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Radiological Assessment of BWR Recirculatory Pipe Replacement

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Pacific Northwest Laboratory
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ABSTRACT

Replacement of primary recirculating coolant pipe in BWRs is a major effort that has been carried out at a number of nuclear generating stations. This report reviews the planned or actual pipe replacement projects at six sites: Nine Mile Point-1, Monticello, Cooper, Peach Bottom-2, Vermont Yankee, and Browns Ferry-1. It covers the radiological issues of the pipe replacement, measures taken to reduce doses to ALARA, estimated and actual occupational doses, and lessons learned during the various replacements. The basis for the decisions to replace the pipes, the methods used for preparation and decontamination, the removal of old pipe, and the installation of the new pipe are briefly described. Methods for reducing occupational radiation dose during pipe repairs/replacements are recommended.

EXECUTIVE SUMMARY

Cracks in the boiling water reactor (BWR) primary recirculation system pipe have become a major issue in the continued operation of many power plants. Damage to the pipes is primarily caused by intergranular stress corrosion cracking (IGSCC) of the austenitic stainless steel. Most cracks are circumferential and occur in the heat affected zone of pipe welds. Concern that these cracks might lead to a pipe rupture and a loss-of-coolant accident has resulted in extensive inspection of BWR primary circulation system welds to determine the length and penetration of these cracks. Significant cracks are repaired primarily with weld overlays. The overlays, however, are viewed as a short-term solution and have to be inspected occasionally to ensure their continued integrity. Plants with large-scale cracking problems have already performed pipe repairs or replacements. Other utilities are in the process of planning repairs or determining whether repairs are necessary.

The extensive inspection and testing programs, as well as any repair actions, take place in relatively high radiation fields. Consequently, radiation protection measures are particularly important in controlling doses to the inspection and repair personnel. Licensees are currently using standard radiation protection measures but the search continues for improved methods to reduce occupational doses. The increased use of remote techniques in pipe repair has potential for reducing the radiological impact in the nuclear industry.

Plants that have completed major repair or replacement projects add to the accumulated experience and to the understanding of the problems that can arise during these nonroutine repairs. The success or failure of solutions they attempted will help other plants avoid the problems and resolve similar problems with less effort and lower dose. Dissemination of their experience with standard as well as site-developed dose reduction strategies will assist Nuclear Regulatory Commission inspectors and licensees in future repair and replacement projects.

This report focuses on the primary recirculation system pipe replacement at six nuclear plants. It reviews the extent of the replacement effort, dose reduction strategies used, measures used to control surface and airborne contamination, techniques for monitoring exposures, radiation doses to workers, and radioactive waste generated. In discussions with personnel from each of the six utilities, a number of important lessons they had learned from their particular experiences were elucidated. A summary of the site-specific experiences and techniques that were found to be successful in reducing dose are presented.

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ACKNOWLEDGMENTS

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1.0 INTRODUCTION

The discovery in the early 1980s of cracks in the primary coolant systems of boiling water reactors (BWRs) led to inspection of pipe welds in these systems. Intergranular stress corrosion cracking (IGSCC) was responsible for suspected widespread degradation of pipe welds, and the Nuclear Regulatory Commission (NRC) required plants to undergo extensive ultrasonic testing of these welds during plant outages. An unexpectedly deep circumferential crack in a residual heat removal (RHR) pipe weld discovered during one of these inspections prompted the acceleration of the inspection schedule. In July 1983, the NRC ordered the immediate shutdown and inspection of five nuclear plants because of the potential for major loss-of-coolant accidents resulting from the loss of pipe integrity in the reactor recirculation coolant systems. These five plants, Browns Ferry-3, Brunswick 2, Dresden-3, Pilgrim, and Quad Cities-2 were served notice that pipe inspections would be required within 30 days. After the NRC was convinced that the potential problem, although major in long-term importance, was unlikely to pose an unacceptable risk in the near term, the order was amended. The plants were allowed to continue operating but were asked to schedule their next refueling or maintenance outages to take place in the next several months.

IGSCC had previously been identified in other nuclear plant components and considerable study had been done to determine its causes and to develop metals more resistant to IGSCC degradation. The problem of IGSCC in recirculatory piping was first discovered at Nine Mile Point. Inspections of the initial five and additional BWR plants during the summer and fall of 1983 showed that the indications of pipe cracks in the primary recirculatory and residual heat removal piping were widespread and especially significant at Browns Ferry-1, Dresden-2, Hatch-2, Peach Bottom-2 and -3, Vermont Yankee, and Cooper. Fewer indications were noted at Browns Ferry-2, Hatch-1, and Brunswick-1, and one indication was found at Fitzpatrick (Inside NRC - November 14, 1983).

As a short-term repair, overlays were welded onto pipes with crack indications. The plants with extensive cracking began scheduling and planning for eventual pipe replacement. A primary pipe replacement involves a major effort and is dose intensive. In accordance with 10 CFR 50.59 (1984), each plant undergoing a primary pipe replacement must determine whether changes to the facility resulting from the replacement involve an unreviewed safety question. In the case of Nine Mile Point, Niagara Mohawk Power Corporation applied for a license amendment before performing the first major pipe replacement. Largely because of the experience gained at Nine Mile Point, the other utilities planning pipe replacement projects have justified their positions that the changes required would not constitute an unreviewed safety question.

Replacement of primary coolant piping along with associated systems is a major activity with significant radiological impact whether or not it entails changes in the operating license. The NRC, in its letter to licensees and applicants of BWRs providing procedural guidance on pipe replacement, anticipated that cumulative exposures could be limited to less than 2000 person-rem through good operational health physics practices (Eisenhut 1984). Even so, this quantity is considerably higher than the average outage dose.

Because of the high occupational doses resulting from pipe replacement, the NRC is interested in identifying effective methods and approaches that have been used in completed or ongoing pipe replacements to minimize occupational dose. Also of interest are those strategies that were not effective or where the benefit of radiation control measures was marginal. This report addresses the radiological impact of the pipe replacement program at six nuclear generating stations -- Nine Mile Point-1, operated by Niagara Mohawk Power Corporation; Monticello, operated by Northern States Power Company; Cooper, operated by the Nebraska Public Power District; Peach Bottom-2, operated by Philadelphia Electric Company; Vermont Yankee, operated by Yankee Nuclear Power Corporation; and Browns Ferry-1, a Tennessee Valley Authority plant. The first four of these sites have completed their repairs. Vermont Yankee is in a replacement outage, and Browns Ferry has postponed their pipe replacement.

2.0 CONCLUSIONS AND RECOMMENDATIONS

Concern that cracking around the circumference of pipes in BWR primary coolant piping might lead to a pipe rupture and a loss-of-coolant accident prompted the NRC to require extensive inspection of BWR reactor coolant piping, previously thought to be immune to IGSCC. The resulting inspections of the reactor cooling water (RCW) and residual heat removal (RHR) systems identified several plants where extensive indications of cracking were evident particularly in the heat affected zone (HAZ) of pipe welds. Other plants showed few or no such cracks. Those plants with deep or long cracks used weld overlays to cover the cracks and prevent leakage even if the cracks continued growing. These welds provided a short-term solution to the problem and allowed continued safe plant operation. However, because the weld overlays made further inspection difficult, crack growth could not be easily monitored. A permanent fix was needed to avoid further frequent and dose intensive inspections.

Most of the plants with numerous significant cracks chose to replace their RCW and RHR piping and associated systems. Their original Type 304 or 316 seam-rolled pipe was fabricated in segments requiring numerous welds at bends. This high-carbon stainless steel is being replaced with Type 304 or 316 nuclear grade (NG) low-carbon stainless steel, which has shown much greater resistance to IGSCC. Much of the new pipe installed to date has been seamless and prebent to reduce the number of welds needed and eliminate sites for future IGSCC attack. Some plants are opting for electropolishing and/or preoxidation treatment of the new pipe to reduce the sites or the rate of contamination buildup. Additionally, hydrogen additive water chemistry has been implemented in several plants to better control oxygen levels and conductivity.

The pipe replacements at six utilities are addressed in this report. Nine Mile Point-1, Monticello, Peach Bottom-2 and Cooper nuclear plants have completed their replacements. The Vermont Yankee replacement will be completed in mid-1986. Browns Ferry, the sixth plant, has postponed its pipe replacement project.

Replacement of the RCW system and associated piping, seals, and safe-ends is a dose intensive undertaking. The NRC anticipated that pipe replacement projects would keep to a maximum of 2000 man-rem and planned to further review efforts which would exceed this level. The six plants adopted several exposure reduction strategies to keep doses below this level and to provide increased radiation protection for their workers. Three major steps that have been highly successful in reducing exposure rates are 1) using extensive shielding in high-exposure-rate and high-traffic areas, and 2) chemical decontaminating the RCW, RHR, and reactor water cleanup (RWCU), and perhaps the associated valves and pumps, and 3) using remotely controlled cutting, grinding/milling, and welding equipment.

Because of the small size of the BWR drywells and general lack of clearance, shielding designs required considerable engineering. However, shielding, especially of hot spots, was successful in reducing exposure rates. General drywell decontamination to reduce incidents of personnel contamination was widely used.

Pipe decontamination helps reduce exposure rates significantly in the drywell, especially in the areas near the RCW areas systems. Three types of chemical decontamination solutions were used. Can-decon®, LOMI, and a combination of Citrox and AP Citrox. These processes removed much of the corrosion and fission product contamination in the systems being decontaminated. Two problems, however, prevented complete success. The first was a problem of isolating the decontamination solutions. Nozzle dams were prone to leak, sometimes causing incomplete decontamination or leakage of the solutions into areas not intended for decontamination. The second problem arose in the mechanical or chemical decontamination of the recirculating pumps. Most plants experienced problems in reducing the exposure rates from the pumps. This appears to be a two-fold problem resulting both from insufficient contact or recirculation of the decontamination solutions (causing insufficient sludge removal) and from a different oxide film composition in the pumps that is less soluble in decontamination solutions. Very poor dose reduction factors were achieved in pumps subjected to chemical decontamination. Peach Bottom-2, in particular, had hot spots in their pumps that eventually required pump disassembly and water/glass-bead slurry hydroblasting to decontaminate them sufficiently. Cooper decided to avoid the problem by shielding rather than decontaminating the pumps. However, isolating and shielding the pumps proved overly difficult as well.

Another major strategy to reduce dose and improve the quality of the operations was the use of remotely controlled automatic cutting, grinding/milling, and welding machines. While these machines require setup time and an operator nearby to monitor the progress and keep cords from becoming tangled, the operator is no longer required to spend as much time in close proximity to the hot pipe. Most of the utilities believed that the automatic machines were responsible for some dose savings.

Each utility had an ALARA coordinator or supervisor and an organization dedicated to ensuring that good health physics practices and dose reduction strategies were considered in the task planning and implementing stages. The organizations were made up of representatives from utility ALARA and health physics (HP) staffs, contractor staff, and utility management. These organizations wrote and reviewed procedures for keeping doses ALARA and helped resolve technical problems with dose intensive tasks. With the assistance of the utility HP staff, personnel were trained for drywell jobs, contamination was reduced, and worker doses were monitored.

A summary of the six pipe replacements studied in this report is given in Table 2.1. The reactors were built by General Electric. NMP-1 is a BWR-2, five-loop system. Monticello is a BWR-3. The Cooper, Peach Bottom, Vermont Yankee, and Browns Ferry plants are BWR-4s. All six plants have Mark-1 containments.

The outage lengths depended on other non-pipe-related activities as well as the extent of the replacement and the progress of replacement activities. Chemical decontamination was performed using Can-decon® at NMP-1 and Peach Bottom-2. LOMI was used at Monticello. Cooper and Vermont Yankee used a Citrox/AP Citrox combination.

TABLE 2.1. Summary Data of Pipe Replacements

Pipe Replacements	MWe	Began Commercial Operation	Primary Loops	Total Outage Dates	Type of Decontamination	Occupational Dose (man-rem) Decontamination	Pipe Cut/Weld	Total
Nine Mine Point-1	610	12/69	5	8/82-7/83	Can-decon®	6.5	814	1,464
Monticello	536	7/71	2	2/84-1/85	LOMI	34	1,076	1,583
Cooper	778	7/74	2	9/84-8/85	Citrox	12.1	861	1,636
Peach Bottom-2	1,065	7/74	2	4/84-6/85	Can-decon®	38	1,364	1,895
Vermont Yankee	514	11/72	2	9/85-5/86 est	Citrox	70.2	844	1,647
Browns Ferry	1,067	8/74	2	postponed	(Can-decon® planned)	10 est	1,030 est	1,780 est

® Can-decon is a registered trademark of London Nuclear Limited.

Doses resulting from the various activities are shown in the table and are listed in more detail in Section 6.3. This summary shows the occupational doses incurred resulting from the decontamination and pipe cutting and welding activities. Total doses for the entire pipe replacements are also given. Additional tasks such as preparation, cleanup, supervision, and induction heat stress improvement (IHSI) are included in the totals shown in the last column. Doses incurred at the various plants cannot be directly compared because of the different dose tracking practices used. Some plants included only those tasks that were directly related to the pipe replacement in their dose totals. Others included tasks that were not strictly a part of the replacement efforts, but part of ongoing outage activities. For example, the inclusion of IHSI on the new pipe welds is a preventative measure against future cracking that several but not all utilities included in their respective total dose numbers. The official total doses for the four completed replacements and the estimates for the remaining two are all under the recommended 2000 man-rem maximum largely because of dose reduction strategies used by the utilities.

Radwaste from pipe replacements consisted of resins from the decontamination process and the removed pipe in addition to routine plant waste. The resins were shipped to a low-level waste burial site. The pipe was shipped to Pacific Northwest Laboratory, Brookhaven National Laboratory, or Oak Ridge National Laboratory for study or to a vendor for further decontamination and reclamation as scrap metal.

The accumulation of experience with primary coolant pipe replacements in BWRs has led to the identification of numerous effective strategies for dose reduction. Not all have been applied with equal success at each plant, emphasizing the need for tailoring the project for the specific site. Some of the most successful of the dose reduction recommendations are highlighted below. The first four are true of most major nuclear projects. The next eight are specific to pipe replacements.

1. Extensive and careful planning is essential to smooth replacement operations. Thorough planning was repeatedly stated as being of great importance and lasting significance. The inclusion of ALARA and HP staff in the planning and procedures development from the beginning and during the entire outage greatly adds to efficient exposure control. Communication between planners and personnel from utilities that had completed pipe replacements proved valuable.

2. Training on full-sized mockups in simulated conditions with systems to be worked on in the drywell allowed workers time to become accustomed to radiation protection procedures and to gain experience with appropriate tools and machinery on actual piping. Mockup training was especially helpful for those preparing to work around the safe-ends. Mockups also proved worthwhile in testing techniques and identifying workers whose speed and accuracy were not satisfactory.

3. Using remote TV cameras in the drywell has two related benefits. It reduces the number of people needed in the very crowded drywell, and it allows personnel outside the drywell to monitor inside activities without receiving dose.

4. Contract agreement between the utility and the primary contractor need an incentive to keep doses low and force a more efficient utilization of workforce. Otherwise, conflicts occur when ALARA concerns interfere with contractor procedure or progress timing. The number of contractors also influences the quality of work and quantity of personnel exposure. Good organization and lines of authority help keep time in the drywell and associated doses to a minimum.

5. Although there may occasionally be cases where cutting and removing pipe from the drywell without decontaminating it first may provide a dose savings, generally, and especially with large pipe, chemical decontamination results in a large dose savings.

6. Satisfactory decontamination of recirculating pumps is difficult, and the strategy chosen must be carefully considered. Chemical decontamination generally does not sufficiently reduce hot spots where chemical circulation is poor or the buildup is insoluble. Specific flushing of systems or partial disassembly followed by mechanical decontamination techniques may be necessary. Avoiding pump decontamination entirely while still providing sufficient shielding is also difficult.

7. Most plants found that the water level in the vessel was not as critical as they expected it to be in providing shielding. Area dose levels increased only slightly when water levels were lowered to avoid leakage into the annulus to do vessel work. Filling of the control rod guide tubes, however, was helpful in providing some shielding.

8. The decision to keep and refurbish or replace valves should be based on expected ALARA benefit as well as the cost of replacement.

9. The use, where possible, of remotely operated cutting, grinding, and welding equipment is recommended as a dose saving measure. The selective use of manual plasma arc cuts may also provide a dose savings in cases where the exposure rates have been sufficiently reduced.

10. Joints to be welded should be closely examined for material crud before they are welded. Many inspection tests falsely revealed evidence of inclusions in new welds; the inclusions turned out to be just crud.

11. IHSI of welds is highly recommended to avoid the possibility of future cracks or problems.

12. To help reduce the number of welds, which consequently reduces future inspections and sites for IGSSC attacks, new piping should be seamless, prebent Type 304 or 316 NG low-carbon stainless steel.

The new NG pipe is expected to be much more resistant to IGSCC and may prevent it throughout the remaining lifetime of the plant. However, techniques to improve this resistance by stress-related (IHSI) or environment-related (H_2WC) measures may add an additional margin of assurance of continued pipe integrity.

3.0 BACKGROUND

The first documented evidence of IGSCC in operating BWR plants occurred in late 1965 (O'Toole and Gordon 1984). Between 1965 and 1971, sporadic cracking occurrences were reported in small (4- to 6-inch) diameter Type-304 stainless steel pipes in the recirculation bypass and reactor water cleanup (RWCU) piping systems. As shown in Figure 3.1, by 1974 the number of IGSCC incidents had increased considerably. In late 1974 and early 1975, the NRC shut down all the domestic operating BWRs to inspect and investigate the cause for the cracking. However, no safety implications associated with the protection of the public were identified. Up until 1980, all the pipe crack failures were in the heat affected zones (HAZs) of Type-304 stainless steel. The first incidence of cracking in higher carbon, Type-316, stainless steel in the field was in March 1980. In 1982, extensive IGSCC in the large diameter, Type-316 stainless steel at Nine Mile Point-1 led to the first domestic recirculation piping replacement, which involved replacing the 28-inch diameter piping and all the safe ends with Type-316 nuclear grade (NG) material. Since then, recirculation piping replacements have been completed at five other operating nuclear plants: Hatch-2, Monticello, Peach Bottom-2, Cooper, and Pilgrim. Vermont Yankee and Dresden-3 are currently completing pipe replacements.

By 1983 about half of the operating BWRs reported that less than 10% of their inspected welds showed indications of IGSCC (DeYoung 1983). The BWRs reporting indications on greater than 10% of their welds included Nine Mile Point-1, Dresden-2, Hatch-2, Peach Bottom-2 and -3, Vermont Yankee, Browns Ferry-1, and Cooper. The highest incidence of IGSCC was found at NMP-1, where approximately 85% of the recirculation system weld HAZs exhibited reportable indications of cracks. Shop welds and field welds were equally affected. A summary of the BWR data on IGSCC incidence is given in Table 3.1.

