solvent base metal compatibility for single metal types tested was shown to be relatively insensitive to (a) changes in the pH, (b) variation of the solution volume to surface area, (c) the surface finish, (d) the degree of aeration of the solution, (e) long shelf life, (f) welds, and (g) having the simulated used pretreatment oxidizer out of specification.
3. The current inhibitor concentration appears adequate in protecting the base metals of 250⁰ F, but doubling the concentration is recommended as a precautionary measure in case temperature transients of +25⁰ F occur.
4. Care must be taken to insure solvent temperature do not exceed 275⁰ F. Stress corrosion cracking was observed in 112 hour exposures at 300⁰ F.
5. No stress corrosion will occur on highly stressed and cold worked type 304 stainless steel in the annealed condition on exposure to the solvent for tests up to 300 hours at 250⁰ F.

6. Testing to date has shown Dow Solvent NS-3 to be no more aggressive toward highly sensitized type 304 stainless steel than deionized water.
7. Testing of the base metals galvanically coupled resulted in no stress corrosion failures. In general, the stainless steels corroded at approximately 2 mpy while the carbon steel they were coupled to corroded less than 40 mils per year. Graphitar[®] 14 corroded at approximately 81 mpy. The structural integrity of the Graphitar[®] 14 did not appear to be compromised by the decontamination process.
8. A stainless steel to carbon steel couple was tested in the proposed solvent (at a 1:1 area ratio), and the carbon steel was found to be sensitive to changes in both pH and the degree of aeration. If the pH can be kept within 1 pH unit of specifications, the worst possible corrosion rates would be 52 mpy. Aeration before testing can also increase corrosion up to 55 mpy. Having the concentration of simulated used pretreatment oxidizer varied did not appear to significantly affect corrosion.
9. If a cladding crack exists, exposing some carbon steel to stainless steel, the carbon steel can be expected to corrode at a rate proportional to the relative surface area of each metal. As more carbon steel is exposed, the corrosion rates will decrease.

Materials of Construction Identified for Test

Eleven different base materials of construction have been identified as being possibly wetted by the proposed decontamination solvent. Each of these has been included in the corrosion testing schedule. In addition, two extra metals were included in the testing. First, 1020 was included because it is a convenient carbon steel, similar in many ways to SA-212, a carbon-silicon steel already on the testing list. And secondly, Inconel 600 was included, primarily because it is a material of construction commonly found in nuclear reactors like Indian Point 1.

It should be noted that SA-212, a boiler plate material, is not supposed to be wetted by the solvent because it has been clad by stainless steel. But, if a cladding crack did already exist, then the SA-212 would be wetted. Hence it was included in the list of materials of construction to be tested. However, corrosion testing using SA-212 is impossible because it is no longer manufactured. Instead, this metal has been replaced by two materials, with similar composition and properties, namely SA-515 and SA-516.

Tables I and II, respectively reflect the comparative compositions and mechanical properties for all materials tested. As can be seen, the compositions and mechanical properties for each heat of the same metal vary slightly. The exact composition and properties of the materials of construction used in building Indian Point 1, however, are unknown. Hence, it was felt that the most probable behavior of the materials at Indian Point 1 could be represented by corrosion testing of several heats and then extracting a composite, "heat averaged" behavior. This was done whenever possible.

Test Basis - Worst Possible Case

It was decided at the beginning of this corrosion study that testing would be done on a "worst possible case" basis. If the metals didn't fail under the most severe cases, then the metals surely wouldn't fail under far less severe conditions. This philosophy was employed primarily to reduce the number of tests needed to have high confidence in the solvent to base metal compatibility. Should the metals fail in the "worst possible case basis," then conditions would be relaxed towards more realistic conditions anticipated by surveys of Indian Point 1. Testing would then continue until threshold values were obtained, below which stress corrosion cracking or other failure would no longer be observed. Testing under relaxed conditions was necessary in one of the specialty screening tests, which will be discussed later.

Two primary aspects used for "worst possible case" testing basis are sensitization and specimen shape. "Sensitization involves the intergranular precipitation of chromium carbide in the austenitic steels when exposed to temperatures in the approximate range of 800 to 1500 F. Precipitation increases with increasing carbon content and exposure time. The principal effect is the depletion of chromium content in a thin envelope surrounding each grain, promoting susceptibility to corrosive attack ... Desensitization of unstabilized alloys can be achieved by ... extending the heating period within the sensitization range until diffusion of chromium from deeper in the grain restores the depleted envelope."* Hence, testing the austenitic steels (300 series stainless steels),

* American Society for Metals Handbook, 8th Ed., Vol. 1, 442 (1961).