Since the investigation of the cause of cracking began, it has been established that there are three major factors contributing to the development of IGSCC. These factors are a sensitized material, a high tensile residual welding stress, and a corrosive environment (Itow, Saito and Sato 1984). Sensitization results from the heating and subsequent cooling that occurs during joining of the pipes via arc welding. This process creates a metallurgical change in the stainless steel adjacent to the weld seam that causes the steel to be more susceptible to IGSCC (JAJ 1982). This change involves the formation of chromium carbide particles along the grain boundaries and the accompanying depletion of chromium from the adjacent austenite matrix (Itow, Saito and Sato 1984). The high tensile stress contributing to the development of IGSCC is the result of welding and postweld grinding. A corrosive environment also supports the formation of IGSCC. Such an environment is due to a combination of dissolved oxygen, ionic impurities, and high temperatures. A dissolved oxygen concentration of approximately 0.2 ppm in normal operating temperatures showed adverse effects at temperatures above 100°C (Itow, Saito and Sato 1984).

TABLE 3.1. Incidence of Reportable Indications of IGSCC at U.S. Operating BWRs in the United States (DeYoung 1983)

Plant	Percentage of Inspected Recirculation Pipe Welds with Reportable Indications
Quad Cities-1	0
Millstone-1	0
Duane Arnold	0
Monticello	4
Browns Ferry-2	4
Oyster Creek	6
Brunswick-1	8
Hatch-1	9
Cooper	13
Peach Bottom-3	15
Dresden-2	20
Hatch-2	32
Peach Bottom-2	33
Vermont Yankee	58
Browns Ferry-1	67
Nine Mile Point-1	85

IGSCC is usually identified using nondestructive examination techniques, primarily ultrasonic testing (UT). UT examinations, both manual and automatic, have improved in recent years and are fairly successful in identifying indications of cracking. However, certain material and welding flaws that do not diminish the weld integrity are identified as crack indications.

Several other methods to identify pipe cracks are under development or are in the field-testing stage. Two techniques in particular are currently undergoing field testing and have important advantages over UT. The acoustic emission (AE) test method operates continually while the plant is on line. The synthetic aperture technique of ultrasonic testing (SAFT-UT) produces highly detailed, three-dimensional descriptions of cracks. This technique is particularly useful in identifying cracks under weld overlays (Inside NRC - August 19, 1985). It also requires less worker exposure than conventional UT methods. New methods such as these that improve the accuracy in crack identification may be applied in the future to reduce the number of false positive crack indications.

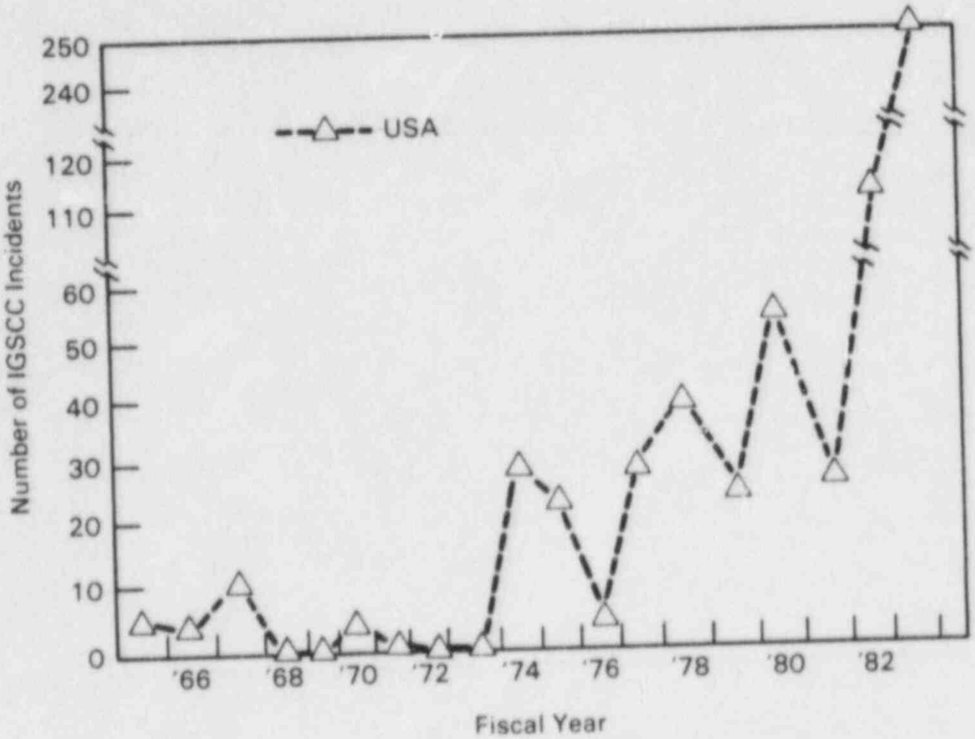

FIGURE 3.1. IGSCC Incidents in the United States (Itow, Saito and Sato 1984)

NRC's guidelines for inspection of stainless steel welds of piping susceptible to IGSCC include pipes equal to or greater than 4 in. diameter, in systems operating over 200°F, that are part of or connected to the reactor coolant pressure boundary out to the second isolation valve (Eisenhut 1984). This includes:

- all unrepaired cracked welds
- inspection of 10% of the welds in each pipe size of IGSCC-sensitive welds not previously inspected and reinspection of 20% of the welds of each pipe size not previously found to be cracked
- inspection of all weld overlays where circumferential cracks longer than 10% of the pipe circumference were previously measured
- welds treated by IHSI but not post-treatment UT acceptance tested
- expanded inspections when new cracks are found or old cracks show significant growth.

General guidance given on the cracks exempts from repair those flaws (indications) less than about 10% of the wall thickness. These are acceptable for further operation without repairs. Cracks greater than 30% of the circumference and cracks with a reported depth of 25% or greater of the thickness would probably need some form of repair. Cracks of intermediate thicknesses would require further evaluation to be acceptable without repair.

4.0 REPAIR METHODOLOGIES

Studies have shown that the initiation and growth of IGSCC involves a combination of processes occurring simultaneously in the metal, in the water environment and at the metal-environment interface (Jones 1984). Studies have also indicated that the approaches that can be taken to reduce the problems caused by IGSCC fall into one of three categories: material-related, stress-related, and environment-related approaches (Smith 1984).

4.1 MATERIAL-RELATED REMEDIES

The susceptibility of Type-304 or Type-316 stainless steel material is related to the welding process used during fabrication (Smith 1984). Chromium carbide precipitation occurs in a small zone of material adjacent to the weld fusion line as the temperature is raised during the welding process. As the carbides precipitate, the overall chromium composition is reduced in a narrow band, the HAZ, which is adjacent to the carbides. This area is susceptible to IGSCC. Material related remedies are aimed at eliminating or minimizing IGSCC susceptibility of the pipe material by altering the sensitization of the material. These methods include obtaining alternative piping material, solution heat treating (SHT) the pipe, applying a corrosion resistant cladding (CRC), and using a mechanical compression device.

4.1.1 Alternative Piping Material

The best known replacement pipes fabricated to avoid cracking problems are Types 304 NG and 316 NG stainless steel. Seamless, blast furnace pipe with bends rather than welds reduces sites for the formation of IGSCC in BWR main reactor coolant water lines. Frequently the replacement piping is imported from Japan or Germany because it is superior to the grade of stainless steel available in this country. Qualification tests on welded 4-in. diameter pipe specimens of NG 304 and 316 pipe demonstrated an increase in the life of the pipe by greater than a factor of 20 for both alloys in comparison to reference pipe specimens of Type-304 stainless steel (Jones 1984). Verification tests on 16-in. diameter welded pipe specimens were also successfully completed, indicating that the NG large diameter pipes would offer superior IGSCC resistance.

Solution heat treating the joints after welding has been shown to improve the life of welded 4-in. diameter pipe by a factor of 20 as compared to as-welded Type-304 stainless steel (Jones 1984). However, SHT is limited to shop welds, and the water quench, which is an integral part of the SHT process, can be difficult to perform properly on piping components with complex shapes.

4.1.2 Corrosion Resistant Cladding

The third sensitization-related remedy is the application of corrosion resistant cladding (CRC) to the inner surface of the pipe adjacent to the girth weld location. This protects the piping from IGSCC by providing a corrosion resistant stainless steel boundary layer over the girth weld HAZ where IGSCC is typically located (Findlan 1984). This CRC provides a barrier between the coolant and the HAZ, and it takes advantage of the established corrosion resistance of weld metal having a duplex microstructure of austenite and ferrite (Smith 1984).

CRC methods may be applied in the shop or in the field. In the shop CRC procedure, a solution heat treatment is performed before the final closure weld. This eliminates the new HAZ that is created by the cladding operation at the end of the CRC deposit. Thus there is no sensitized material in contact with the coolant after welding. Use of the shop CRC procedure in the preparation of pipe test specimens led to a factor of greater than 20 improvement in the life of the pipe compared with conventionally welded pipes.

It is not possible to solution heat treat the material in the field applications of CRC. Consequently, a new zone of sensitization is formed at the inner pipe surface. However, the procedure minimizes the heat input from welding, and the location of this new HAZ is outside of the high tensile residual stress field that is associated with the girth weld (Smith 1984). Application of the field CRC procedure during tests on welded 4-in. pipe specimens resulted in an increase in the life of the pipe by a factor of 6.5 compared to conventionally welded specimens (Jones 1984).

4.1.3 Pipelock Devices

Mechanical devices have been developed that may offer an alternative to weld overlays. "Pipelock" devices, which compress pipes with IGSCC to retard further cracking, are currently being tested. If these pipelocks are accepted, they will provide two important advantages. They can be removed to allow easy pipe access for UT, and they could potentially reduce or eliminate further crack growth (Nucleonics Week-July 12, 1984).

4.2 STRESS-RELATED REMEDIES

A number of stresses are normally considered in the design of a pipe joint. These stresses include primary stresses from the water pressure and pipe and water weight, secondary stresses from thermal expansion, and peak or local stresses associated with stress concentrators. These stresses are limited under normal operating conditions to values well below the yield stress. However, in the weld HAZ, there are also residual stresses caused by fit-up, cold work, and welding. These residual stresses may lead to local stress levels that exceed the yield stress of the material (Jones 1984).

The stress related remedies are developed to control the residual stresses. The weld-induced residual stress in the HAZ of Type-304 stainless steel is generally tensile at the inner surface of the pipe (Jones 1984). The stress improvement methods improve the resistance of IGSCC by making these residual stresses compressive at the susceptible weld HAZ. The four types of residual stress improvements are heat sink welding (HSW), last pass heat sink welding (LPHSW), induction heating stress improvements (IHSI) and weld overlay repair (WOR). The latter method is considered to be an interim remedy.

4.2.1 Heat Sink Welding

Heat sink welding is a welding process where the root pass and the first two or three layers are welded conventionally. Water is added into the pipe and the remaining layers are welded with relatively high heat input. Upon pipe cooldown, the inner diameter of the material is placed into residual compression, which must be overcome by a sufficient applied tensile load before IGSCC is possible. A factor of improvement between 4.5 and 14 has been validated even under severe loading conditions (Smith 1984).

4.2.2 Last Pass Heat Sink Welding

Last pass heat sink welding is similar to HSW except that it applies to completed welds. Last pass heat sink welding involves remelting the weld crown while providing a heat sink inside the pipe. In the LPHSW procedure, a high heat input is applied to the outside of the pipe weld, while the inside is cooled with flowing room-temperature water. Upon cooldown, the inner pipe surface is axially compressed to produce a favorable residual stress distribution that resists IGSCC (Olson et al. 1984). Stress tests conducted on 4-in. schedule 80 pipes demonstrated a factor of improvement of approximately 6.5 over the reference tests (Herra, Horn and Offer 1984).

4.2.3 Induction Heating Stress Improvement

The method of IHSI improves the resistance to IGSCC by producing compressive residual stresses at the susceptible weld HAZ. Compressive stresses are produced by induction coil heating the outer surface of a pipe weldment while simultaneously cooling the inner surface with water (Figure 4.1). The application of this process is directed at Types 304 and 316 stainless steel piping systems now in BWR plants where other improvement methods are not practical (Offer 1984). Environmental pipe test data show a factor of improvement of greater than 15 for IHSI-treated pipes, relative to IGSCC-susceptible reference heats of Type-304 stainless steel (Offer 1984).

4.2.4 Weld Overlay Repair

The application of a weld overlay on the outer diameter of a pipe weld is used to mitigate the effects of IGSCC (Newell 1984, Smith 1984). Weld overlay repair is used as an interim repair method to extend the life of welds exhibiting IGSCC. Duplex weld deposits are used to cover the existing weld and the HAZ (Figure 4.2). This provides a barrier to through-wall crack propagation and restores the structural integrity of the pipe to comparable uncracked levels. It also introduces a strong compressive residual stress field to the inner diameter of the pipe. Before mid-1983, WOR was considered as a viable permanent fix. Recent UT sizing difficulties with IGSCC indications cast doubt on the acceptability of WOR applications. It is, however, a temporary

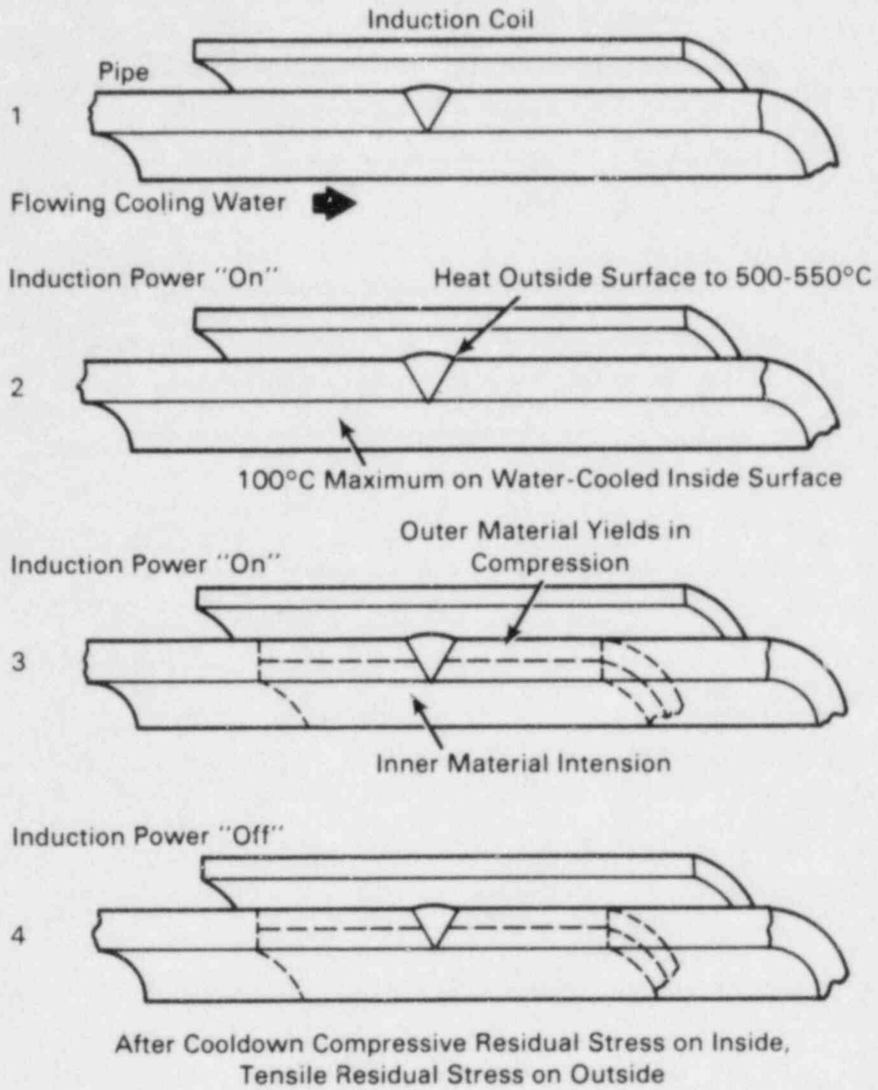

FIGURE 4.1. Induction Heating Stress Improvement Process

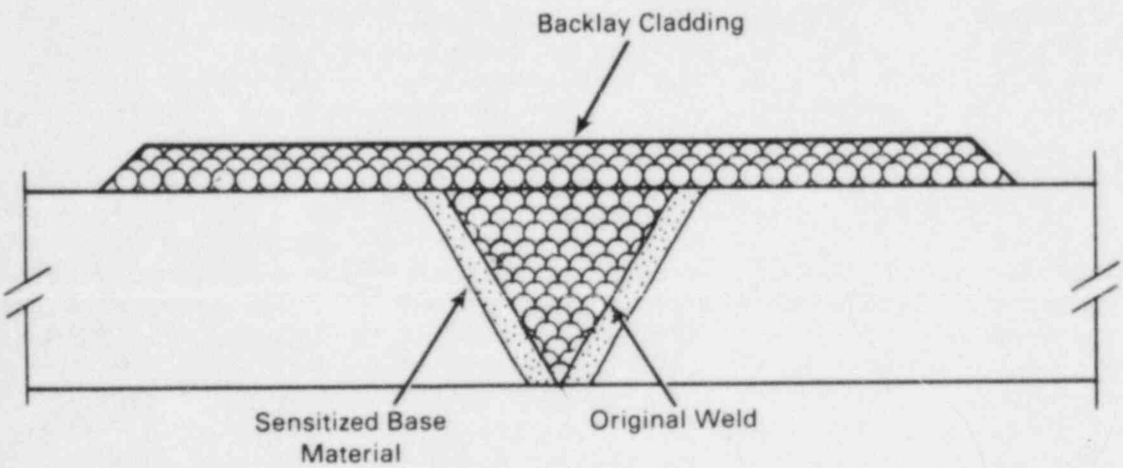

FIGURE 4.2. Weld Overlay Repair

repair method, and advances in UT sizing of IGSCC may make WORs an acceptable long-term repair in the future (Newell 1984). The NRC views weld overlays as an alternative to pipe replacement on a cycle-by-cycle basis only but will consider requests in certain cases for extended operation of BWRs with weld overlays.

4.3 ENVIRONMENT-RELATED REMEDIES

Environment-related remedies have the potential of protecting the entire cooling system instead of just a specific pipe weld (Jones 1984). Studies have shown that IGSCC is promoted by high levels of dissolved oxygen and by ionic impurities that are found in the primary coolant in BWR plants. The oxygen results from radiolysis in the reactor core, which decomposes a small amount of water into free oxygen and hydrogen. The ionic impurities enter the water from a variety of sources.

Four methods are used to reduce the potential for IGSCC formation by adjusting the oxygen and ion composition of the cooling water. These methods include startup deaeration, conductivity control, oxygen suppression during power operation, and hydrogen water chemistry.