Table I

BASE METAL COMPOSITIONS

Specimen Label	Source of Metal	Heat #	C	Mn	Si	Cr	Ni	Co	Mo	P	S	Cu	Fe	W
SS-304 Nominal			0.08 max.	2.0 max.	1.0 max.	18.00 - 20.00	8.00 - 12.00			.045 max.	.030 max.		Balance	
SS-304 III	Allegheny Ludlum	346815	.07	1.73	.69	18.60	8.62			.028	.005		Balance	
SS-304 VI	Allegheny Ludlum	420244	.066	1.74	.57	18.42	8.52			.023	.017		Balance	
SS-304 VII	Washington Steel Co	166476	.065	1.69	.60	18.13	8.23			.021	.004	.24	Balance	
SS-304 VIII	Armco Steel Corp	261213	.053	1.41	.63	18.56	9.08	.13	.17 .05	.021	.012	.10	Balance	
SS-304L Nominal			.03 max.	2.0 max.	1.0 max.	18.00 - 20.00	8.00 - 12.00						Balance	
SS-304L IV	Washington Steel Co	238355	.026	1.48	.60	18.11	9.44		.03	.018	.019	.01	Balance	
SS-304L V	Allegheny Ludlum	421292	.018	1.55	.38	18.66	10.34			.030	.012		Balance	
CF-8 Nominal			.08 max.	1.50 max.	2.00 max.	18 - 21	8 - 11						Balance	
CF-8 I	Birdsboro Corp	Unknown	.068	.68	1.35	19.75	8.40		.09	.020	.027	.07	Balance	
SS-316 Nominal			.08 max.	2.0 max.	1.00 max.	16.00 - 18.00	10.00 - 14.00		2.00 - 3.00				Balance	
SS-316 II	Eastern Stainless Steel	K60041	.053	1.26	.48	17.31	10.90	.26	2.07	.027	.019	.39	Balance	
SS-316 III	Allegheny Ludlum	422334	.020	1.47	.35	17.21	13.38			.024	.015		Balance	
			also (N ₂ .052)											
SS-410 Nominal			.15 max.	1.00 max.	1.00 max.	11.50 - 13.50							Balance	
SS-410 II	Allegheny Ludlum	311839	.13	.37	.43	12.36	.34		.14	.023	.005	.11	Balance	
			also (Sn .010, H ₂ .022)											
SS-416 Nominal			.15 max.	1.25 max.	1.00 max.	12.00 - 14.00					.15 min.		Balance	
SS-416 I	Unknown													
SA-105 Gr II Nominal			0.35 max.	.90 max.	0.35 max.					.05 max.	.05 max.		Balance	
SA-105 Gr II I	Pacific State Steel Corp	38875	.34	.82	.19					.001	.017		Balance	
SA-106 B Nominal			0.30 max.	.29 - 1.06	.10 min.					.048 max.	.058 max.		Balance	
SA-106B I	Phoenix Steel Corp	66549	.26	.81	.19					.012	.028		Balance	

(continue)

Table I (continued)

Specimen Label	Source of Metal	Heat #	C	Mn	Si	Cr	Ni	Co	Mo	P	S	Cu	Fe	W
SA-212B Nominal			.31 max.	.90 max.	.13 - .33					.04 max.	.05 max.		Balance	
SA-516 Nominal			.27 max.	.85- 1.20	.15- .30					.035 max.	.040 max.		Balance	
SA-515 Nominal			.31 max.	.90 max.	.13 - .33					.035 max.	.04 max.		Balance	
SA-516 I	U.S. Steel Corp	267522	.25	1.07	.23					.08	.025		Balance	
SA-516 II	U.S. Steel Corp	165132	.20	1.11	.20					.004	.017		Balance	
SA-515 I	Armco Steel	36024	.29	0.63	.23					.007	.025		Balance	
Haynes Alloy 25 Nominal			.15 max.		19.0 - 21.0	9. - 11	Balance					2.0 max.	14. - 16	
Haynes 25 I	Cabot Corp	1860- 6-1308	.10	1.34	.20	19.47	10.34	Balance		.021	.007		2.51	15.36
Haynes Stellite 6B Nominal			.9 - 1.4	1.0 max.	1.50 max.	27.0 - 31.0	3.0 max.	Balance	1.5 max.				3.0 max.	3.5 - 5.5
Haynes Stellite 6B I	Cabot Corp.	1818- 5-1090	1.01	1.40	.50	29.96	2.76	Balance	1.47				2.51	3.82
Graphitar 14	Wickes Corp	Furnished by manufacturer - No analysis												
1020 Nominal			.18 - .23	.30 - .60							.040 max.	.050		
1020 I		No mill analysis but certified to be 1020												
Inconel 600 Nominal			.15 max.	1.00 max.	0.50 max.	14.00 17.00	72.0 min.				0.015 max.	0.50 max.	6.00- 10.00	
Inconel 600 III	International Nickel	NX5802	.03	.23	.16	14.89	76.38				.006	.23	8.05	

Table I

BASE METAL MECHANICAL PROPERTIES

Specimen Label	Tensile Strength (psi)	.2% Yield Strength (psi)	Elongation in 2 Inches (%)	Reduction of Area (%)	Hardness
SS-304 Nominal	85,000	35,000	55	65	B 80
SS-304 III	89,000	44,000	53.5		B 84
SS-304 VI	88,000	44,000	50.5		B 83
SS-304 VII	90,100	35,780	60.0		45T 48.5
SS-304 VIII	91,200	41,800	60.0		B 81
SS-304L Nominal	80,000	30,000	55	65	B 76
SS-304L IV	80,720	34,640	55.5		45T 47.0
SS-304L V	81,000	40,000	51.		B 80.0
CF-8 Nominal	72,500	37,750	58.0	62.1	
CF-8 I	Unknown				
SS-316 Nominal	85,000	35,000	55	70	B 80
SS-316 II	89,700	48,800	52.		B 85
SS-316 III	83,000	44,000			
SS-410 Nominal	75,000	40,000	30	65	B 82
SS-410 II	76,000	50,000			
SS-416 Nominal	75,000	40,000	30	65	B 82
SS-416 I	Unknown				
SA-105 Gr II Nominal	70,000	36,000	22	30	
SA-105 Gr II Heat I	76,250	54,100	32	62.8	
SA-106B I Nominal	60,000	35,000	22		
SA-106B Gr I Heat I	78,500	44,000	29		

Table II (continued)

Specimen Label	Tensile Strength (psi)	.2% Yield Strength (psi)	Elongation in 2 Inches (%)	Reduction of Area (%)	Hardness
SA-212 B Nominal	70,000 - 85,000	38,000	21		
SA-516 Gr 70 Nominal	70,000 - 85,000	38,000			
SA-515 Gr 70 Nominal	70,000 - 85,000	38,000	21		
SA-516 Gr 70 I	88,200	70,700	16.0		
SA-516 Gr 70 II	82,200	63,300	21.0		
SA-515 Gr 70 I	77,300	55,900	26.0		
Haynes® Alloy 25 Nominal	155,000	70,000	50		130 Brin
Haynes® Alloy 25 I	135,000	64,000			
Haynes Stellite® 6B Nominal			Unknown		
Haynes Stellite® 6B I	141,500	83,000			
Graphitar® 14			Unknown		
1020 Nominal	61,000	51,000	15	40	121 Brinell
1020 I			Unknown		
Inconel 600 Nominal			Unknown		
Inconel 600 III	96,500	46,500	40		B 79

according to the "worst possible case" possible philosophy, required sensitizing them. The sensitizing parameters selected were 50 hours at 1200^o F. The microstructures of SS-304 in the annealed, as received condition and after sensitization are depicted in Figures 1 and 2. These can be compared to the ASM standards.**

The reason behind such extreme sensitization times is due to the manner in which the reactor pressure vessel was fabricated. Due to the thickness of this vessel a stress relieving treatment of approximately 50 hours duration could have been necessary. The vessel would have been clad with stainless before stress relieving and thus the weld clad of 304 probably represents the most highly sensitized area of type 304 stainless steel in the plant.

The other way in which "worst possible case" conditions were imposed during testing was through the use of U-bend test coupons. Whenever specimen thickness did not prevent it, the test coupons were bent to a 3/4-inch inside diameter. The U-bends were formed after sensitization for the 300 grade stainless steels. The stresses imposed on each U-bend exceed yield strengths, as is evidenced by the plastic deformation necessary for shaping the coupons. The selection of U-bend specimen assured corrosion testing over a wide range of stresses ranging from tensile on the outside surface to compression on the inside surface. Stress relief procedures were purposefully skipped here in contrast to standard construction procedures, again to establish "worst possible case"

**American Society for Metals Handbook 8th Ed., Vol. 7, 134 (1972).

conditions. Bolting the specimen together prior to testing imposed constant strain during testing.

Digest of Results

Corrosion test results on the first NS-3 solvent were unacceptable. Subsequently a new solvent was developed.

The following is a condensation of the test results on the reformulated NS-3.

Single Type of Metal Testing - Base Metal Wetted

Table III gives the results of single type metal corrosion testing. The samples were subjected to 112 hours exposure to NS-3 at a temperature of 250 F. No stress corrosion cracking nor significant pitting was observed for any specimens tested. Since Graphitar 14 was the only material which is significantly affected by the pretreatment oxidation step, pretreatment was included in this test. Microscopic examination of the surface of the test coupons did not reveal any compromise in the surface integrity.

pH Testing

The compatibility of the solvent with the materials to changes in pH was analyzed. Table IV gives the results of these tests. In general, the corrosion rates appear to decrease as the pH increases. No particular pH sensitivity was noted for any metal, and all corrosion rates were under 10 mils per year.

TABLE III
SINGLE METALS TESTING
Corrosion Rates (mpy)

Metal	Corrosion Rate	
SS-304 (Sensitized Double U-Bend)	1.35	0-58
SS-304 (Sensitized Single U-Bend)	0.66	0-18
SS-304 (Double U-Bend)	0.07	0-8
SS-304L (Sensitized Double U-Bend)	1.14	0-30
SS-304L (Sensitized Single U-Bend)	0.43	0-2
SS0304L (Double U-Bend)	0.17	0-20
CF-8 (Straight Sample)	0.10	A-5
SS-316 (Sensitized Double U-Bend)	0.78	0-30
SS-316 (Sensitized Single U-Bend)	0.74	0-8
SS-410 (Double U-Bend)	0.99	0-4
SS-410 (Single U-Bend)	7.36	0-3
SS-416 (Single U-Bend)	2.28	0-2
SS-416 (Straight Sample)	1.12	A-3
SA-105 B (Straight Sample)	1.13	A-2
SA-106 B (Straight Sample)	1.53	A-2
SA-515 Gr 70 (Straight Sample)	2.73	A-5
SA-516 Gr 70 (Straight Sample)	3.59	A-12
1020 (Straight Sample)	1.48	A-9
Haynes® Alloy 25 (Single U-Bend)	0.04	0-5
Stellite® 6B (Single U-Bend)	0.07	0-5
Inconel 600 (Sensitized Single U-Bend)	1.42	0-5
Inconcel 600 (Unsensitized Single U-Bend)	1.29	0-5
Graphitar® 14 (Pretreated Straight Sample)	31.2	A-2

x-y Legend

x = Number of U-bends which cracked.
y = Number of specimens tested.
A = Not U-bend.