4.3.1 Startup Deaeration

During reactor startup, oxygen undergoes radiolytic conversion to H_2O_2, which subsequently decomposes to H_2O and O_2 at higher temperatures (Andersen 1984a). Startup deaeration is practiced in a number of U.S. plants and is routinely used in Japan and Sweden. Laboratory experiments have shown that deaeration definitely mitigates IGSCC initiation and growth in simulated startup environments (Jones 1984, Andersen 1984a, Hale and Pickett 1984). Although startup deaeration will not totally prevent IGSCC, it does provide protection during startup and shutdown procedures and should be considered as part of a total remedy package.

4.3.2 Oxygen Suppression During Power Operation

The results of tests indicate that most of the damage leading to pipe cracking problems occurs during steady-state power operations (Jones 1984). IGSCC can be suppressed if the oxygen concentration of the water can be reduced below about 200 ppb. One possible method for decreasing the oxygen content of the reactor water is to inject hydrogen into the feedwater, which suppresses the in-core radiolytic oxygen formation.

4.3.3 Conductivity Control

The water in BWRs contains small quantities of ionic impurities (dissolved salts), which enter from a variety of sources. Some impurities such as Cl^- induce stress corrosion cracking under circumstances where it would not normally occur under static loading conditions (Andersen 1984b). Most impurities (H^+, SO_4^{--}, CO_3^{--} and F^-) accelerate IGSCC. By lowering the ionic impurity levels as low as possible, the rate of IGSCC progression will be minimized and the lifespan of the pipe will be maximized.

4.3.4 Hydrogen Water Chemistry

Hydrogen water chemistry (H_2WC) combines oxygen suppression via hydrogen addition to the feedwater with conductivity control (Roberts, Jones and Naughton 1984). Tests on welded pipe specimens showed an improvement factor greater than 18 in crack initiation time under H_2WC conditions compared with normal BWR water (Jones 1984). The use of H_2WC does adversely affect the radiation levels in the plant turbine buildings and the plant environs because of a change in the equilibrium level resulting in higher ^{16}N activity in the turbine building. The extent of concern depends on the specific plant (Jones 1984).

4.4 SUMMARY

None of the remedies listed will totally eliminate the possibility of IGSCC. They may however, increase the expected lifetime of the piping and can be used to decrease the possibility of the formation of IGSCC. Various techniques have been adopted by different plants. The adoption of several of the techniques, particularly IHSI and good water chemistry, is becoming common and is expected to decrease the possibility of IGSCC even further.

5.0 DESCRIPTION OF PIPE REPLACEMENTS

When the pipe cracking is extensive and the use of repair methods is unacceptable or cost prohibitive, replacement of the pipe may be required. Replacement of the primary recirculating system and associated piping is a major undertaking necessitating considerable personnel exposure to radiation. Two large-scale procedures are generally taken to help reduce drywell dose rates. These include extensive shielding and decontamination of the pipe.

Pipe Shielding

The dose intensive tasks associated with pipe repairs include inspecting, removing, and welding pipes and dealing with unexpected problems in containment. The highest levels of radiation are found particularly at the carbon and stainless steel pipe junctions. Shielding is a problem because of the crowded conditions in containment and the configuration of the pipe at the hot spots. Little insulation or shielding space is available at the nozzles. The dose rate, primarily from shine when the nozzle is cut, can be up to 20 R/hr so specialty shielding is sometimes designed and installed to reduce exposure levels. Welding techniques also require that insulation be stripped off the pipe near the pipe junction, which increases nearby dose rates. Remote cutting and welding equipment may help reduce personnel exposure to these fields but generally cannot be used for the critical welds or cuts where the geometry prevents proper equipment setup.

Pipe Decontamination

Pipe decontamination can be very effective for reducing dose to workers in the drywell. Although tests are being continued to determine whether certain decontamination solutions may potentially promote IGSCC under operating conditions, the general process has wide support. Decontamination solutions are especially effective in areas where good chemical circulation can be attained. In crud traps and areas of poor solution circulation, mechanical means such as vacuuming and hydrolasing (also known as hydrolancing or hydroblasting) may provide better dose rate reduction.

Three chemical decontamination solutions were used in the five replacements. NMP-1 and Peach Bottom-2 used the London Nuclear Services, Inc. Can-decon® treatment; Monticello used the Quadrex low-oxidation metal ion (LOMI) solutions; and Cooper and Vermont Yankee used Pacific Nuclear's Citrox process.

The Can-decon® process uses a low-temperature dilute decontamination solution of acidic organic chelating agents (pH 3.5 - 4.5) plus excess hydrazine for oxygen scavenging. The solution is circulated through the recirculation system and then passes through a system of purifying filters and

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ion exchange columns. The loose crud removed by the solution from the recirculation system is deposited on the filters while an ion exchange cation bed removes iron, cobalt, nickel, chromium, molybdenum, and any other heavy metals that may be present.

The LOMI process used at Monticello reduces the oxidation state of iron, the predominant ion in the corrosion products of both BWR and PWR primary systems (Munson, Card and Divine 1983). The reduction process destablizes the iron so that it becomes readily soluble. Vanadous picolinate is used to reduce Fe(III) to Fe(II) enabling it to be easily dissolved in picolinic acid (a mild chelating agent). Heaters are used to maintain the process temperature at 80-90°C. Reagents are added to the coolant and pumped into the system in solution. The reagent and dissolved radioactivity are removed from the system by diversion of the flow through ion exchange columns at the end of the circulation period.

Alternate procedures use preoxidation; however, the outage time schedule prevented the planned preoxidation step at Monticello. London Nuclear Services, Inc. decontaminated Dresden-3 with a LOMI process.

A third decontamination process used to reduce exposure rates in the drywell was the Citrox process performed at Cooper and at Vermont Yankee. The Citrox solution contains oxalic acid, dibasic ammonium citrate along with ferric nitrate or ferric sulfate and an inhibitor (Munson, Card and Devine 1983). Citrate ions form complexes with iron and inhibit precipitate formation.

An alkaline permanganate (AP) solution is frequently used in combination with the Citrox solution as it was at the Cooper plant and at Vermont Yankee. The AP solution consists of approximately 10% NaOH and 3% $KMnO_4$ (by weight). It is administered as the first step in the decontamination process, with the Citrox solution being administered as the second step. A typical treatment involves the application of AP at 105°C for four hours, a rinse with a dilute Citrox solution (to lower the pH below 10), application of concentrated Citrox at 80°C for eight hours, and administration of final rinses (Munson, Card and Divine 1983).

Pipe Replacement Sequence

The general sequence to the reactor coolant water pipe replacement projects involves extensive planning, removal of interfering pipe and equipment, decontamination of RCW and auxiliary piping, severing and removing the old pipe, prepping the joints and welding the new pipe, and replacing the interfering equipment. Tasks such as drywell decontamination and placement and removal of shielding continue throughout the projects.

The extent of pipe replacement and the type and extent of decontamination differed with each plant as did the extent of machine cutting versus manual cutting. Other site-specific features, such as dose rates at various plant locations, contributed to slightly different engineering and health physics procedures at the six sites.

Brief descriptions of the pipe replacement processes are contained in this section. The amount and type of information varies with the plant. In those cases where a great deal has been written, there is more to discuss than in those cases where the utility personnel have not completed their own evaluation. Additionally, utilities planning replacements studied previous replacements and were frequently able to take advantage of accumulated experience.

Typical two-loop BWR recirculating systems are illustrated in Figures 5.1 and 5.2. Nine Mile Point-1 (NMP-1), where the first pipe replacement occurred, is discussed in greater detail and in some ways provides the prototype for the

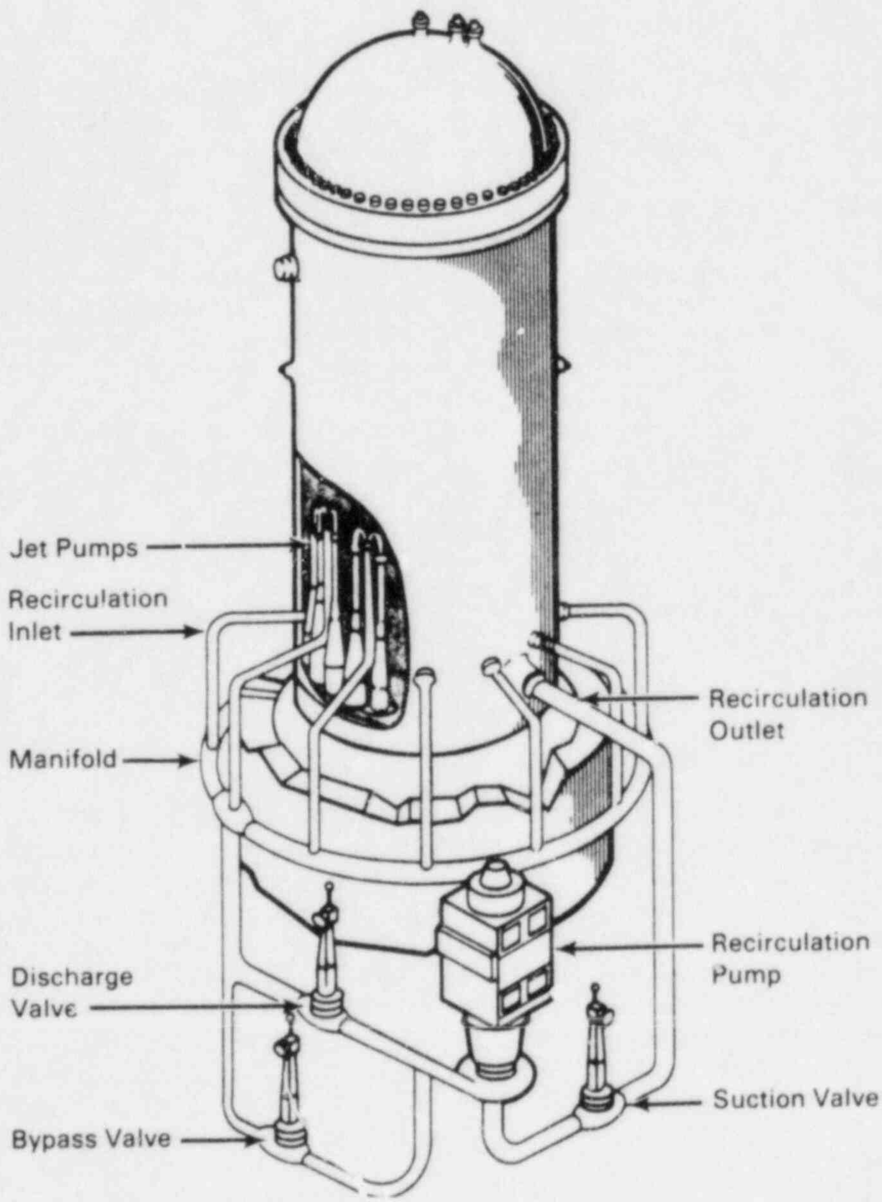

FIGURE 5.1. Typical Two-Loop BWR Recirculating System

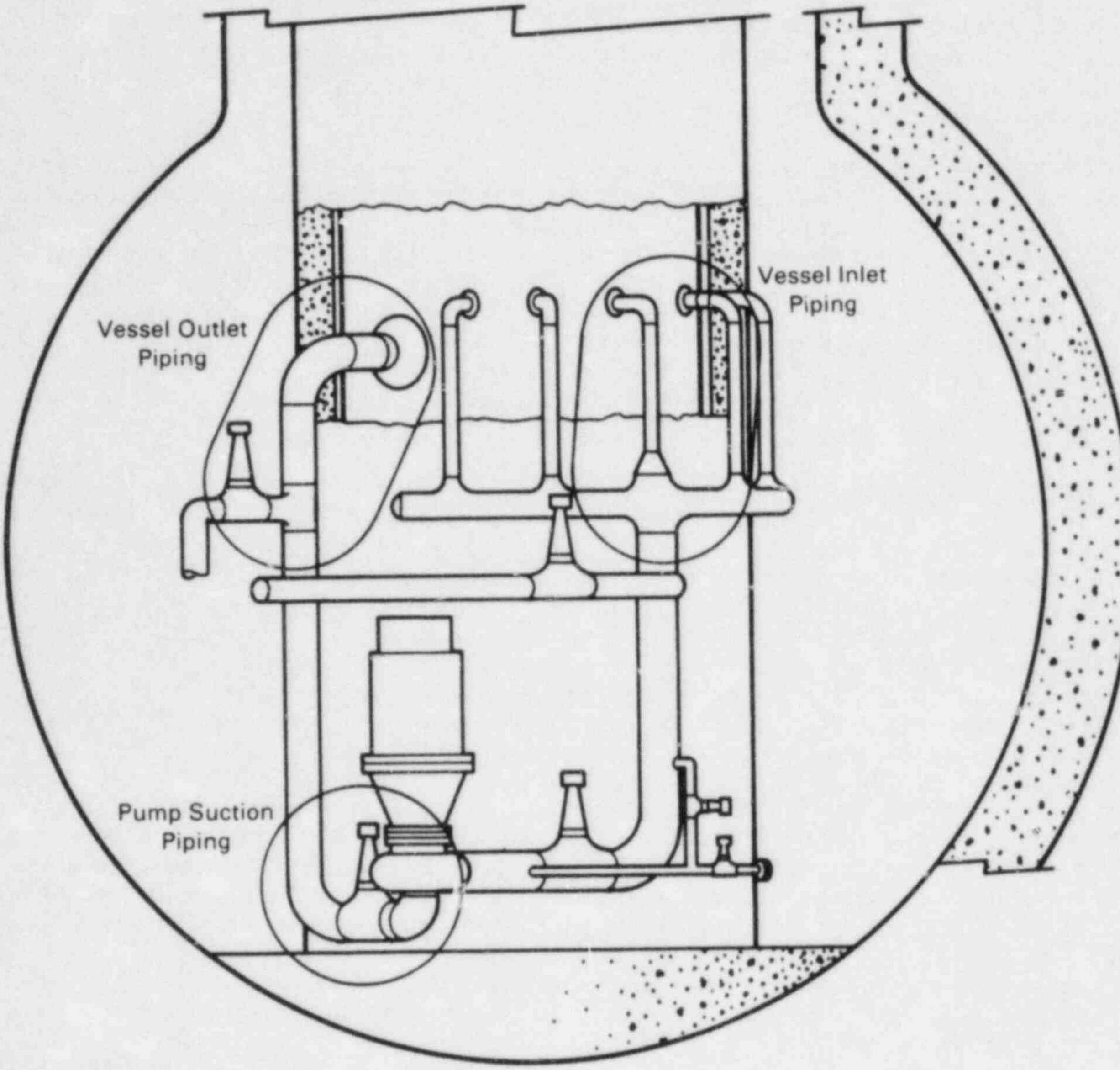

FIGURE 5.2. Typical 28-Inch Recirculation Loop with Mark I Type Containment

other replacements. NMP-1, however, has a slightly different system from the others because it has five recirculating coolant loops and no jet pumps. The other five plants have two recirculating loops and have jet pumps.

5.1 NINE MILE POINT UNIT 1

The NMP-1 Nuclear Power Plant is a Model-2 BWR owned and operated by Niagara Mohawk Power Corporation (NMPC) with headquarters in Syracuse, New York. It is a 610-MWe system supplied by General Electric Corporation; full power was achieved for the first time in January 1970.

The reactor design incorporates five reactor water recirculation system (RWRS) loops and provides forced circulation of primary water through the reactor core. This forced circulation allows for a higher specific power than natural circulation and also permits control of flow distribution to all channels within the core. There are no jet pumps.

As a result of the detection of IGSCC in the recirculation system safe ends, the entire Type-316 stainless steel recirculation piping system at the NMP-1 Nuclear Generating Complex at Scriba, New York, was replaced during the 1982-1983 outage. At the time, the repair effort by NMP-1 staff and their contractors, was the largest in-plant primary-piping replacement job undertaken at a U.S. nuclear facility. However, NMPC was prepared to handle this work and had planned for the possibility of repair efforts since 1978. This planning was based on the well known pipe cracking of smaller diameter pipes. In addition, their planning efforts were driven by the need to minimize the duration of any outage and thus cut costs.

The presence of IGSCC at NMP-1 was not discovered by the results of a routine test but was determined by a series of events that spanned a several-month period during 1982. The event that triggered NMPC action occurred on March 21 when the reactor operators noticed a high-humidity condition in the drywell. Upon investigation, a pump seal was suspected to be the cause and reactor unit number 1 was shut down to replace the seal. On March 23, during a 900-psig hydrostatic test of the reactor vessel and piping, the before-startup inspection revealed through-wall leaking cracks in two of the ten furnace-sensitized recirculation loop safe-ends. Further visual inspection indicated three pinholes and a 1/2-inch-long axial crack in the HAZ of the safe-end-to-piping welds. These safe-ends had been ultrasonically tested for a routine inspection during mid-1981, but no reportable leaks had been detected. Based on the visual results, NMPC in 1982 decided that a safe-end replacement program was necessary. Subsequent UT inspections revealed more crack-like indications in the 28-inch recirculation piping system HAZs, and on June 16, 1982, NMPC decided to replace the entire recirculation piping.

All of the original Type-316 stainless steel recirculation system safe-ends, elbows, and piping between the inlet/outlet nozzles and the recirculation pumps were removed between June and October 1982. This required a 300 member workforce made up of NMPC staff and its contractors. After the original piping was removed, work was begun to install 10 safe-ends, 20 elbows, and 500 feet of piping, all constructed from Type-316 NG stainless steel. The installation work was completed by April with all related work finished by the end of June 1983. NMPC had anticipated the potential for pipe replacement after being notified by the NRC in November 1978 of the potential IGSCC problem at BWRs. By July 1979, NMPC began purchasing and storaging long-lead-time items including safe-ends, elbows, piping, shielding, and welding materials. In addition, the design and manufacture of specialized tools, such as cutting and welding machines, was authorized and paid for by the utility. Fortunately, NMPC took advantage of this long lead time to plan for the repair and replacement of the primary water recirculation system at their NMP-1 plant.

IGSCC Assessment

In an attempt to ascertain the reason for extensive IGSCC, primary water chemistry parameters were reviewed. Those monitored weekly at NMP-1 included: 1) average and maximum conductivity, 2) average and maximum Cl^-, 3) pH, 4) silica, 5) dissolved oxygen, and 6) metallic impurity concentrations of Fe, Cu, Ni, and Cr. Chemistry conditions were found to be equivalent to those at other operating BWRs. However, there have been at least two significant water chemistry transients during the NMP-1 plant history that were not typical of the average BWR conditions within the United States. The first was a high conductivity level that existed for 5 hours and ranged from 8 to 30 times the operational limit. The second was a high chloride transient with a peak concentration of three times the operational Cl^- limit that existed for several hours. Studies indicate that these two transients and the normal steady-state primary water chemistry conditions did not play a significant role in accelerating IGSCC at NMP-1 (Delwiche, u.d.).

Besides evaluating the primary water chemistry, the weight, thermal, and vibratory stresses at NMP-1 were reviewed, were found to be low for both normal and abnormal operating modes, and were within the design limits. None of the contributing factors (chemistry, stress, sensitized material) were found to be excessive enough to expect the degree of IGSCC that occurred at NMP-1.