Table IV

pH VARIATION TESTING FOR DIFFERENT METALS
Corrosion Rates (mpy)

Metal	pH 6	pH 7	pH 8
SS-304 (Sensitized Double U-Bend)	1.17 0-16	1.35 0-58	0.81 0-12
SS-316 (Sensitized Double U-Bend)	0.80 0-4	0.78 0-30	0.48 0-4
SS-316 (Sensitized Single U-Bend)	0.25 0-2	0.74 0-8	0.54 0-2
SS-410 (Single U-Bend)	1.94 0-2	7.36 0-3	1.00 0-2
SA-516 Gr 70 (Straight Sample)	5.40 A-2	3.59 A-12	1.72 A-2
1020 (Straight Sample)	8.08 A-4	1.48 A-9	0.99 A-4
Haynes® Alloy 25 (Single U-Bend)	0.04 0-2	0.04 0-5	0.04 0-2
Stellite® 6B (Single U-Bend)	0.11 0-2	0.07 0-5	0.09 0-2

x-y Legend

- x - Number of U-bends which cracked.
- y - Number of specimens tested.
- A - Not U-bend.

Inhibitor Concentration Testing

Variations in inhibitor concentration did not significantly change the corrosion rate of carbon steel. (See Table V) Carbon steel was selected for test because its corrosion rate in uninhibited NS-3 was 68.9 mils/yr, distinctly higher than the stainless steels.

Tests were made to determine the performance of inhibitor under out of spec temperature conditions. The results of the tests, given in Table VI, show that doubling the inhibitor can avoid cracking stainless steel at 300 F, and available corrosion rate, which indicates breakdown of protection mechanism, does not occur at temperature of 275 F. This suggests that the inhibitor concentration solvent NS-3 should be doubled or even tripled in order to insure adequate protection in case of a temperature excursion of + 25 to + 50 F.

Impurity Ion Testing

Test to determine the effect of deposit dissolved in solvent upon solvent to base metal compatibility resulted in cracking of sensitized stainless u-bends but the same thing occurred when distilled water was used. A more sophisticated test was tried on standard tensile specimens loaded uniaxially at a slow rate. Again sensitized 304 cracked in both water and solvent. Additional tests will be made in Phase II work to develop meaningful conditions for comparison.

TABLE V
INHIBITOR CONCENTRATION TESTING

Corrosion Rates (mpy)

1020 (Straight Samples)

<u>No Inhibitor</u>	<u>25% Norm</u>	<u>50% Norm</u>	<u>100% Norm</u>	<u>300% Norm</u>
A-2 68.9	A-3 2.36	A-3 1.67	A-9 1.48	A-3 1.21

SS-304 Sensitized Double U-Bends

<u>100% Norm</u>	<u>200% Norm</u>
0-58 1.35	0-4 0.26

x-y Legend

x - Number of U-bends which cracked.
y - Number of specimens tested.
A - Not U-bends.

Table VI

HIGH TEMPERATURE INHIBITOR CONCENTRATION TESTING

Corrosion Rates (mpy)

Metal	Inhibitor Concentration	250°F	275°F	300°F
SS-304 (Sensitized Double U-Bend)	100%	1.35 0-58	5.55 0-30 53.3 0-4 &	
SS-304 (Sensitized Double U-Bend)	200%	0.26 0-4	3.94 0-16	
SS-304 (Sensitized Single U-Bend)	100%	0.66 0-18	1.32 0-10	11.0 3-7 101. 0-3 &
SS-304 (Sensitized Single U-Bend)	200%			8.67 0-6 102. 0-4 &
SS-304L (Sensitized Double U-Bend)	100%	1.14 0-30	2.15 0-6	
SS-304L (Sensitized Double U-Bend)	200%		1.65 0-6	

x-y Legend

- x - Number of U-bends which cracked.
 y - Number of specimen tested.
 A - Not U-bend.

Surface Finish Testing

Polished surface specimens of sensitized 304 stainless experienced the lowest corrosion rates in solvent NS-3, but these rates were not significantly lower than rates obtained from either sandblasted or unaltered surfaces. No stress corrosion cracking was observed. There were no significant variations in the corrosion rate of carbon steel as well. The range of corrosion rates over all tests was 0.37 - 1.56 mpy.

Aeration Testing

Aeration testing indicated solvent NS-3 is not particularly sensitive to aeration.

Temperature Variations Testing

Corrosion rates at 300 F were significantly greater than at 275 F for both 304 stainless and carbon steel. Split order of magnitude corrosion rates were observed at 275 F. The results are given in Table VII; split rates are avoided by doubling the inhibitor concentration.

Long Term Testing

Tests of NS-3, 97 days after preparation of the solution yielded no unusual results. Nor did tests of specimens exposed to the solvent for 306 hours indicate any stress corrosion rate of sensitized double U-tubes of 304 stainless was 0.33 mpy.

Weld Testing

Stainless to stainless 304 stressed weld specimens were prepared and exposed to NS-3 solvent. There was no appreciable attack at the weld; the corrosion rate was 0.06 mpy.

Table VII

TEMPERATURE VARIATION TESTING
Corrosion Rate (mpy)

<u>Temperature</u>	<u>Single U-Bends Sensitized SS-304</u>	<u>Double U-Bends Sensitized SS-304</u>	<u>Straight Weight Loss 1020</u>
200°F	0.22 0-5		1.65 A-5
250°F	0.66 0-18	1.35 0-58	1.48 A-9
275°F	1.32 0-10	5.50 0-30 53.3 0-4 &	2.48 A-5
300°F	11.0 3-7 101. 0-3 &		13.1 A-5

x-y Legend

- x - Number of U-bends which cracked.
- y - Number of specimens tested.
- A - Not U-bends.

Simulated Use of Pretreatment Out of Spec

There was no indication of stress corrosion cracking or significant corrosion rates when specimens were subjected to pretreatment oxidizer concentrations from 75 to 125% of normal.

Galvanic Couples Testing

Galvanic test results of metal couples in NS-3 for 112 hours at 250 F are given in Table VIII. Only two couples had corrosion rates over 40 mpy. Microscopic examination of the graphitar 14 after testing revealed no compromise of structural integrity.

When surface area ratios were varied from the 1:1 of the above, increasing the stainless steel surface to simulate a cladding crack, the corrosion rate of the carbon steel surface increased to 206 mpy when the ratio reached 10:1 at a linear rate. The effect of stress on a sharp notch imposed on galvanic couples is projected for future testing.

Variations in pH were made under galvanic conditions. For the stainless steel to carbon steel couples, the corrosion rate of carbon steel decreased with increase in pH. The pH was adjusted ± 1 from the pH 7 of the decon solution. Results are shown in Table IX.

Exposure of couples to an 8 hour period of aeration showed the carbon steel to increase from 16.7 mpy with no bubbling to 55.4 mpy with an 8 hour oxygen bubbling period. Stainless corrosion rate was 0.70 mpy.

Table VIII

BASE METAL COMBINATIONS POSSIBLY WETTED
Corrosion Rates (mpy)

Area Ratio 1:1

Metal "A"	to	Metal "B"	Corrosion Rates	
			Metal "A"	Metal "B"
SS-304 (Sensitized)		1020	1.61 0-9	16.7 0-9
SS-304L (Sensitized)		1020	0.83 0-10	10.6 0-10
CF-8		1020	0.13 A-5	9.7 A-5
SS-316 (Sensitized)		1020	1.47 0-4	17.1 0-4
SS-410		1020	1.06 0-5	7.8 0-5
SS-416		1020	104. A-5	2.34 A-5
SA-515 Gr 70		1020	4.13 A-5	2.99 A-5
SA-516 Gr 70		1020	5.12 A-5	2.71 A-5
SS-304 (Sensitized)		SA-515 Gr 70	0.54 A-5	18.4 A-5
SS-304 (Sensitized)		SA-516 Gr 70	0.54 A-5	24.3 A-5
SS-304 (Sensitized)		SA-105 Gr B	0.63 0-2	35.5 A-2
SS-304 (Sensitized)		SA-106 Gr B	0.76 0-2	36.4 A-2
SS-304 (Sensitized)		SS-410	1.03 0-5	0.98 0-5
SS-304 (Sensitized)		Graphitar® 14	0.35 0-5	81.2 A-5

x-y Legend

- x - Number of U-bends which cracked.
y - Number of specimens tested.
A - Not U-bends.

TABLE IX
GALVANIC pH TESTING
Corrosion Rates (mpy)

<u>pH</u>	<u>Sensitized SS-304</u>	<u>1020</u>
	0-4	0-4
-1	2.27	52.3
	0.9	0-9
0	1.61	16.7
	0-4	0-4
+1	2.61	7.30

	<u>SS-410</u>	<u>1020</u>
	0-2	0-2
-1	5.5	21.7
	0-5	0-5
0	1.06	7.8
	0-2	0-2
+1	0.95	2.50

x-y Legend

x - Number of U-bends which cracked.
y - Number of specimens tested.
A - Not U-bends.

The pretreatment oxidizer step was varied from 75% to 125% of specification and the effect on the couples was determined. Both the 304 stainless steel/1020 carbon and 410 stainless steel/1020 carbon showed no sensitivity to this variation. Corrosion rates were under 20 mils/yr.

APPENDIX D

DEMONSTRATION COST DATA

APPENDIX D
DEMONSTRATION COST DATA

	<u>Page</u>
Included in Appendix D are:	
(1) Cost data sheets for 5 demonstrations	D-2
(2) Cost data sheets for construction	D-17
(3) Cost/Benefit calculation data sheets	D-18

COST OF DECONTAMINATION DEMONSTRATIONS
Thousand \$

							<u>Total</u>
Training							145
<u>Demo's</u>	<u>Syst</u> <u>Check</u>	<u>Strong</u> <u>Chem</u>	<u>Redox</u>	<u>NS-3</u>	<u>Mild</u> <u>Chem</u>	<u>Redox</u> <u>Mild</u>	
Manpower	150	137	163	228	168	168	1,014
Chem		13	1	16	3.5	3.5	37
Resin			20		20	20	60
Waste		47.2	35	49	40	40	211
Corrosion Testing							178
Post Insp.							220
Gen'l & Admin. Exp.							<u>110</u>
Total							1,975

Operator Training (Hiring/training)