The pipe cracks at NMP-1 resulted from IGSCC of HAZs, which began on the inner surface. Shop and field welds were equally affected with some crack propagation occurring into the weld material. However, regions of higher ferrite content impeded or minimized crack propagation.

Two types of cracks existed, those that were axially oriented and those that were circumferentially oriented. Through-wall leakage was primarily due to axially oriented cracks. The leaking safe-ends at NMP-1 all had cracks that were axially oriented. Most cracks were determined to be less than 1 inch long; however, a significant number did range up to 6 inches long. Generally, there were indications of more than one axially and/or circumferentially aligned crack in each affected weld.

The detection of IGSCC was accomplished in the field using UT and dye penetrant testing. In addition, specialized analysis was performed on pipe samples by independent laboratories (Battelle-Columbus, General Electric, and Sylvester Associates) using optical metallographic and scanning electron fractographic techniques. It was concluded from these metallurgical evaluations that the material degradation resulted from IGSCC in the sensitized region of the weld's HAZs.

Replacement Pipe Quality

The piping, elbows, safe-ends, and fittings of the original five 28-inch recirculation loops at NMP-1 were constructed of Type-316 stainless steel. The safe-ends were fabricated from the same 0.054% carbon piping, then solution heat treated before welding, and finally postweld heat treated in conjunction with the reactor pressure vessel resulting in a furnace sensitized condition. This Type-316 stainless steel was found to have a significant number of cracks in the HAZs from IGSCC.

In response to this problem, NMPC decided to replace all of the 28-inch recirculation loops including safe-ends with Type-316 NG stainless steel. The Type-316 NG material is expected to be fully resistant to IGSCC problems throughout NMP-1's lifetime.

Pipe Repair/Replacement Program

The decision was made by NMPC at the end of March 1982 to replace the ten safe-ends in the primary water recirculation system of NMP-1. A pipe repair/replacement program (Delwiche u.d.) was submitted to the NRC that included the chemical decontamination of the system, shielding considerations, an ALARA program, and personnel dose estimates. The NRC approved this plan; however, NMPC submitted a revised plan on June 16, 1982. This plan was for the repair/replacement of the entire recirculation system safe-ends, elbows, and piping between the inlet/outlet nozzles and the recirculation pumps. The modified plan was approved by the NRC and work started with removing the recirculation piping at NMP-1.

Contractor Support

The overall organization for the program to repair and replace the primary water recirculation piping system was managed by NMPC personnel. However, NMPC did not have sufficient labor or management staff to perform the needed work. They solved this problem by using contractors instead of hiring and training more personnel.

Preliminary planning had been underway for several years before the actual repair/replacement work began because of NMPC's and NRC's concern that NMP-1 was a likely candidate for IGSCC. In 1979, NMPC contracted with Newport News Industrial Corporation (NNIC) to prepare a contingency program for replacing the reactor vessel's safe-ends. On March 26, 1982, NMPC awarded NNIC an initial contract to replace the ten safe-ends and their associated piping elbows. This was expanded in June to cover the replacement of the entire recirculation system. Radiological Chemical Technology Corporation was contracted to provide onsite consulting services for the duration of the pipe repair/replacement outage. Other contractors were also hired to support the removal and replacement of the recirculation system and to provide specialized analysis and consulting services.

Recirculation System Decontamination

The five NMP-1 recirculation system loops were decontaminated by a water-surging decontamination process before the bulk of the UT inspections. This was performed to reduce radiation levels allowing workers to remain longer in the drywell for UT inspections and pipe removal efforts, thus reducing personnel exposures. The NMP-1 recirculation pumps were decontaminated in April 1981, and the entire recirculation system was decontaminated from April to August 1982. A two-phase decontamination process was used during the second time frame. The upper elbows were decontaminated in phase 1 and no appreciable change was observed in the drywell radiation levels. In phase 2, the rest of the system was decontaminated and drywell radiation levels were reduced to 10-20 mR/hr in the vicinity of the piping. The overall decontamination factor for phases 1 and 2 was approximately 10.

The decontamination process was performed by London Nuclear, Ltd. using with the Can-decon® process. This method was effective for decontaminating the piping but was much less effective for decontaminating pumps and valves. NMPC believes that the pipe decontamination process appears to have made IGSCC more visible to UT examination.

Shielding

The primary external radiation sources in the drywell of NMP-1 were the radioactive deposits (fission and activation products) on the inside of pumps, valves, elbows, and piping. These sources may also become a source of internal exposure when systems are cut. The radioactive material can either be ingested or inhaled by personnel through airborne or surface contamination pathways. To minimize these external and internal sources during the repair/replacement of the recirculation system, NMP-1 performed a chemical decontamination of the system as discussed previously. The hot spots in the primary system were flushed to decontaminate them before any shielding was erected. After that decontamination process, the reactor pressure vessel (RPV) became the major source of exposure for personnel working on the recirculation system. The locations that presented the highest potential for exposure were the openings in the biological shield where pipes from the RPV penetrated the shield. These areas were of concern because this was where the majority of the cutting and welding on the recirculation system was to take place. In addition, one of the major crud traps for the RPV was located in the vessel annulus between the core shroud and the outlet nozzles as shown in Figure 5.3. The upper portions of the control rod assembly and the control rod blades were also a significant radiation source (primarily ^{60}Co).

Shielding was used at NMP-1 to reduce personnel exposure in the drywell from these radiation sources during the recirculation system repair/replacement. Shielding was found to significantly reduce doses. This shielding included: 1) speciality shielding within the biological shield such as inlet/outlet, recirculation system, nozzle plugs and control rod guide tubes filled with water, 2) exterior shielding between the biological shield wall and the recirculation nozzles, 3) shield curtains around equipment and piping, and 4) other specialized shielding as recommended by the health physics staff and the ALARA committee.

Nozzle shield plugs, nicknamed the silver bullets, were inserted in the recirculation outlet and inlet nozzles after the adjacent elbows had been cut out. Shield plugs were necessary because the natural shielding provided by reactor water was removed when the water was drained below the lower core support plate to the level shown in Figure 5.3. The plugs reduced the beam of radiation emanating from the lower core support plate, the activated core structure, and the crud traps. Work crews were trained on mockups, and specialized tools were used to install and remove the plugs. In addition, the positioning of the control rods in conjunction with the filling of control rod guide tubes with water was found to be an easy and effective dose reduction technique at NMP-1.

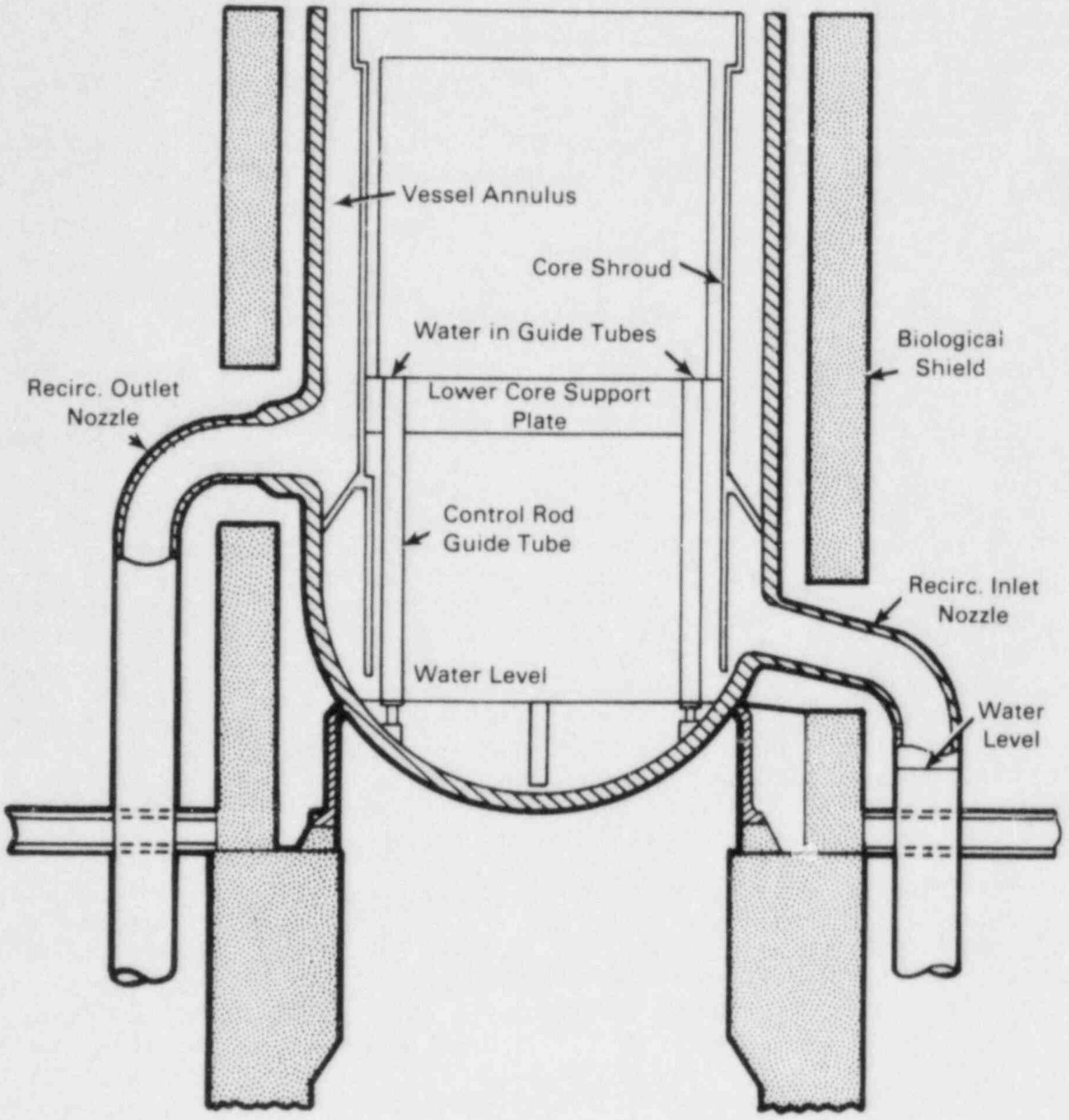

FIGURE 5.3. NMP-1 Reactor Pressure Vessel Configuration During the Recirculation System Repair/Replacement

Temporary shielding was used when it would result in an overall man-rem savings, that is when the dose saved in performing the particular recirculation system task outweighed the estimated exposure received from installation, maintenance, and removal of the shielding. Some areas with low radiation levels were shielded when the areas were to be occupied by large numbers of personnel for long periods of time. These locations were designated as "stand-by" areas and were used for personnel on call for a particular task. The personnel, dressed in appropriate protective clothing, remained in these areas to reduce delays.

Remote Equipment/Monitoring

As early as 1979, NMPC started designing and purchasing automated equipment for the repair/replacement of the recirculation system at NMP-1. Automated equipment used in the drywell for this work included: 1) pipe cutting machines, 2) welding equipment, 3) recirculation outlet and inlet nozzle plug installation tools, and 4) weld crown reduction tools. Cutting machines were designed to cut, weld prep, and counterbore from the same basic setup. Both mechanical and plasma arc cutters were used. This eliminated the setup and disassembly times for separate machines plus produced an accurate weld prep and counterbore because the same setup was used for all operations. The cutting machine fed automatically, which allowed the operator to move to a lower radiation area while the machine was running. Automatic welding equipment was designed to obtain the same objective of minimizing operator exposure while producing a high-quality weld that needed little or no repair effort. In addition, remote fiber-optic systems focused on the welding location allowed remote welders to watch weld progress and control commands sent to the welder unit. Automatic inlet/outlet pump and nozzle installation tools allowed workers to move away from any potential radiation beams coming from the core through openings in the reactor pressure vessel/biological shield.

Audio communication equipment and remotely-controlled television monitoring were used at the various job locations so that supervisors could direct work activities from areas with low radiation levels. In addition to the work locations, these systems were installed at the drywell entrance and other locations as needed.

5.2 MONTICELLO

The Monticello Nuclear Generating Plant (MNGP) is a 536-MWe plant that began commercial operation in July, 1971. During a routine in-service inspection on September 28, 1982, a shallow crack was found in the end cap of one of the recirculation system riser manifolds. Further investigation showed three small axial cracks in the 12-inch risers to the nozzle safe-end welds and one crack in a riser pipe-to-elbow weld. An additional small through-wall axial crack was located through hydrostatic testing. The cracks were determined to be caused by IGSCC. While preparing the recirculation safe-ends for repair, through-wall leakage developed in each of the safe-ends. Weld overlays were made where cracking was found. Subsequently, the recirculation piping and associated safe-ends were replaced during an outage that lasted from February 4, 1984, to January 17, 1985. A section of the RHR system was also replaced at this time because of its contribution to the background radiation in the area of the discharge risers and safe-ends and because it interferred with moving other piping.

The pipe replacement project was managed by Nuclear Engineering and Construction (NE&C), a Northern States Power Company (NSP) organization based in the NSP home office. NE&C coordinated operations with the assistance of contract personnel at MNGP. The contractors and suppliers reported to NE&C. The major contractors included, Bechtel Power Corporation, responsible for engineering, procurement and related services; Impell, for third party review; Quadrex, for pipe decontamination; General Electric, for construction and welding; Hitachi, for induction heat stress improvement (IHSI); Power Cutting, for machining and cutting; and the Electric Power Research Institute, for evaluating the repair methods.

The pipe replacement task force consisted of personnel from MNGP, NE&C, the NSP corporate office, and the major contractors. Decisions were made by the task force, NE&C and MNGP instead of by an individual or organization responsible for the replacement. This worked acceptably, although an audit from an independent organization indicated that a project manager would have been appropriate for a project of this size.

Preparation

Several activities were performed early in the outage to reduce the exposure rates in the drywell during pipe replacement. The fuel bundles and 32 peripheral control rod blades were removed. The remaining control rod blades were inserted 4 feet to further reduce radiation exposure. Jet pump plugs were installed on the jet pump assemblies so the water in the shroud could be maintained as high as possible. Two small pumps were placed in the annulus to remove any water seepage during safe-end replacement. A standpipe was installed to control the water level inside the shroud during safe-end replacement activities.

Originally there were no plans to rebuild the recirculation pumps. These plans were changed when an inner gasket on one pump was discovered to be leaking. During the repair of the gasket, further problems were discovered (the thermal barrier in the pump was loose) so the pumps were rebuilt.

Pipe Decontamination

Before decontamination, debris was vacuumed from crud traps such as those between the jet pump diffuser and the shroud and between the core barrel and the core support plate. Unfortunately not all the debris between the jet pump diffuser and the shroud could be removed, and readings near 5000 R/hr on contact were obtained using an underwater probe.

The gap between the thermal sleeves and the safe-ends was hydrolased from inside the reactor to remove contamination that had built up since the previous outage. A very small decontamination factor was achieved, although during the previous outage the dose on the outside of the discharge nozzles was reduced by a factor of ten.

Both loops of the recirculation system were decontaminated, including the reactor vessel suction nozzle and the discharge nozzles. Specially designed suction plugs were installed to prevent decontamination fluid from entering the annulus. The decontamination included the recirculation pumps and the bypass lines. The RHR system was decontaminated from the return line to the in-board, air-operated check valve and from the supply side to the motor-operated gate valves. The Quadrex Company performed the decontamination using the LOMI dilute chemical process.

The decontamination worked quite well on large-bore piping, but was not very effective on small-bore piping. Average contact decontamination factors of 22 and a general decontamination factor of 21 were achieved. The general area dose rates in the drywell were reduced from 30 mR/hr to 3-5 mR/hr. The recirculation pumps were decontaminated but the pump impellers still read 50 R/hr on contact after the decontamination process.

Unfortunately, the decontamination process took much longer than the utility had originally expected because of mechanical difficulties on the part of the contractor. The decontamination of the recirculation system began on March 6, 1984, and took 21 days of critical path time. It originally was scheduled for a period of 6 days of critical path time. The delay was due to leaks in the hose fittings, which created a potential for the spread of contamination. To eliminate the problem, welded pipe was installed to replace the hoses. This resulted in a loss of engineering time as well as a delay in the outage. The man-rem savings obtained by decontamination were estimated to be 827 man-rem. The dose attributed to the decontamination process was 34 man-rem.

Shielding

Various types of shielding were used during the pipe replacement. A 33-inch-thick by 179-inch-diameter concrete shield plug was placed inside the reactor pressure vessel to provide shielding to the refueling floor. Lead blankets were placed over the metal plates of the shield plug assembly on top of the annulus. Lead shields were lowered into the annulus and placed behind the suction nozzles to provide shielding to drywell workers equivalent to the amount of shielding provided by the vessel wall.

As soon as the risers were cut away from the safe-ends, soft lead bricks (2 x 3 x 6 in. oblong bags filled with No. 8 and No. 12 lead shot, weighing approximately 10 lbs each) were placed inside the nozzles. This reduced the radiation until the safe-ends were removed. The biological shield blocks surrounding the suction and discharge nozzles were removed, and soft lead bricks were installed in the biological shield windows. Aluminum frames were fabricated to surround the suction and discharge nozzles and hold the soft lead bricks in place. Later in the outage, additional soft lead bricks were added to compensate for the settling and flattening of the first bricks. Lead blankets were also used outside of the biological shield openings during work on the nozzles.

Water was used to shield the vessel internals whenever possible. Originally the level of water in the vessel was maintained even with the top of the upper grid plate for shielding. The work on the safe-ends, however, required that the annulus be drained and shortly thereafter, the jet pump plugs began to leak. One of the two small pumps that had been placed in the annulus to remove the water leakage failed, and the second pump was not able to remove the water fast enough to keep the annulus from filling to the point where water ran out the nozzles and into the drywell. The water level in the shroud was then lowered to just below the jet pump diffusers to keep the water from flowing into the annulus. Lowering the water level to this point caused no significant increase in the radiation levels at the nozzle.

During the outage, the decision was made to replace the jet pump instrumentation (JPI) safe-ends and one standby liquid control (SBLC) safe-end. The SBLC safe-end work required that the water be drained to the level of the control-rod-drive stub tubes. The water was lowered to this level in case the work could begin while the work on the discharge nozzles was in progress; however, the radiation levels at the discharge nozzles were too high, so the water level was raised back to the level of the jet pump diffusers. After the work on the discharge nozzles was completed, the vessel was redrained and the work on the JPI and SBLC safe-ends started. The radiation levels were still significant near the discharge nozzles, but personnel were no longer working in this area.

The recirculation pump impellers were removed under water. A steel table was used to shield the workers during work on the recirculation pumps. Soft lead bricks and blankets were used for additional shielding.

Pipe Cutting and Removal

The first pipe cut occurred on March 27, 1984, and all piping was removed within nine days. All of the pipe was machine rough cut. A space study was conducted by Bechtel (using video cameras and pictures) to determine the maximum size of pipe that could be removed from the drywell. The cutting was performed to produce pipe of the largest size possible to be conveniently moved out of the drywell so that additional cuts were not required. In general, the cuts were made at the location of the old welds.

The cut piping and the recirculation pump motors were removed via the drywell equipment hatch. The pipes and pump motors were rigged to a 6-by-6-ft air pallet and then transferred to the reactor building railroad doors. Because the air pallet floated on a cushion of air, ramps had to be placed through the drywell equipment hatch, and the floor drains in the reactor building had to be taped over. Quadrex removed the old pipe for disposal. The cutting and removal of the piping occurred much faster than had originally been estimated.

Pipe Quality

The new replacement pipe was Type-316 NG stainless steel with less than 0.02% carbon. Nitrogen was added to the pipes to meet the strength requirements of Type 316. The 12-in. risers were prebent seamless pipe manufactured by Japan Steel Works. The 22-in. to 24-in. seam welded pipe was procured from Texas Pipe Bending.

Prebent pipes and reducers were used to eliminate welds and end caps. The new pipe was solution annealed, electropolished, and cleaned. Some of the new bent pipe had induction bending problems, and crazed cracks were found, primarily in association with the ring header. These pipe manufacture problems resulted in about 75% of the new pipe being rejected.

Automated equipment was used to improve weld consistency and to minimize welder radiation exposure. Efforts were made to minimize the number of field welds to reduce the potential for IGSCC and to redesign the system so that it was 100% UT inspectable.

Pipe Alignment and Rewelding

The in-place measurements of the piping were performed by General Electric before the pipe was cut. After cutting, the large diameter pipes were easily aligned during the replacement process. The critical problem was installing the discharge safe-ends and thermal sleeves. Before they could be prepared for welding, the thermal sleeves had to be aligned radially and axially to the as-found position. Unfortunately, the original measurements could not be reproduced because of procedural errors by the contractor. Additional difficulty in achieving alignment was attributed to the jet pump plugs that were installed in place of the actual jet pumps. Since proper alignment could not be attained with the jet pump plugs in place, caps were fabricated and placed over the nozzles allowing the vessel to be flooded. The jet pump plugs were removed and a trial fit of the functional jet pumps was made; then the vessel was flooded in an attempt to verify that the acceptance criteria had been met for the fit of the thermal sleeves and the safe-ends. The vessel was flooded twice before the test was successfully completed. The vessel was then drained to below the diffusers to facilitate thermal sleeve and safe-end installation.

Replacement of the two suction safe-ends proceeded with very few problems. There were no thermal sleeves to align, and dose rates were lower than anticipated.

Remote arc machines were used for the welds. Four hours were needed to set up each machine and up to eight machines were set up, with five to six operating at a time. The operator manually buffed the weld after each weld pass.

The carbon-to-stainless-steel welds were performed by adopting a Japanese technique for dissimilar metal welds. A 309 weld was done on the carbon steel. This was welded over with 308, and then a weld prep of 308 material was made. This resulted in a superior chemical composition for the final weld. The last pipe weld occurred on September 16, 1984. The last x-ray and complete acceptance of the recirculation piping was on September 30, 1984.

Last-pass heat-sink welding was originally planned, but because of operational limitations, IHSI was chosen instead.

5.3 COOPER

The Cooper Nuclear Station, owned and operated by the Nebraska Public Power District (NPPD), is a GE 778-MWe, Mark-1, BWR-4 that has been in commercial operation since 1974. During a pipe inspection the fall of 1983, their Type 304 recirculation and RHR piping was inspected for cracked welds. Twenty crack indications were found in the recirculating pipe, none in the RHR. Further inspection revealed a total of 47 circumferential cracks, 42 of which were repaired. In September of 1984, the plant shut down to begin an 11-month outage during which recirculating, RWCU, RHR, and core spray pipe systems were replaced.

Cooper contracted with Nissho Iwai, a Japanese firm, for pipe engineering, Chicago Bridge and Iron for pipe installation, and Pacific Nuclear Services for pipe decontamination. GAPCO provided welding services and Radiological Chemical Technology Inc. provided assistance with writing the radiological protection plan and with dose tracking. The pipe replacement was run by the Construction Management Group, formed within the NPPD, to provide guidance and review.

Pipe Decontamination

Pacific Nuclear Services performed the pipe decontamination at Cooper using the Citrox process. It took 15 days to prepare for decontamination and 4 days to actually decontaminate the pipes. The utility chose not to decontaminate the recirculation pumps because of difficulties other utilities had experienced. To prevent the decontamination solution from entering the pumps, 12-inch pieces of pipe were cut between the pump and valves. A plate with a 4-inch flange was welded onto the end of the pipes being decontaminated.

A number of steps were taken to ensure thorough cleaning of the pipes. Before the decontamination, Cooper personnel vacuumed around crud trap areas and found that it significantly minimized crud and reduced area dose rates. They also performed some hydrolasing of the lines and decontaminated the reactor water cleanup line.

During the decontamination, the chemical level rose to about an inch below the riser elbows. To decontaminate the rest of the risers, a hole was cut out in the top side of the elbow and a whirlybird sprayer was put into place to decontaminate the nozzles. Cooper ordered special suction plugs with 4-inch Pb to shield the nozzles. One of these was apparently not snubbed off properly, and there was some leakage into the annulus. After the decontamination chemical had been cleaned up, the nozzles were overflowed to flush out the annulus.

Decontamination was more effective than originally estimated and a decontamination factor of approximately 40 was actually achieved rather than the estimated decontamination factor of 25. The utility believes that the decontamination operation saved a minimum of 350 man-rem.

In addition to the man-rem savings, the utility realized other benefits from pipe decontamination. There was less possibility of creating airborne contamination during the cutting and moving of the used piping. Disposal of the pipe was also easier because of the lower activity.

Pipe Quality

The Cooper engineers experimented extensively with pretreated pipe samples before selecting new pipe. They chose Type-316 L NG pipe purchased from Japan. The pipe is a seamless, continuous run with no welds at the elbows. This minimized the number of measured welds thus reducing the potential for future IGSCC.

During the fall of 1983, pipe test specimens were placed in the reactor water cleanup system and the recontamination rates were compared. The pipe specimens had the following pretreatments:

- as-received material preoxidized in deoxygenated water for 1000 hr at 550°F
- electropolished and preoxidized in air-saturated steam for 150 hr at 550°F
- electropolished only
- as received
- GE zinc application
- GE control specimen
- electropolished and preoxidized in moist air for 150 hr at ambient pressure and high temperature.

The electropolished and preoxidized test sample had the lowest recontamination rate and overall best performance, so that method of pretreatment was chosen for the replacement pipe. The pipe was pretreated with Radiological Chemical Technology Inc.'s Process No. 2 to prevent contamination buildup (Inside NRC, September 2, 1985).

Shielding

Minimizing exposure time during the various operations was the focus for keeping exposures low where practicable before decontamination. Lead blankets were applied around elbows, at penetrations, and at valves. Lead blankets were also used to shield one of the suction nozzles that leaked and was not thoroughly decontaminated. Shielding frames loaded with lead bricks were set in the N1 and N2 nozzle openings. After decontamination was completed, there was little need for lead blanket shielding in general areas, but it was used to reduce doses from the recirculating pumps, drain lines, and several residual heat removal lines. During cutting activities, shielding was also used on the headers where the decontamination was not very effective. Temporary shielding was generally found to be more effective than had been estimated in the piping replacement radiation protection plan.

Pipe Cutting and Welding

Plans were made from the beginning to minimize pipe alignment problems and proved reasonably successful. Remotely operated cutting and welding equipment was used at Cooper during the pipe replacement project. The design of the equipment allowed the operator to remain in a lower dose rate area during operation. The required setup time and the need for someone to observe the equipment to ensure smooth operation somewhat offset the obvious benefits of remote operation. The remote cutting and welding equipment at Cooper included a self-centering internally mounted mandrel, a thermal sleeve cutting machine, an internal diameter automatic tack welding machine, an internal diameter automatic welding machine, an automatic cutting machine on outside diameter cuts, and an automatic welding machine for outside diameter mounts.

All of the pipe was machine cut except for the holes cut into the elbows for the decontamination operation. In this case, a hole saw was used initially, but the metal was so hard that an arc gouge had to be used to complete the holes. The automatic cutters were difficult to use around valves, but the cutting problems were solved and there was no significant airborne release. The utility believed that the use of automatic cutters had little dose savings value because, although it did allow the operator to stand back from the pipe, it did not eliminate the need for an operator to observe all cuts and prevent entanglement of cables. Automatic cutting also required equipment setup and removal, forcing the operator to work close to the pipe.

Prepping and machining of new pipe took place in clean areas whenever possible to avoid unnecessary exposure and cramped working conditions. The old pipe was cut into segments short enough to remove from containment and was plugged with plastic pipe covers to prevent beta exposure and spread of contamination. A staging area was set up at a central location away from drywell traffic to prepare the pipe for disposal. This increased productivity in the drywell and reduced the overall exposure from pipe preparation to radiation sources other than the pipe itself.

A multipurpose facility was built that housed the tool decontamination operations during the outage. Most of the tool and equipment cleaning was performed with a water bead blast.

5.4 PEACH BOTTOM UNIT 2

Peach Bottom Unit 2 is a 1065-MWe Mark I BWR run by Philadelphia Electric Company (PECo). The unit began commercial operation in 1974. Indications of

IGSCC were detected during midcycle weld examinations in July 1983. Cracks primarily circumferential, were detected at 26 weld locations, and further fracture mechanics analyses confirmed the need for weld overlays at 21 of the locations. The overlays provided short-term repair until pipe replacement could be initiated during the next fuel outage. The initial plans were to replace recirculation, RHR shutdown cooling, RHR head spray and portions of the RWCU piping. Inspection of the recirculating safe-ends and the jet pump instrument safe-ends and seals was also scheduled. The pipe replacement program began April 27, 1984. An early finding that the safe-ends and seals needed replacing increased the project scope, potentially exposing personnel to more dose than had been originally anticipated.

PECo hired General Electric Company to design and procure recirculation system piping and provide technical guidance during the pipe installation. Bechtel Power Corporation designed and procured replacement RHR and RWCU piping. Chicago Bridge and Iron (CB&I) was responsible for removing old equipment and installing the new piping and associated system. They also coordinated pipe replacement activities and were responsible for developing and managing the project ALARA program. The pipe replacement was completed on June 30, 1985.

Pipe Decontamination

Since the primary radiation source in the drywell is radioactive deposits on the inside walls of pumps, valves and piping, chemical decontamination of the recirculation, RHR, and RWCU piping, valves, and pumps was performed at Peach Bottom Unit 2. London Nuclear performed the decontamination with the Can-decon® process. Before decontamination began, the reactor pressure vessel was isolated by cutting and capping the two suction nozzles, ten discharge nozzles, three RHR pipes, and one RWCU pipe.

The Can-decon® process was effective in decontaminating the piping, but was not sufficiently effective in decontaminating pumps and valves. Consequently, there was little or no dose rate reduction from decontamination in the area of the pumps and valves, and contact dose rates on the pumps remained at 15-20 R/hr. Because workplace dose rates (the quotient of collective exposure divided by exposure hours in the drywell) at the lower level was dominated by radioactive sources in the pumps and valves, only a fraction (44%) of the estimated man-rem savings was achieved. The initial average dose rate in the drywell before and after decontamination was 136 mR/hr versus 43 mR/hr.

Personnel exposure was required to isolate the system before decontamination, to connect, operate, disconnect, and decontaminate the equipment, and to dispose of the radioactive waste. Because of the persistent high dose rates adjacent to them, the pumps and valves were hydrolased, disassembled, and mechanically decontaminated, and the adjacent components were shielded. The net exposure savings from chemical decontamination was estimated to be 1400 man-rem rather than the 3200 man-rem originally expected. Isolating the system required 12 critical path days, and chemical decontamination required 6.5 critical path days. All other associated work was performed off the critical path.

Additional decontamination efforts occurred in the annular space between the thermal sleeve and the discharge nozzels. This is a major crud trap in the reactor pressure vessel and a major source of radiation to the workplace around the safe-end. Personnel from the service platform hydrolanced this space to dislodge the trapped radioactive material and move it away from the workplace. The estimated dose savings from hydrolancing the safe-end crevice was 100 man-rem. Hydrolance of the nozzles required two critical path days; however, it could have been performed off the critical path.

Operating the RWCU system as long as possible helped maintain water clarity and made hydrolancing operations on the service platform easier and faster. This operation had no impact on the schedule and had minimal cost. Although no measure of dose saving was determined, shorter stay times on the service platform led to lower doses.

Shielding

When all fuel is removed from the reactor before the pipe replacement activities, the major sources of radiation exposure in the drywell are the activated components of the reactor vessel and radioactive crud deposited or trapped in systems containing primary coolant. The pipes were chemically decontaminated and removed from the drywell, and the pumps were disassembled and decontaminated. Subsequently, the high dose rates in most work areas were from reactor vessel sources; the rates were high mainly near openings in the biological shield where pipes penetrate the shield, and these sources were shielded before the cutting and welding operations took place.

The upper portions of the control rod assembly and the control rod blades were a significant source of radioactive ^{60}Co. Full insertion of the control rods moved these radioactive sources as far away as possible from the nozzle work areas and resulted in an estimated dose savings of 200 man-rem. There was no critical path impact or cost from this measure. Proper control rod positioning is an easy and effective dose reduction technique.

Peach Bottom had planned to use water in the shroud and vessel to shield radiation originating from the reactor vessel internals. The water level was to be maintained by using jet pump slip joint plugs, nozzle plugs and two annulus pumps. However, leakage past the plugs was greater than anticipated. Use of both pumps to maintain a suitable water level left no backup pump, and the idea was abandoned.

To shield the suction nozzle work area, lead slabs were installed inside the reactor vessel in the annulus to cover the recirculation suction nozzles. Installation of the slab shield required two critical path days and resulted in an estimated dose savings of 200 man-rem.

Additional shielding in and around the recirculation pipe nozzles was used in the drywell at the biological shield penetrations. This shielding replaced shielding normally provided by the biological shield doors before their removal and by water inside the recirculation piping before the pipes were emptied. To replace the biological shield doors, soft lead bricks were stacked inside steel forms around the nozzles. Lead shielding inside the nozzles and the nozzle shield plugs replaced shielding normally provided by water. Shielding in and around the nozzle work area required few critical path days and cost approximately $50,000. Significant exposure rate reduction was achieved in the work areas around the nozzles, and over 150 man-rem was saved.

Miscellaneous hot spot shielding was used throughout the drywell during pipe replacement activities. Many of the hot spots shielded were associated with pumps, valves, and drains. Significant dose savings resulted and the work was done off the critical path; however, better management could have resulted in additional savings in dollars and man-rem. The estimated labor costs for hot spot shielding was $250,000 for a dose savings of approximately 250 man-rem.

Pipe Replacement

General Electric selected and supplied the pipe and fittings used in the Peach Bottom Unit-2 pipe replacement operations. The replacement pipe and fittings were Type-316 NG stainless steel with ≤0.02% carbon content and nitrogen controlled between 0.06% and 0.10%. The 12-in. riser pipe was seamless. The ring headers and 28-in. pipe were rolled and welded. ALARA considerations were an integral part in the replacement pipe design. The inner surface of the new pipe was mechanically polished and electropolished to minimize buildup of radioactive deposits. The pipe was prebent resulting in nearly 50% fewer welds consequently reducing the number of weld needing in-service inspection (ISI). Numerous crud traps were eliminated, which will reduce future dose rates. The required welds were planned to allow access for remote automatic welding and ISI equipment.

As stated earlier, inspection of recirculatory pipe safe-ends and jet pump safe-ends and seals alerted the utility of the need for replacing these components. This major scope change significantly increased replacement activities.

Pipe cutting began on July 5, 1984. Most of the pipe cuts were performed remotely by machine. Two pipe cuts were made manually with plasma arc torches, which cut quickly and can be used in areas inaccessible to machine track set up. However, these cuts leave rough edges that must be milled off before applying weld prep. Four "windows" were also cut into the old recirculation pipe using the plasma arc technique. These openings facilitated hydrolasing of the suction and discharge pumps.

PECo used the experience gained by the NMP-1 pipe replacement to avoid problems in realigning pipe. The new pipe was locked in place and adjusted piece by piece to provide the alignment needed for welding the new pipe in place. Pipe welding operations were completed by March 24, 1985.

The machinery and welding equipment was designed to minimize setup time and the fixtures were mounted while water levels in the vessel provided some shielding. TV monitors viewed from outside the drywell reduced the need for supervision and for observers entering the drywell.

5.5 VERMONT YANKEE

The Vermont Yankee Nuclear Power Corporation began operating the 514-MWe Vermont Yankee BWR 4 Mark I containment plant in December 1972. By 1983, as a result of NRC's inspection order, 33 indications of cracked welds in the primary recirculating system and one indication in the RHR system had been discovered (Inside NRC - November 14, 1983). At that time, 22 mini-overlap repairs were performed, and moisture tape was applied to seven uninspected welds. As a more permanent solution, the decision was made to replace the recirculating and RHR piping in an outage beginning in September 21, 1985, and anticipated to last about seven months.

Vermont Yankee purchased controlled-chemistry NG Type-316 replacement pipe from Sumitomo Company in Japan to replace about 400 feet of high-carbon pipe with low-carbon stainless steel pipe. This seamless pipe with greater resistance to IGSCC was electropolished but was not preoxidized. The pipe was selected following evaluation of an EPRI panel of deposition coupons in the Vermont Yankee system. It was determined that the electropolished, preoxidized sample had a lower overall recontamination rate and better IGSCC resistance. However, it was also believed that any benefit from the preoxidizing treatment would be lost within the following operating cycle and therefore would not be cost-effective.

The piping installation contractor selected for the replacement was Morrison-Knudsen, with General Electric contracted for engineering services. Pacific Nuclear Systems, Inc., provided decontamination and radwaste removal services. Vermont Yankee provided overall direction to the involved contractors and organized the project with a dedicated project team responsible for engineering, construction, design, procurement, disposal, and quality assurance activities. Each contractor provided its own quality assurance plan to be reviewed and approved by the project team and periodically audited by Yankee Nuclear Services Division.

The new Type-316 pipe and fittings replaced the following drywell components:

- jet pump inlet nozzle safe-ends
- recirculation suction nozzle safe-ends
- all high-carbon stainless-steel recirculation system piping and components
- recirculation bypass line
- carbon stainless steel portions of the RHR system
- reactor vessel bottom head drain.

Vermont Yankee upgraded their recirculating system to reflect the advances in piping systems designed to reduce IGSCC. Some of the design improvements included:

- use of seamless material to eliminate longitudinal welds
- use of integrally forged fittings and bent pipe to eliminate welds
- deletion of ring headers and caps to eliminate crud traps
- deletion of loop crosstie valves to eliminate crud traps
- improved internal diameter counterbore to facilitate ISI
- cylindrical prolongation on all fittings to facilitate ISI
- electropolishing of inner pipe surfaces to minimize future radiation buildup.