Letdown system operation
and axiliary equipment

20 men x 30 days x 8 hrs. = 4800 mhrs

4800 x \$22 x 1.37 (escal.) = \$145,000

Pre-operational checkout

All decon equipment and
interconnecting systems

35 men x 142 hours = 4950 mhrs

4950 x \$22 x 1.37 (escal.) = \$150,000

	<u>Strong Chem</u>	<u>Dilute Chem</u>	<u>Redox</u>	<u>Redox/Chem</u>
<u>Manpower</u>				
Chem. Feed - Mix - Monitor	1	1	0	1
H - 0 Feed	0	0	1	1
Control @ pump & Htr.	1	1	1	1
H.P.'s	2	3	3	3
Monitor Instr. - Circulation	1	1	1	1
Monitor Chem. & Sple.	3	3	3	3
Analysts	2	2	2	2
Letdown recycle op.	0	3	3	3
Waste Drains op.	2	0	0	0
Waste Trtmt. Syst. op.	3	3	3	3
Rinse drains Ion Exch. op.	3	0	0	0
Ion Exch. Splr. - Analyzer	2	3	3	3
Pkg'g Syst. Op.	3	3	3	3
Waste Evap.	1	1	1	1
Supervision	<u>2</u>	<u>2</u>	<u>2</u>	<u>2</u>
	26	26	26	27

Manpower for demonstrations
 For actual application of proven method the
 manpower req'ts would be less.

DECON DEMONSTRATIONS

TIME REQUIREMENTS & LABOR COST

Citrox (or other strong chem.)

	<u>Hours</u>
DI waterheat-up	4
Fill circuit	2
Recirculation	6
Drain	3
Heat up solvent	2
Recirculate	36
Drain	3
Rinse fill	2
Rinse	3
Drain (DI)	10
Rinse sequence 2	15
Rinse sequence 3	<u>15</u>
	101
Waste Evaporation	7
Waste Packaging	<u>5</u>
	113
Associated time & Conting.	<u>113</u>
	226
22 x 202	4,450
4 x 24	<u>96</u>
	4,546
4,546 x \$22 =	\$100,000
Excal. 100,000 x 1.37 =	\$137,000

DECON DEMONSTRATIONS
TIME REQUIREMENTS & LABOR COST

<u>Redox</u>		<u>Hours</u>
DI water heat-up		4
Fill circuit		2
Add hydrogen		2
Recirculate		36
Rinse		3
Drain		10
Fill circuit		1
Add oxygen		2
Recirculate		36
Rinse		3
Drain		10
Transfer Resin		7
Waste Packaging		<u>10</u>
		126
Assoc. Time & Conting.		<u>126</u>
		252

<u>Men</u>		<u>Hrs.</u>		
23	x	232		5,350
3	x	20		<u>60</u>
				5,410
5,410	x	\$22	=	\$119,000
Escal. 119,000	x	1.37	=	\$163,000

DECON DEMONSTRATIONS
TIME REQUIREMENTS & LABOR COST

Dow Solvent NS-3

	<u>Hours</u>
Prepare Chem & fill loop	4
Vent	1
1st Stage circulation	24
2nd Stage add & heat	5
2nd Stage circulation	100
Drain	4
Fill rinse	2
Vent	1
Rinse (DI)	3
Drain	10
2nd rinse sequence	<u>16</u>
	170
Evap. of waste	7
Packaging of waste	<u>5</u>
	182
Associated time & conting. (100%)	<u>182</u>
	364

<u>Men</u>		<u>Hours</u>		<u>Manhours</u>
22	x	340		7,480
4	x	24		<u>96</u>
				7,576
7,576	x	\$22	=	\$167,000
Escal. 167,000 x 1.37 = \$228,000				

DECON DEMONSTRATIONS
TIME REQUIREMENTS & LABOR COST

Dilute Chemical

	<u>Hours</u>
DI water heat-up	4
Fill circuit	2
Recirculate	6
Drain (DI)	5
Heat solvent	2
3 Cleaning Cycles	48
Drain	10
Rinse fill	2
Rinse procedure	6
Drain	10
Rinse sequence 2	<u>13</u>
	107
Resin Processing	14
Waste Packaging	<u>10</u>
	131
Assoc. time & Conting.	<u>131</u>
	262
23 x 242 =	5,560
3 x 20 =	<u>60</u>
	5,620
5,620 x \$22 =	\$123,000
Escal. 123,000 x 1.37 =	\$168,000

Chemical Costs

Single Strong Chemical (ref. cost CitroX)

$$\begin{aligned} 4,000 \text{ gal.} \times 1/2 \text{ lb/gal} \times \$2.61/\text{lb} &= \$ 5,220 \\ \$5,220 \times 2.5 \text{ (Escal. \& Conting.)} &= \$13,000 \end{aligned}$$

NS-3

$$\begin{aligned} 4,000 \text{ gal.} \times \$3.00/\text{gal} &= \$12,000 \\ 12,000 \times 1.33 \text{ (Escal. \& Conting.)} &= 16,000 \end{aligned}$$

Redox

Nitrogen	8 cyl. @ \$20	\$160
Hydrogen	3 cyl. @ \$105	\$315
Oxygen	3 cyl. @ \$22	<u>\$ 66</u>
		\$541

$$\$541 \times 1.85 \text{ (Escal. \& Conting.)} = \$1,000$$

Dilute Chemical (ref. cost Candecon)

$$\begin{aligned} 4,000 \text{ gal.} \times 8.3 \text{ lb/gal.} \times .002 \times \$25 &= \$1,660 \\ 1,660 \times 1 \text{ demo} \times 2.1 \text{ (Escal. \& Conting.)} &= \$3,500 \end{aligned}$$