Additionally, all weld joints are being designed to withstand an "acceptable minimum" of weld-induced residual stress. Solution annealing of butt-shop welds will be used to eliminate welding residual stresses. Replacement of their recirculation and RHR piping is being performed under the provisions of the Yankee Operational Quality Assurance Program as an engineering design change request.

Decontamination

Chemical decontamination of the recirculation and RHR systems was performed to reduce doses for personnel working in the drywell and reduce the potential for airborne releases during pipe cutting and realignment activities. An added benefit is that the radioactivity levels were reduced in the decontaminated pipe and allowing for simpler transport and cheaper disposal. The decontamination process selected was the Citrox solvent process performed by Pacific Nuclear Systems, Inc. The recirculation system from the N1 nozzle through the pumps and back up through the ring header to nozzle N2 was decontaminated along with the RHR (up to the isolation valve) and the reactor water cleanup system. The process was scheduled for about a week of critical path time.

The overall decontamination factor (DF) for the recirculation piping was 3.1 (after hydrolasing) with a general area dose reduction factor of 2.5. The DF for the RHR was 1.8 and for the RWCU piping was 7.4 (weighted harmonic means). Chemical decontamination of the recirculating systems, which was not very effective, was followed by hydrolasing. Water surges of the pumps were used to further reduce pump contamination but were not very successful. Laboratory tests confirmed that the surface film in the recirculating system was different from that in the RWCU system and was not as soluble in the chemical decontamination solution.

Hot spots remaining after completion of the decontamination included elbows in the suction and discharge 28-in. pipe, the N2 safe-end, and the upper surfaces of the ring header.

Shielding

Each potential use of shielding was evaluated to determine whether the benefits expected justified the dose incurred. Morrison-Knudsen, in conjunction with Vermont Yankee, used the following criteria to determine the usefulness and position of shielding:

- ensure that a net savings would result when balancing the dose received installing the shielding with the anticipated dose savings
- minimize the need to move shielding once it has been installed
- install shielding where it can be properly supported and does not affect the integrity of piping or other components.

Shielding was placed in the drywell around localized hot spots and at the open ends of pumps and major valves. Shield frames were constructed around the N1 and N2 nozzles, and lead plugs were placed in the nozzles. Shield plugs were put in valves and pumps to reduce doses during cutting operations.

Additional strategies to reduce area dose rates included maintaining the water level in the core to just below the jet pump slip joint, positioning the control rods to minimize exposure in the N2 nozzle region, ventilating the reactor vessel to reduce the potential for airborne radioactivity, and replacing at least one layer of cavity shielding when draining the reactor vessel.

Piping Removal and Installation

Vermont Yankee had planned to perform all major pipe cuts using remotely operated cutting machines. However, after completing the pipe decontamination, Vermont Yankee staff reevaluated the setup time and dose required for machine cuts. They decided that performing certain pipe cuts using the plasma arc torches was likely to save dose. Most destructive cuts were performed using the plasma arc. Others, including some valves, were machine cut. The pipe cutting task began on November 15, 1985.

Welding operations are underway using machines that are remotely controlled primarily from outside the drywell and viewed with optics mounted on the viewing heads. The welding activities are expected to be finished in late February 1986.

Cutting, machining, and welding the thermal sleeves to the jet pump risers require a different technique. The cutting, machining, and welding equipment mounts and operates on a stationary, self-centering, internally mounted mandrel that minimizes movement of equipment. This increases the accuracy of the operations and eliminates the need for alignment repositioning.

5.6 BROWNS FERRY UNIT 1

The Tennessee Valley Authority (TVA) planned to replace reactor coolant system piping at Browns Ferry Unit 1 that showed evidence of IGSCC. The replacement outage, scheduled for March 1985, was postponed indefinitely as Browns Ferry personnel turned their attention to complying with environmental qualification of their three units. The postponement came shortly before the pipe replacement was to begin. Considerable planning had been completed and construction contracts had been awarded before the postponement decision was reached. TVA personnel documented much of their planning, which will avoid future duplication of effort. Morrison-Knudsen Company, Inc., was to be the general contractor, and the services of London Nuclear had been secured for pipe decontamination.

In anticipation of the pipe replacement, personnel from TVA visited several other plants involved in pipe repairs to benefit from their experience and learn the types of problems encountered and how they were handled. This information from other's experience as well as plant-specific factors at Browns Ferry provided the basis for their plans. The following section discusses their planned pipe replacement.

Browns Ferry Operations

Browns Ferry Nuclear Station, consisting of three units, is owned and operated by TVA. All three are GE Mark-1 BWRs. Of the three, Unit 1 a 1067-MWe 2-loop plant that has been operating for 10 years, had the most indications of IGSCC of the three, and extensive weld overlays were applied to the pipe as a short-term solution to the IGSCC problem. Unit 2 showed an indication of a possible crack and did not require any overlays. The only evidence of cracking in Unit 3 was in the jet pump instrumentation nozzles which were repaired with overlays. IHSI was performed on Units 2 and 3 as a preventative measure against IGSCC cracking.

Unit-1 Pipe Replacement Plans

During extensive pipe inspections to determine the extent of cracks in the Unit 1 pipe welds, a total of 80 indications were found on 47 welds in the recirculatory and RHR systems. The majority of the cracks were circumferential, although some axial cracks were also discovered.

The Unit-1 pipe replacement scheduled during the fuel cycle 6 outage was to include replacing the recirculation, RHR, core spray, and RWCU pipe with 316 NG material. TVA selected Morrison-Knudsen Company, Inc. (M-K), as their general contractor for the pipe replacement. M-K and its subcontractors were responsible for removing all mechanical interferences, removing present pipe, installing new pipe, and replacing the mechanical structures previously removed. The contractors were also in charge of quality assurance. TVA was to provide health physics support. London Nuclear was contracted to decontaminate the pipes using their Can-decon® process. The cracked pipe was to be sent to Battelle, Pacific Northwest Laboratories, for examination.

When the pipe replacement was postponed, the plans were documented to guide a future replacement. To help prevent further pipe cracking in the meantime, the use of hydrogen water chemistry in the primary coolant system was being considered for the next operating cycle.

Replacement Pipe

The replacement pipe, purchased from a U.S. distributor, is NG stainless steel manufactured in Japan. This pipe is believed to have greater resistance to IGSCC than Type 304 pipe currently in use. The pipe was prebent to minimize the number of welds necessary for installation and to eliminate possible future pipe cracking sites. The pipe was sufficiently smooth that electropolishing was unnecessary, but it was preoxidized to provide a protective film on the inner surfaces.

Cutting and Welding Equipment

Plans called for the use of remote cutting and welding equipment wherever possible. Plasma arc torches would be avoided if possible to reduce airborne contamination during pipe cutting. All of the remote devices require setup time and operators present in the drywell during use. Therefore, even though it is operated remotely, the use of this equipment would result in exposure to personnel.

Pipe Decontamination

To help reduce doses to workers inside the drywell, Browns Ferry had decided to decontaminate the pipe before removing it. This would especially help alleviate the continual problem of welders incurring their dose limits at about the time they become proficient at working in the drywell. Their contract with London Nuclear Ltd. called for a contact decontamination reduction factor of at least 10.

London Nuclear had planned to decontaminate the recirculation, RHR, and RWCU piping with their Can-decon® process. They also intended to pass the chemical solution through the annulus and the recirculation pumps. A significant dose reduction could be achieved in reactor piping and in the drywell in general with this process. Their proposed equipment layout favored by health physics is sketched in Figure 5.4. The dose estimates given in section 6.3 assume extensive decontamination as discussed above. However, General Electric expressed concern about extending decontamination beyond the pipe being replaced, into the annulus, and through the recirculating pumps. If Browns Ferry reduced the extent of their decontamination, their dose estimates would have to be adjusted accordingly.

Decontamination of tools and equipment was another matter. The plant was installing a new facility capable of several decontamination techniques including electropolishing and hydroblasting in addition to using acid baths, steam cleaners, sand blasters, and freon generators.

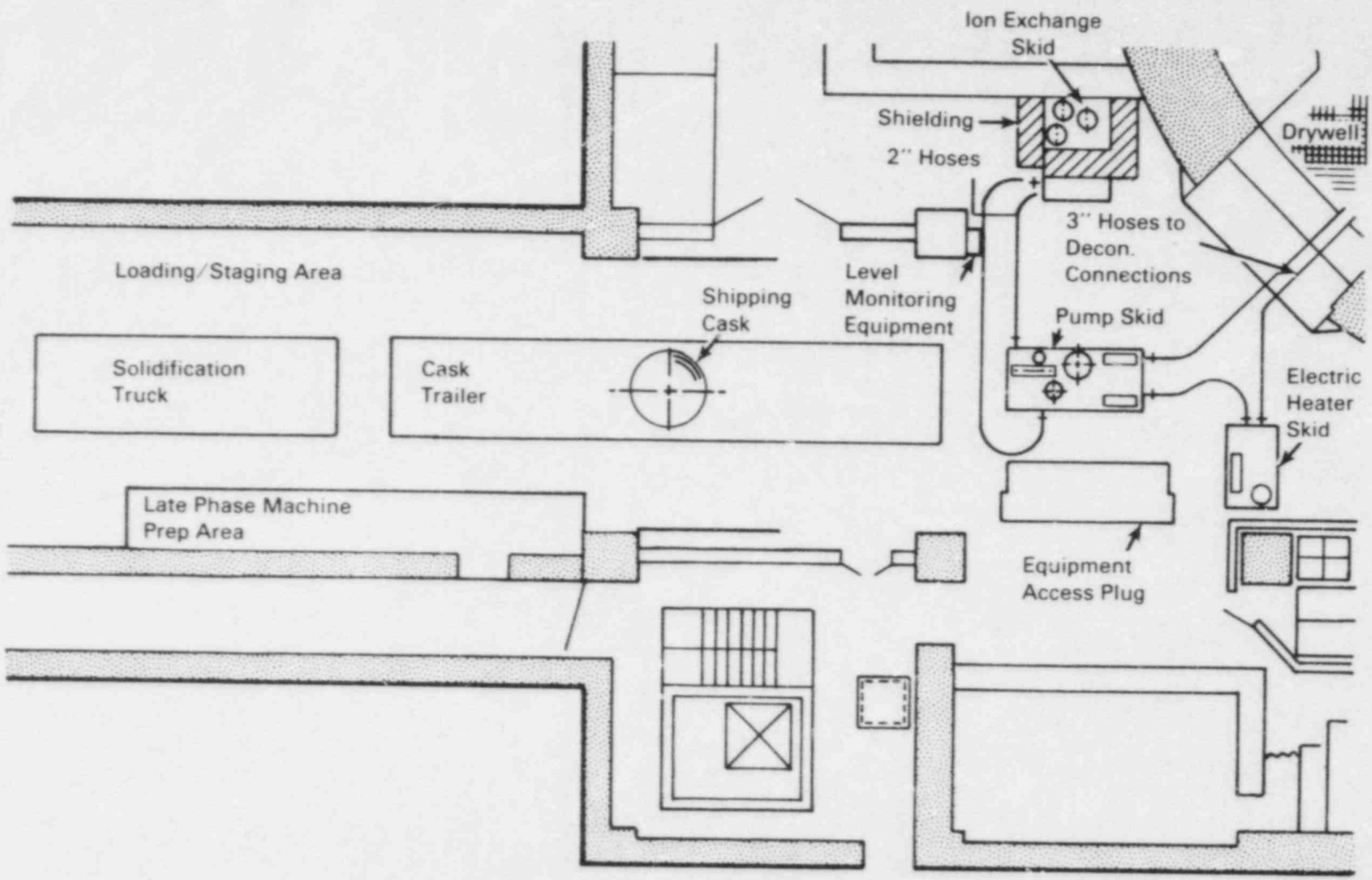

FIGURE 5.4. Proposed Decontamination Equipment Layout

Shielding

TVA's documentation shows that considerable thought went into using shielding throughout the planned pipe replacement. Shielding of most high-exposure-rate components of the reactor and associated components was planned as a dose reduction technique. Reactor piping not scheduled for replacement and located in high-traffic areas would be shielded primarily with lead blankets. No shielding would be provided around pipes being removed in low traffic areas. Cutting and welding operations require that insulation and shielding be removed around the area, and prompt removal of radioactive sections of pipe would remove source material and would lower drywell dose rates. Since setting up the shielding results in personnel exposure, it was concluded that would lower overall doses would result by removing the source material rather than by shielding it.

Water was to be removed from the annulus, and water in the core barrel was to be drained to a level below the jet pump slip joints. All control blades were to be inserted into the core before the water level was lowered. Because of the amount of work to be done around the nozzles and the negligible water shielding, plans were generated for extensive in-vessel shielding. An assessment was made to determine the probable net benefit of constructing annulus shields for placement behind the two suction nozzles and in-vessel shields for the core spray nozzles. There was considerable concern that the annulus shield would be very difficult to set into place, and there was a risk that such positioning might damage the RPV internals. To assist in determining whether to use the annulus and core spray nozzle shields, measurements of radiation levels around the nozzles in Unit 2 during the "draindown" were planned. The placement of these shields is shown in Figure 5.5.

Shielding in the biological shield in and around the recirculation and core spray piping penetrations was to be provided using soft lead bricks placed in a metal frame. About 7500 soft lead bricks were to be used for the 12 recirculation openings and the two core spray openings. Soft lead bricks would also be used to plug the ends of valves and the discharge on the recirculation pumps. A special pump inlet shield was being designed to rest evenly on the two flow splitters without damaging the pump.

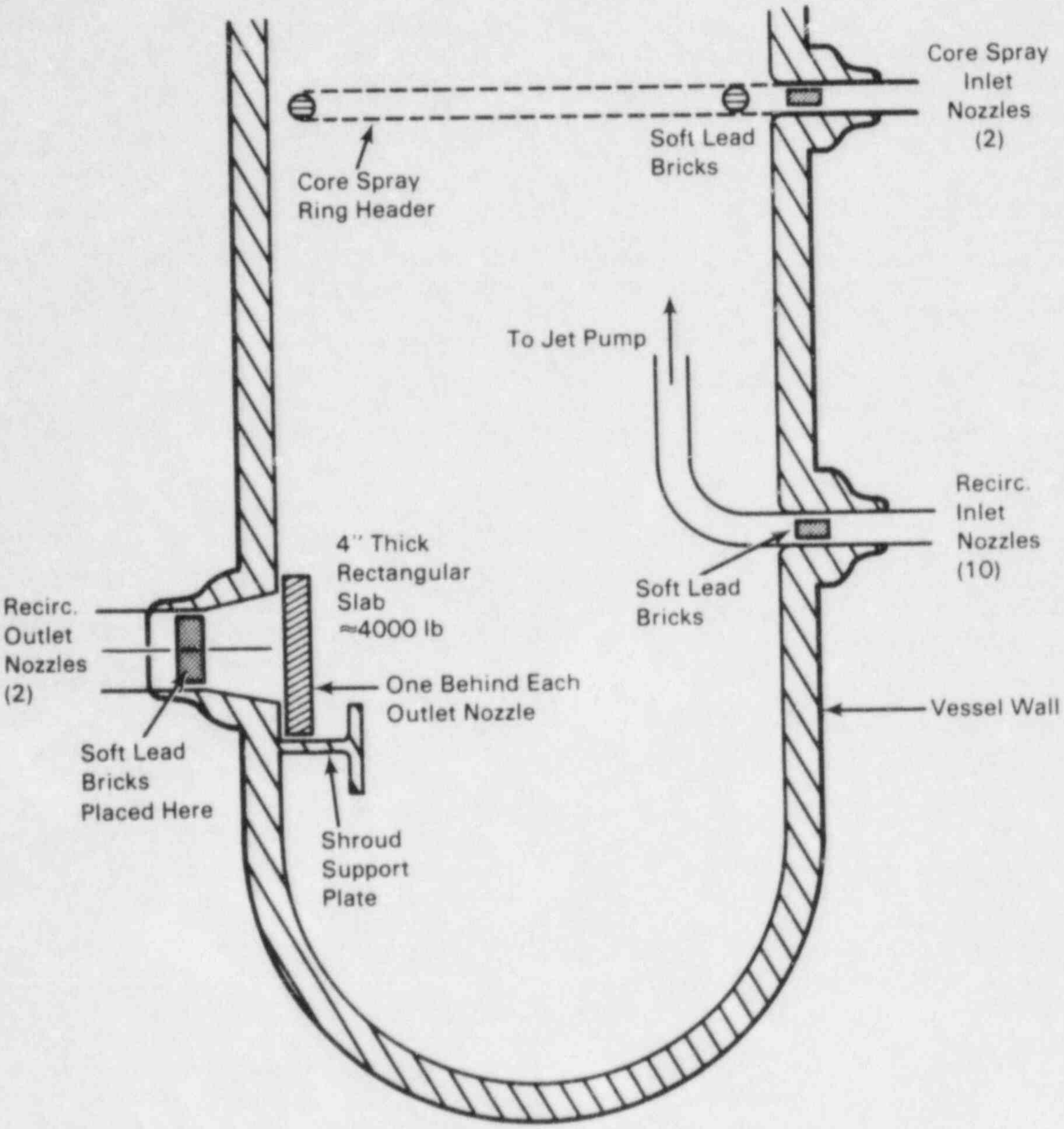

FIGURE 5.5. Proposed Shielding of Reactor Nozzles at Browns Ferry

6.0 ASSESSMENT OF OCCUPATIONAL DOSES

Reducing radiation exposure to as low as reasonably achievable (ALARA) should be the goal of any radiation protection program. This is especially important when undertaking a major outage at a nuclear facility where the potential for significant man-rem exposures exists.

6.1 STRATEGIES FOR KEEPING DOSES ALARA

Each of the six plants had ALARA supervisors/coordinators to ensure that good health physics practices and dose reduction strategies were being considered from the planning stages of the pipe replacement through the task implementation. Although comprehensive ALARA program plans and procedures suggest that the plant management is committed to the ALARA philosophy, they are not the sole indicator. The planned and/or implemented strategies to keep doses ALARA during the pipe replacement at the six plants are summarized below.

6.1.1 Nine Mile Point Unit 1

A formal ALARA/health physics program existed at NMP-1 before the recirculation system outage; it was modified and expanded to meet the needs of reducing radiation exposures to ALARA during the outage (Lempges 1982a). The major elements of the program consisted of 1) organization/job responsibilities, 2) ALARA committee, 3) ALARA procedures, 4) planning/training/mockups, 5) exposure limitations, 6) external dosimetry, 7) health physics facilities, 8) remote equipment/monitoring, 9) contamination control, 10) radioactive waste, and 11) dose tracking. Shielding, an important dose reduction technique, as well as contamination control were used extensively during the replacement.