Resin Costs

Redox

$$50 \text{ lb Fe} \times 7,000 \text{ gr/lb} = 350 \text{ gr Fe}$$

$$350,000 \text{ gr/11,000 gr/CF resin} = 32 \text{ CF resin}$$

Utilize existing exchanger in blowdown purification system

-- 63 CF resin

$$63 \times \$50 = \$3,150$$

Utilize existing mixed bed

$$42 \times \$50/\text{CF cat.} = \$2,100$$

$$21 \times \$125/\text{CF anion} = \$2,620$$

$$\text{Total resin cost} = \$7,870, \text{ estimate } \$8,000$$

$$8,000 \times 2.5 = \$20,000$$

Dilute Chemical

Utilize same combination - \$20,000

Dilute Chemical (after Redox) 20,000

COSTS OF WASTE DISPOSAL

Calculation of activity yielded 140C /loop (date base 1/1/79)

Assume + 30% = 182C /loop

Use 182C/loop

Activity in drains

	<u>DF Best Case</u>	<u>C in drains</u>		<u>%Removal</u>
Citrox	10	182-18	164	90
Redox	3	182-60	122	67
Mild Chemical	4	182-45	137	75
NS-3	15	182-12	170	93

Wastes disposal costs are based on the following:

Fields equal 15 R/hr for a 55 gallon drum

Total Curies removed by decon are estimated and the number of drums required are calculated to meet the 15 R/hr. criteria. This governs the degree of concentration which will be accomplished by evaporation.

The cost are calculated on this basis:

Transport	\$151/drum
Burial	<u>\$ 79/drum</u>
Total	\$230/drum

Using the estimated decontamination effectiveness for each method under consideration, the radiation level of 15 R/hr. is converted to Ci per drum for an average shipment of 12 drums as follows:

$$R/hr = 6 Ci E/f$$

$$\text{Average energy, } E = 1.25$$

$$\text{Average distance from side of drum, } f = 1 \text{ ft.}$$

$$R/hr = \frac{6 Ci 1.25}{1}$$

$$R/hr = 7.50 Ci, \text{ \& for } R = 15 \text{ R/hr.}$$

$$Ci = 15/7.5 = 2$$

Redox - Waste Cost

$$\frac{122 \text{ ci}}{N \text{ drums}} = 2$$

$$N = 61$$

Cost

$$\$230 \times 61 = \$14,000$$

$$\text{Escal. \& conting. } 14000 \times 2.5 = \$35,000$$

Waste evaporation required

$$\begin{array}{r} \frac{122 \text{ Ci}}{V} \\ \frac{.6}{.6} \end{array} \quad \frac{1}{54}$$

$$\begin{array}{r} \frac{2}{.6} \quad \frac{V}{54} \\ \times \end{array}$$

Redox-Waste Cost - (Cont'd)

$$V = \frac{122 \times .6 \times 54}{2}$$

$$V = 1975 \text{ gallon}$$

Original volume of waste 4,000 gallons is concentrated to 1975 gallons.

$$\text{Evaporate } 25 \text{ M} = 4,000 - 1975 = 2,025$$

$$M = \frac{2,075}{25} = 81 \text{ minutes}$$

$$M = 1.42 \text{ hours}$$

An alternative is to pass decon solution through deionizers to remove radioactive products. The cost of waste disposal would be the same as given above since its based on acceptable field per drum. (same in each case)

Citrox - Waste Cost

$$\frac{164 \text{ Ci}}{N \text{ drums}} = 2$$

$$\frac{N}{d} = 82$$

Cost

$$\$230 \times 82 = \$18,850$$

$$\text{Escal. \& Conting. } 18250 \times (2.5) = \$47,200$$

Waste evaporation required

$$\frac{\frac{164}{V} \times \frac{1}{54}}{.6} = 2$$

Citrox - Waste Cost - (Cont'd)

$$V = \frac{164 \times .6 \times 54}{2}$$

$$V = 2,660 \text{ gallon}$$

At 25 gpm evaporation rate:

$$25 \text{ M} = 4,000 - 2,660 = 1,340$$

$$M = 1,340/25 = 54 \text{ minutes}$$

$$M = 54/60 = 0.9 \text{ hours.}$$

NS - 3 Waste Cost

$$\frac{170 \text{ Ci}}{N_d} = 2$$

$$N_d = 85 \text{ drums}$$

$$\begin{array}{l} \text{Cost} \\ \$230 \times 85 = \$19,550 \end{array}$$

$$\text{Esc. \& Conting. } 19,550 \times 2.5 = \$49,000$$

Waste evaporation required:

$$\frac{170}{V/0.6 \times 1/54} = 2$$

$$V = \frac{170 \times .6 \times 54}{2}$$

$$V = 2,760 \text{ gallon}$$

At 25 gpm evaporation rate:

$$25 \text{ M} = 4,000 - 2,760 = 1,240 \text{ gallon}$$

$$M = \frac{1,240}{25} = 49.6 \text{ minutes}$$

$$M = 0.83 \text{ hours}$$

Dilute Chemical - Waste Cost

$$\frac{137 \text{ Ci}}{\text{Nd}} = 2$$

$$\text{Nd} = 69$$

Cost

$$\$230 \times 69 = \$15,900$$

$$\text{Escal. \& Conting. } 15,900 \times 2.5 = \$40,000$$

Corrosion Testing

Testing assemblies (2)	\$ 75,000
Samples preparation	12,000
Samples examination & analysis (5 demos-12 sples)	24,000
Consultant	<u>19,000</u>
Total	\$130,000
Escal. (37%)	<u>48,000</u>
	\$178,000

Post Inspections

Eddy current 4 heat exchangers	\$160,000
Escal.	<u>60,000</u>
	\$220,000

Cost for Construction of Facility

Major Equipment	\$ 112,000
Piping, valves, etc.	92,200
Additional loops (3)	71,500
Sample conns. & instr.	<u>65,000</u>
	340,700
Labor	<u>400,000</u>
	740,700
A. Proj. Mg't 15%	110,000
B. Eng'g 16%	136,000
Payroll, Taxes & Pens. 32% of A & B	<u>78,800</u>
	1,065,500
Refurbishment	700,000
HP Support	130,000
QA Support	<u>52,000</u>
	1,947,500
A & S 5.88%	<u>115,000</u>
	2,062,500
Escal. 37%	<u>763,000</u>
	\$2,825,500

Cost/Benefit Calculation

If the anticipated exposure is 500 manrem and the outage time is 30 days in one quarter then:

$$\text{Number of workers} = \frac{500 \text{ Man Rem}}{1.25 \text{ R/quarter (limit)}} = 400 \text{ men}$$

Assume the rate of pay is \$25 per hour including overhead but contract workers are guaranteed 3 months pay if quarterly exposure allowance is exhausted:

If 200 men are contracted for the outage then:

$$C_1 = 200 \left(\frac{\$75}{\text{premium}} \times \frac{2}{3} \times 30 \times 8 \right) = \$2.4 \times 10^6$$

and a reduction in exposure would reduce this cost as follows:

$$\$2.4 \times 10^6 \times \frac{R}{1.25R} = \text{SAVINGS}$$

where $\frac{R}{A}$ is actual reduction in manrem exposure.

If the utility work force is 200 men and the exposure allowance is used up, the worker is assigned another job of less worth. Assuming the value lost is \$10/manhour, and the average man's allowance is filled in 15 days, then:

$$C_2 = 200 \left(\$10 \times \frac{1}{2} \times 30 \times 8 \right) = .24 \times 10^6$$
$$C_1 + C_2 = 2.4 \times 10^6 + .24 \times 10^6 = \$2.64 \times 10^6$$

A reduction in exposure reduces the cost as follows:

$$\text{SAVINGS} = \$2.64 \times 10^6 \times \frac{R}{1.25R}$$

and \$ per manrem is:

$$C_M = \frac{2.64 \times 10^6}{400 \times 1.25} = \$5,280/\text{manrem}$$

Assumptions:

Total Exposure for outage	500 Manrem
Outage time each year	30 days
Exposure limit per man	1.25 R/quarter
Total man hour rate	\$25/hr.
Total man hour rate for 30 days	
if quarterly exposure is used	
up (paid for quarter for 30 day	\$75/hr.
outage)	

Cost/Benefit-Alternative Calculation

Assume worker is paid for 2 weeks if exposure limit is reached prior to 2 weeks. Extremes are:

1. Assume worker reaches limit in 4 hours

Cost 80 hrs. x \$25	=	\$2,000
Value 4 hrs. x \$25	=	100
Radiation cost	=	1,900
Hourly cost $\frac{\$1,900}{4}$	=	\$475/hr.

2. Worker reaches limit after 80 hours

Radiation cost	=	0
Average cost	=	
$\frac{475 + 0}{2}$	=	\$237/hr.

If half the contracted workers fall in this category then the radiation cost =

$\frac{\$237}{2}$	x	160	x	200	=	\$3.78 x 10 ⁶
rate		hours		workers		

Other 200 men are utility men and are utilized as in 1st calculation, then

$$\text{COST} = \$.24 \times 10^6$$

Total =

$$\$3.78 \times 10^6 + \$.24 \times 10^6 = \$4.02 \times 10^6$$

and \$ per manrem is:

$$C_{\text{Malt}} = \frac{\$4.02 \times 10^6}{500} = \$8,000$$

Use lesser value of 2 methods, i.e $C_M = \$5,280/\text{manrem}$

Typical Calculation for Cost/Benefit (Strong Chemical)

Calculated Method

500 manrems

90% decon efficiency

Save 450 manrem

$$\text{Savings} = \$5,280/\text{manrem} \times 450 \text{ manrem} = \$2,376,000$$

$$\begin{aligned} \text{Benefit} &= \text{Savings} - \text{cost of decon} \\ &= 2,376,000 - 953,000 \\ &= \$1,423,000 \end{aligned}$$

NRC Method

500 manrem exposure

90% efficiency of decon

Save 450 manrem

$$\text{Savings} = \$1,000/\text{manrem} \times 450 \text{ manrem} - \$450,000$$

$$\begin{aligned} \text{Benefit} &= \text{Savings} - \text{cost of decon} \\ &= 450,000 - 953,000 \\ &= -\$503,000 \end{aligned}$$