Organization/Job Responsibilities

The onsite radiation protection organizational structure that was in place for the recirculation system repair/replacement at NMP-1 is shown in Figure 6.1. In addition, the corporate staff of NMPC provided functional and technical support that related directly to the repair/replacement efforts and the radiation protection activity. However, before the outage, the work force level of NMPC was not adequate to meet the needs of the repair/replacement effort. Therefore, additional radiation protection personnel were obtained by hiring directly or through contracting organizations, such as Radiological Chemical Technology Corporation and Newport News Industrial Corporation. Radiation protection personnel were selected based on their ability to meet the qualifications of ANSI-18.1 (1971).

Each position in the organization had a clearly defined responsibility for radiation protection. Organizational positions or groups having major radiation protection responsibilities, and reporting directly to the Superintendent, Chemistry and Radiation Management included the 1) Supervisor, Chemistry and Radiation Protection, 2) Radiological and Chemical Technology

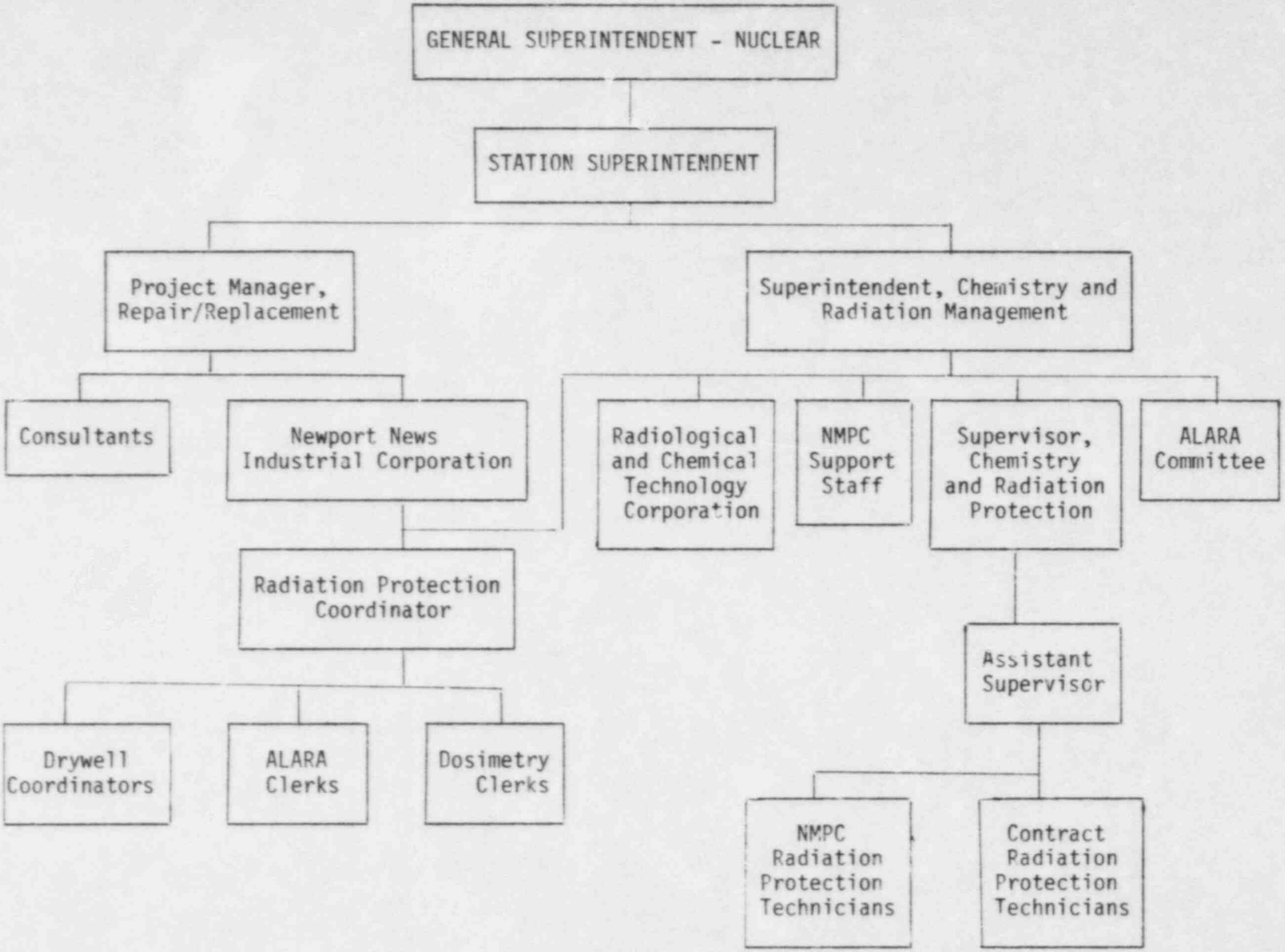

FIGURE 6.1. Radiation Protection Organization for Repair/Replacement of Recirculation System

Corporation Radiation Protection Personnel, 3) Newport News Industrial Corporation (NNIC) Radiation Protection Personnel, and 4) ALARA Committee.

The Supervisor, Chemistry and Radiation Protection was responsible for reviewing all aspects of radiation, radioactive material, and chemical control including formulating and approving procedures and policies to minimize radiation exposure in the plant and in the environment. During the recirculation system repair/replacement outage, he worked in conjunction with the NNIC Radiation Protection Coordinator to continually monitor work activities, procedural applications, and safety aspects related to radiation protection.

The Radiological and Chemical Technology Corporation provided onsite consulting services for the duration of the outage. Their specific duties included 1) review of source terms and shield designs, 2) review of ALARA procedures, 3) review of dose rate measurements and calculations, and 4) other consulting services related to radiation protection as requested.

Radiation protection staff were made available by NNIC to support the NMP-1 repair/replacement. In particular, a radiation protection coordinator, drywell coordinators for each shift, ALARA clerks, dosimetry clerks, and radiation protection technicians were provided. The major responsibility of the Radiation Protection Coordinator was to discuss and decide upon ALARA objectives with the project manager, repair/replacement. In addition, he was to maintain an overall knowledge of work in progress, provide technical assistance on radiological safety problems, review radiation survey data, and keep track of individual and collective radiation exposures. The drywell coordinators were responsible for checking that the proper protective equipment was being used for maintenance activities in the drywell. They also served as timekeepers/time study coordinators. Tabulation of exposure data was coordinated between the ALARA clerks and the dosimetry clerks based on the specific requests of management.

ALARA Committee

Performance of a task in a radiation environment with efforts made to reduce personnel exposure to ALARA is desirable. The NMPC took into account the ALARA philosophy for the activities involved in the repair/replacement of the NMP-1 recirculation system. This philosophy does not impose quantitative limits on exposure but rather guides that exposures should be maintained ALARA by examining all possible alternatives and deciding on the best solution. To accomplish this, NMPC established an ALARA committee to identify problem areas and develop recommendations for specific ALARA solutions.

The ALARA committee was chartered to determine the costs, both in exposure and dollars, and the benefits of all dose rate and dose reduction options such as shielding, decontamination, etc. Both external and internal personnel exposures were to be considered. The main responsibilities of the ALARA committee were to 1) review NNIC procedures for ALARA considerations, 2) assign ALARA tasks to committee members for execution or tracking, and 3) review man-rem and man-hour data, compare them to planning estimates, and initiate appropriate corrective action if necessary.

The permanent ALARA committee members were the Superintendent, Chemistry and Radiation Management; the Supervisor, Chemistry and Radiation Protection; the NMP-1 radiological engineer; the NMP-1 dosimetry coordinator; the NMPC health physicist; the NNIC radiation protection coordinator; and an onsite representative from the Radiological and Chemical Technology Corporation. Temporary members were added as required.

Upon completion of a particular task, a post-task ALARA review was held by the ALARA committee. The reviewers suggested improvements for reducing exposures even further on related future tasks.

ALARA Procedures

Formal procedures generated by the site or corporate headquarters provided a written statement of the NMPC position with regard to ALARA. These procedures reflected the ALARA aspects presented in NRC Regulatory Guides 8.8 and 8.10 (1975), including consideration of radiation exposure, cost, time, and quality control.

The ALARA procedures were developed by personnel in the field. Once the techniques, methods, and processes required for a particular repair/replacement task were decided upon, detailed formal ALARA procedures were written to minimize exposure for that task. Each procedure was written to be a complete document in itself and contained a minimum of references to other documents/procedures with special precautions/requirements clearly marked. Completed procedures were then reviewed and approved with signoffs by the appropriate work groups and finally by the ALARA Committee.

Planning/Training/Mockups

The performance of decontamination and the use of shielding play an important role in minimizing man-rem exposures. However, these exposure savings can be negated if an ALARA philosophy is not adopted and proper planning, training, and mockups are not used for the recirculation system repair/replacement efforts.

Planning of tasks for the repair/replacement of the primary water recirculation system at NMP-1 was given a major priority by NMPC. Alternative methods of repair and replacement were considered and evaluations of alternative techniques for specific tasks were performed. In particular, planning specifically related to reducing man-rem exposures to ALARA was considered for the recirculation system decontamination activities and in the use of 1) temporary shielding, 2) audio-visual communication equipment to minimize the number of personnel in high-dose-rate areas, 3) portable ventilation equipment to reduce airborne radioactivity levels, and 4) water shielding in the primary recirculation system where appropriate. The use of automated pipe cutting machines, welding equipment, and weld crown reduction tools was also a result of planning as early as 1979 when design and purchase of the equipment was initiated. A significant amount of preinstallation work was performed in nonradiation areas including weld preps and other machining of the safe-ends, spool pieces, and elbows. In addition, NMPC placed a major emphasis in their planning ALARA efforts on the training of workers and the use of mockups.

All personnel (NMPC and contractor) involved in the primary water recirculation system repair/replacement efforts at NMP-1 were required to attend various types of radiation protection and ALARA training sessions. These classes were held at the NMP-1 training center before personnel were allowed to work onsite. The program consisted of three types of radiation protection/ALARA training: 1) general radiation protection orientation for all personnel expected to enter a restricted area, 2) in-depth training for contractor radiation protection personnel, which was also attended by NMP-1 radiation protection staff and served as their annual retraining, 3) and specialized training for nonradiation protection plant and contractor personnel who may be required to perform self-monitoring (however, NMP-1 self-monitoring procedures require personnel to call radiation protection staff for surveys if dose rates exceed 2500 mrad/hr).

Both classroom and hands-on training were used. The following areas were covered by the training: 1) biological effects of radiation, 2) types of radiation, 3) limits and guides, 4) dosimetry, 5) shielding, 6) radiation protection and ALARA procedures, 7) protection against radiation, 8) protection against contamination, 9) personnel decontamination techniques, 10) area designations, 11) radiation work permits, 12) dose rate problem areas, 13) radiation detection and measurement, and 14) self-monitoring techniques. In addition, pipe fitters and welders were given extensive training on specialized or unfamiliar equipment, with an emphasis on ALARA, such as pipe cutting machines and automatic welding equipment. Training concentrated on equipment setup and removal operations because these operations can be very time consuming.

The construction of reactor system mockups and the use of mockup training were considered integral parts of the ALARA program at NMP-1. This was especially true during major outages for tasks having a high exposure potential. Mockups were fabricated to simulate the area(s) around the equipment to be worked on as closely as possible. This allowed workers to familiarize themselves with the physical orientation and restraints of the work area and thus helped to minimize radiation exposure by improving job performance.

Two full-scale mockups of the piping and the areas around the recirculation system inlet nozzles, outlet nozzles, and safe-ends were built. During an eight week period in 1982, 50 outage personnel (primarily pipe fitters, welders, and nondestructive testing personnel) practiced on the mockups and the specialized cutting and welding equipment. The mockup training was used to reduce the total number of man-hours required to perform the recirculation system repair/replacement efforts. This training allowed for the testing of procedures, tools, and equipment to ensure efficient functioning during the actual work. The training in the mockups included the use of anticontamination and respiratory protection equipment as would be required in the drywell depending upon the specific task. In addition, the mockups and mockup training provided the potential for identifying possible ways of improving or simplifying the tasks and thereby reducing personnel radiation exposures.

Carefully considering all aspects of a task is very important when determining whether or not temporary shielding and/or contamination control methods will satisfy the ALARA objectives. The man-rem dose savings from planning, training, and using mockups can be substantial and are discussed in Section 6.3 of this document.

Health Physics Facilities

Availability of adequate health physics facilities is an important element for an effective ALARA program. The facilities may meet the needs of the health physics program, but their location, design, and use may be counterproductive to the ALARA philosophy. Two of the potential major impacts of a poor location for the health physics facilities are:

- unnecessary increase in occupational exposures because the facilities are near established radiation areas in the plant (dosimetry issue, instrument calibration, counting room, personnel change facilities, etc.)

- unnecessary increase in exposures and the spread of contamination to personnel and clean facilities inside and outside the restricted area (contaminated equipment storage, radioactive material storage, and respirator/equipment/personnel decontamination facilities).

The health physics facilities established at NMP-1 for the recirculation system repair/replacement efforts were located and used consistent with ALARA while still meeting the health physics needs. Both permanently established and temporary facilities (such as trailers) were available for NMP-1 and contractor use. These included 1) radiation protection offices, 2) dosimetry issue, 3) instrument calibration, 4) radioanalysis/counting lab, 4) respirator decontamination, 5) respirator fitting, 6) contaminated laundry, 7) training classrooms, 8) mockups, 9) personnel decontamination, 10) change areas, 11) equipment decontamination, 12) contaminated equipment storage, 13) radioactive source/material storage, 14) radioactive waste processing/storage, and 15) low-background frisking and standby areas.

The health physics facilities were located together in the eastern part of the turbine building within easy access to the drywell airlocks on the 250-foot level but well away from (or shielded from) major sources of radiation so that unnecessary exposure and contamination would not occur (Figure 6.2).

Contamination Control

There was a significant potential during this outage for the release of airborne and surface radioactive contamination from the breaching of the primary system. This released material could either be ingested or inhaled by workers in the vicinity. The drywell at NMP-1 was considered to be a contamination zone. Therefore, appropriate procedures, instrumentation, and protective clothing and equipment were used to control the spread of radioactive contamination.

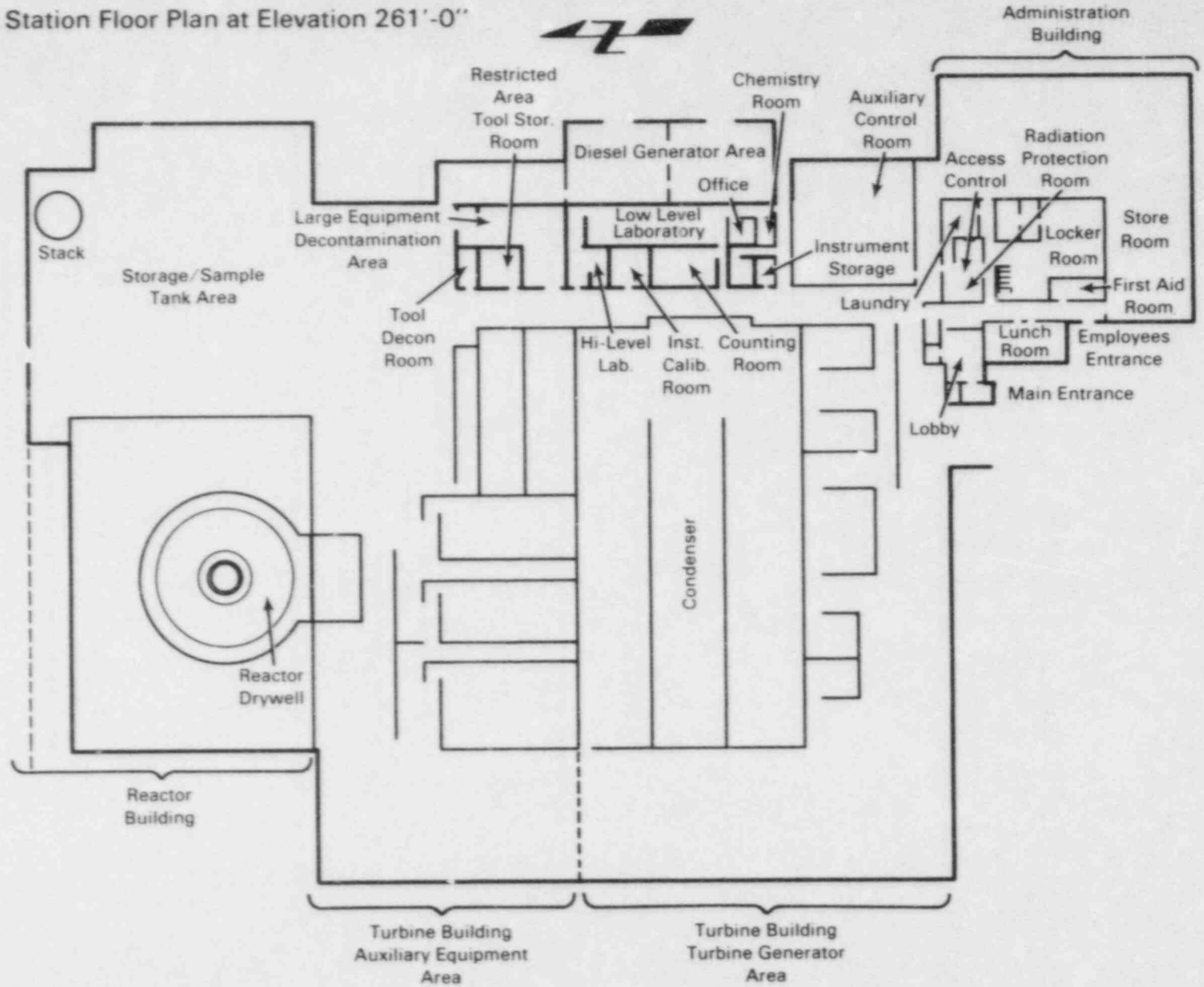

FIGURE 6.2. Location of Health Physics Facilities in Relation to the Drywell

To minimize the effect from surface contamination, NMP-1 policy was to require anticontamination clothing and equipment to be worn routinely for entry into the drywell and other areas designated as contaminated. It was also site policy to keep surface contamination levels to a minimum by sealing any openings in the primary system through the use of plugs or containment tents. Radioactive contamination can accumulate inside the drywell to the point that airborne radioactivity is possible and exposure rates are significant. Before this point is reached, areas within the drywell were decontaminated by NMP-1 personnel to acceptable levels.

Welding and cutting operations can potentially generate airborne radioactivity. Pipe cutting techniques at NMP-1 were selected to reduce this potential. In addition, tasks that could result in airborne contamination levels approaching or exceeding the levels specified in 10 CFR 20 were identified. For these tasks, portable ventilation equipment was used to control the airborne contamination levels by removing potentially contaminated air from the drywell and the worker's breathing zone. In addition, respiratory protection equipment was used to minimize the exposure to personnel. As in the case of surface contamination, sealing the primary system and decontamination methods were used to minimize the airborne radioactive contamination levels. Some respiratory protection equipment, both filtered and supplied air full face masks, were equipped with neck microphones to improve communications in the work areas thus potentially reducing exposure times. Hands-on and classroom training on the proper use of anticontamination clothing and respiratory protection was conducted onsite with special emphasis give to the ALARA philosophy.

The need for protective clothing and equipment was based on surface contamination surveys and airborne monitoring surveys conducted by the Radiation Protection Department personnel. These surveys were performed before entry into a work area and during the actual performance of the work. The effectiveness of these control measures was determined by conducting contamination surveys of personnel as they left the work area and through the use of the site biosurveillance program (whole-body counting, urinalysis, thyroid counting, etc.).

6.1.2 Monticello

The ALARA organization at Monticello consisted of elements from both the General Electric and Northern States Power Company organizations. The ALARA engineer, from General Electric, was responsible for chairing the ALARA review team and conducting a review of work task procedures as necessary. The ALARA engineer was also responsible for interfacing with the MNGP supervisor, Radiological Services. The Supervisor, Radiological Services chaired the outage ALARA committee and was informed of any problems by the NSP, Shift ALARA Coordinator.

Health physics (HP) assistance was provided by the Institute for Resource Management (IRM). They had been employed by MNGP before, and most of the personnel were familiar with the Monticello plant.

Five HPs were used on each shift throughout the outage. Two were located at the desk near the entrance to the drywell, two were inside the drywell, one was the supervisor and the fifth was on break. The HPs worked two 12-hour shifts and worked 12 out of 14 days.

There were 12 TV monitors placed in the drywell. Each monitor had its own pan and tilt control. The HPs had a 12-position scanner so they could view any of the 12 cameras. All 12 of the cameras were continually videotaped. The supervisors located on a work platform on the drywell access also had access to video monitors. Headsets were used for communication.

Initially the HPs felt that too many people were in the drywell at one time. There was also some difficulty with the coordination of all groups: insulators, electrical, plant and mechanical. Finally, each group was limited to a certain number of people in the drywell. Even so, there were problems with all the personnel taking breaks at the same time. This created long lines at the HP desk where the employees were required to log in and out and pick up their dosimetry and procedures. This, in turn, reduced the work time available. An effort to stagger the breaks was not successful.

An alarming portal monitor was used to indicate skin contamination. The portal alarmed to greater than 100 counts per minute above background. There were incidents of skin contaminations that were easily cleaned with soap and water.

The training program consisted of a full 8 hours of general employee training, as well as specific radiation protection training as needed. Pipefitters and welders were given training on specialized or unfamiliar procedures and equipment, such as automatic pipe cutting, weld preparations and welding machines. Training efforts concentrated on the setup and removal of equipment, since the setup and removal is usually time consuming and significant radiation exposure can be saved by performing these operations efficiently.

Mockups were fabricated to verify the adequacy of procedures and tooling and to provide training for individuals involved in selected work tasks. They were also used to determine ways to simplify the task or to improve the radiological and quality control. Mockups were provided for the discharge safe-ends, the suction safe-ends, the closure spool measuring device, the piping elbow, the jet pump instrument safe-end, and the standby liquid control safe ends.

Contamination Control and Ventilation

Full face respirators were used during the pipe cutting. Respirators were required for concentrations greater than 0.25 MPC. Respirators were also used during thermal sleeve alignment because of high contamination levels in the nozzles. Decontamination of the nozzles was not effective enough to eliminate the respiratory requirements. Although a large number of respirators were used, there were no significant internal depositions.

Hardhats were required in contaminated areas unless a waiver was completed. The majority of the workers did not wear hardhats.

Contamination was generally not a problem for the majority of drywell work. Anticontamination clothing was worn for entry into the drywell and other areas designated as contamination areas. A separate controlled area was established and maintained around the discharge nozzles, confining contamination to these areas until the safe-ends were welded in place. Double coveralls and respirators were worn by personnel working on the nozzles. When the discharge nozzle work was completed, the controlled area around the discharge nozzles was decontaminated and the double protective clothing and respiratory requirements were lifted.

Tools and equipment were decontaminated with a sandblaster, high-pressure freon, freon boil sump and ultrasonic vapor degreaser, and freon hose cleaner. The refueling floor also contained areas for sandblasting and hydrolasing.

A containment fabricator contractor was present at the site during the outage. Glovebags were fabricated for the safe-ends to minimize airborne particles during the hydrolasing that removed loose oxides on the safe-end after piping removal.

Localized air ventilation was used for grinding operations and for weld preps. The flow of air at the open safe-ends was from the drywell and into the reactor vessel. The head was held up about 4 in. using three blocks, and four ducts were run into the vessel. The gap between the vessel and the head was sealed with herculite-reinforced plastic and NG duct tape. The ducts carried air to a portable HEPA filter unit on the refueling floor.

6.1.3 Cooper

The Cooper plant maintained two separate committees to review ALARA concerns. The ALARA committee consisted of representatives of CNS, CB&I, and Construction Management. The CNS technical staff assistant chaired the committee. The chemistry and HP supervisor and the ALARA coordinator also attended representing CNS. The CB&I project manager, CB&I ALARA staff supervisor, and ALARA staff from subcontractors were assigned to the committee. Construction management in attendance included the construction manager, technical supervisor, and their ALARA consultant.

The second committee, named the TWP Review Committee, was set up to address ALARA and technical problems. This committee was made up of the technical and ALARA or HP personnel mentioned above along with the construction management covered by the individual TWP being reviewed.

The plant and construction ALARA people reviewed all work instruction (WI) and task work package (TWP) plans specifically for ALARA concerns and incorporated their comments into the plans. Revisions had to be approved by the ALARA coordinator, construction senior engineer, and the ALARA consultant.

The most current radiological conditions were to be known at all times at the control access point. Each worker entering the drywell knew the conditions in the work area just before entry. In addition, special precautions about high-radiation or high-contamination areas within the drywell were posted.

The drywell was controlled as a special work permit (SWP) area throughout the IGSCC outage. Most of the SWPs were written in advance of the work. The number of SWPs required were based on the ALARA job reviews and work packages submitted by the contractor. The SWPs were grouped into major phases of the outage, and the ones required to complete a phase of the work were generated and implemented at the same time. The major phases of the IGSCC project were categorized as:

1. Prepipe decontamination work
2. Pipe decontamination
3. Pipe cutting and removal
4. Safe-end/thermal-sleeve removal
5. Safe-end/thermal-sleeve replacement
6. Pipe replacement work
7. Postreplacement work
8. Pipe disposal.

Even though a majority of the SWPs were written before a phase started, additional SWPs were initiated when unexpected work came up or radiological conditions changed. The CNS HP or the lead health physics technician (HPT) decided whether a new SWP should be initiated or whether an existing SWP would suffice. Jobs in the same area that fell under the same radiological requirements and had essentially the same restrictions were grouped under the same SWP.

CNS used task work packages (TWPs) to control work and track dose. Hold points were set up to monitor and control the work. The SWP forms had to be signed and dated before the work could continue. Generally, nothing could be done without getting health physics' approval. This system was set up largely because of the lack of radiation experience of the contractor. There was difficulty making the system work, but it was eventually accepted. The authority given the HPs and HPTs helped implement good work practices. Contractor personnel were fired after their second "health physics offense." This got the workers' attention and forced them to follow practices they had learned in training.

The drywell at Cooper Nuclear station is small and contains many interferences that restrict movement. Personnel at the control point monitored drywell activity and limited personnel entry to maintain a safe and orderly work progression.

Cooper had a lead HPT, a lead drywell technician, a rover, and additional HPTs as necessary assigned to specific jobs in the drywell. As a deliberate ALARA practice, the number of HPs in the drywell was limited to as few as possible (usually two unless more were needed). The plant supplemented its workforce with contractor HPTs.

The ALARA committee set up a structure whereby job categories would be evaluated based on expected man-rem. These categories were <1, 1-25, and >25 man-rem. The idea was that an extensive review should be conducted of jobs falling in the higher classifications to determine whether they could be completed with less required dose. This system really was not necessary as all jobs were reviewed whether they were going to require 1 or 25 man-rem.

Six television cameras with zoom lens and automatic focus were specifically dedicated to the ALARA/health physics staff during the outage. During the early part of the outage and through old pipe removal, this saved exposure because the supervisors and health physics staff could observe the work on remote monitors. However, some cameras were damaged during transport of new materials into the drywell and others were blocked by equipment or scaffolding. This removed many areas from remote viewing.

Identification of HPTs was a problem during the pipe replacement. Station personnel generally do not wear hardhats inside containment where the hats are considered more of a hazard than a benefit. The construction crew, however, did not feel safe without their hard hats. Most construction crafts had colored hats or colorful stripes on their hats to identify their trade. Health physicists were initially put into nondistinguishing hats. This caused confusion with the workers and quality assurance personnel so the HPs turned to another method of identification--blue arm bands.

Classroom instruction and mockup training were required of personnel performing work in critical areas. The mockup training included wearing anti-contamination clothing and respiratory protection when required for performance of the task. The instruction and associated training ensured that the workers could perform the task to acceptable standards in the minimum time before entering the critical areas, and that the equipment was operating properly.

Protective Clothing and Respiratory Protection

Cooper employed a decontamination crew during the whole outage. When an area inside containment reached 5000 dpm, the decontamination crew cleaned it up. The constant cleanup reduced problems occurring with the external contamination that would become loose and flaky after 6 to 8 weeks.

A tentmaker was hired to assemble isolation tents of herculite and clear plastic for work where airborne contamination could be a problem. The clear plastic allowed the HPTs to observe pipe cutting and welding from the outside. A HEPA-filtered suction ventilation system was available in the cutting and grinding areas to remove airborne particles as they were created and direct the flow of air away from workers. An additional ventilation system was used on the 1001-foot level to pull a negative pressure on the reactor vessel.

The very low airborne contaminant levels in containment reduced the need for respirators except around nozzles, valves, and pumps. After the old pipe was removed, the nozzles were sealed and the new welds were made, so no respiratory protection was required. The plant used the limit of 3×10^{-9} µCi/mL for respiratory protection. The highest level detected was 10^{-7} µCi/mL and the typical concentration was 10^{-9} µCi/mL. Filter masks with supplied air provided the respiratory protection when necessary.

For N1 and N2 pump work and RHR tie ins, protective clothing initially included a double layer of anticontamination coveralls and plastics, but there contamination was much less than expected. After the first nozzle was removed, only one pair of coveralls was necessary. The need for a plastic layer depended on the job.

6.1.4 Peach Bottom Unit 2

An ALARA program was established with the main objectives of mitigating exposures from primary radiation sources in the drywell. The ALARA group organized for the pipe replacement outage included health physics and engineering support from PECO, CB&I, and GE (Figure 6.3). The CB&I radiological engineers were given the ALARA responsibility during the replacement outage. Tasks were evaluated before initiation for access routes, work location, duration of operation, stay times at various locations, and expected dose rates in the area. Each task was categorized according to the probable dose required to complete it. Those tasks with estimates exceeding 25 man-rem required approval by the PECo support HP.

A number of specific actions that could potentially save exposure were identified and implemented. The effectiveness of each action was evaluated following its implementation and documented on an ALARA checklist that included the task and purpose, exposure, schedule impact, cost, lessons learned, and recommendation. The mitigating actions included the following in-vessel and in-drywell actions.

In-vessel mitigating measures:

- full insertion of the control rods
- use of jet pump slip joint to maintain water shielding in the reactor vessel
- hydrolancing of the crevice in the discharge nozzles
- placement of in-vessel shielding for the suction nozzles
- removal of peripheral control rods
- vacuuming of crud traps in the vessel
- use of jet pump nozzle plugs for water shielding in the reactor vessel
- operation of the reactor water cleanup system.

In-drywell mitigating measures:

- chemical decontamination of piping, valves, and pumps
- shielding of penetrations in the biological shield
- shielding of open suction and discharge nozzles
- hot spot shielding in the drywell
- mockup training
- valve rebuild and weld preparation location
- drywell decontamination.

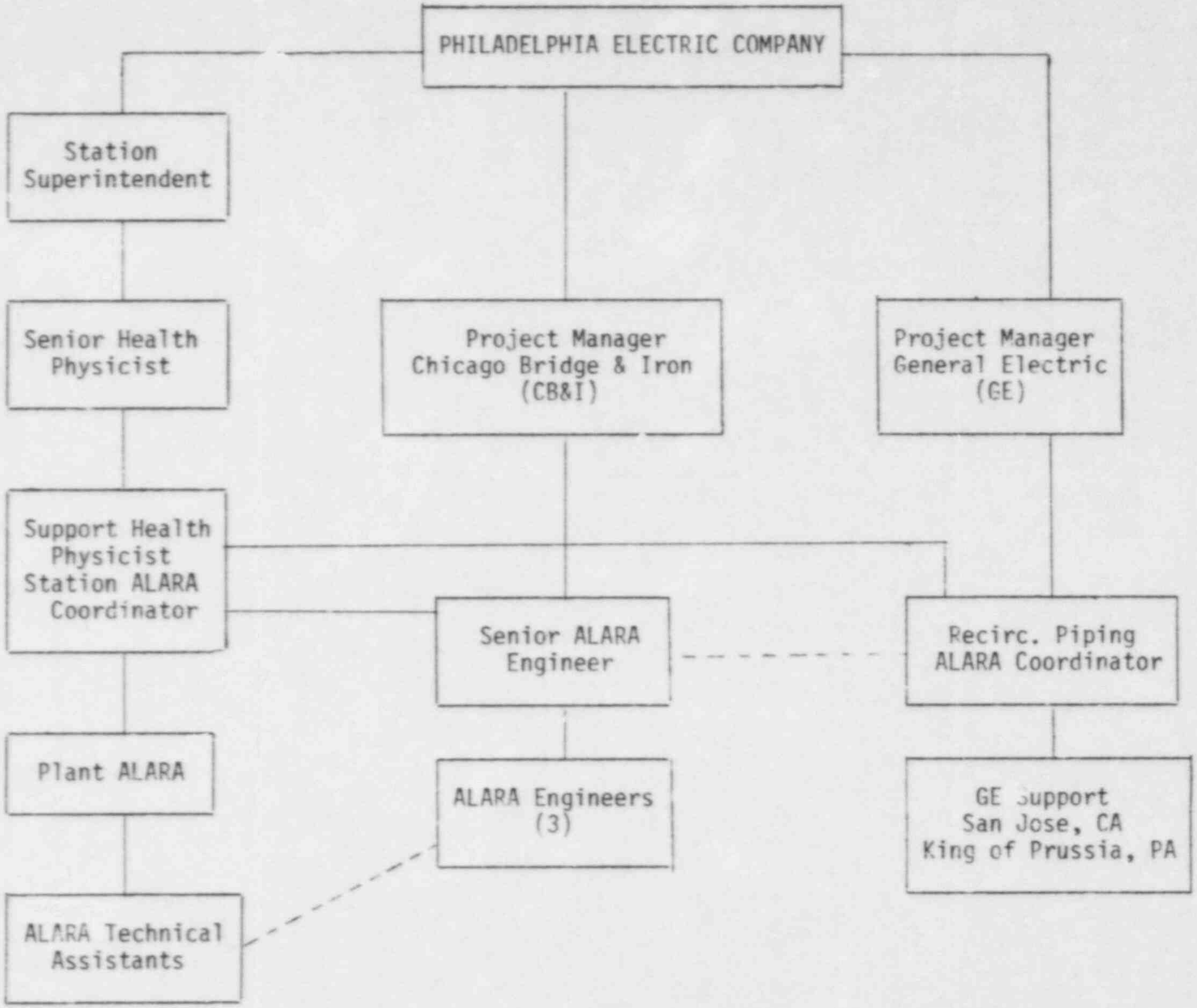

FIGURE 6.3. Peach Bottom ALARA Organization

Planning, post-task, and daily progress meetings were held to discuss specific tasks and keep radiation doses ALARA. Attendance at these meetings included the PECo ALARA administrator and CB&I radiological engineers. The ALARA administrator acted as the PECo liaison between CB&I administration, crafts, and radiological engineers. The shift radiological engineer was responsible for reviewing radiological conditions and informing personnel, reviewing the work on a continuing basis, and keeping track of individual and collective doses.

Mockup training was a major component in the control of worker exposure. Mockups set up outside of radiation dose rate areas were used to experiment with different setups, techniques, and procedures before the actual pipe replacement task began. ALARA awareness training conducted along with mockup training was thought to result in dose reduction factors of 2 to 2.5. During mockup training, personnel successfully demonstrated the use of procedures and equipment before the actual performance of a pipe replacement task. Quality assurance purposes were also served by demonstrating that the work could be performed to required tolerances. Mockup demonstrations used included temporary valve supports, cutting for pipe removal, nozzle loading, weld preparation, joint fit-up, welding operations, and NDE examination. Several demonstrations were performed with the same mockups.

Remote TV cameras and monitors were used in the drywell during pipe replacement activities. Day-to-day progress, quality of work, and ALARA considerations could thus be monitored from low-dose-rate locations.

Respiratory Protection and Contamination Control

During most of the early phases of work (April 27-August 2, 1984), decontamination of exterior surfaces in the drywell was successful in maintaining airborne radioactivity below respirator requirement levels at and above the 135-foot elevation. The surface decontamination methods included low-pressure, low-temperature water flushing with detergents and/or hand wiping of surfaces to remove loose radioactive materials. Daily housekeeping and removal of materials maintained air quality below respirator levels until pipe removal was initiated and then respirators become necessary.

A major clean up and decontamination effort in December 1984 was successful in lowering the airborne radioactivity concentrations at all levels to below respirator required levels for the remainder of the project. The effort included a washdown of all surfaces painting the floor on the 119-foot elevation, and re-establishment of adequate ventilation. The benefits of good ventilation were readily apparent when spikes in drywell air activity caused by burning, grinding, flapping, welding, etc., (for which station policy required respirators) rapidly returned to normal levels.

Glovebags were also used to control airborne contamination. They were used to isolate the suction and discharge nozzles and all safe-ends during cutting, grinding, and welding activities.

Personnel protective clothing included anticontamination coveralls, plastics for work in areas of liquid contamination, and respirators in areas of airborne contamination.

Freon hydroblitzing of tools decontaminated them for reuse.

6.1.5 Vermont Yankee

Vermont Yankee's stated goal of their radiation exposure control program is to ensure that occupational exposures are maintained ALARA with an integrated approach to exposure control and reduction. They believe that they made a conscientious effort to learn from previous industry experience and benefitted from their aggressive operational radiation protection program. This program of occupational radiation protection is controlled by the Vermont Yankee health physics staff who perform radiological coverage and perform routine and RWP-related radiological surveys. They provide protective clothing and respiratory protection, nonroutine dosimetry processing and routine dose tracking, radiological surveys, and counting instrumentation. Morrison-Knudsen provides radiological engineering and ALARA support staff and interfaces between work groups.

Procedural Controls

The Radiation Exposure Control Program was designed to use existing procedures where possible and supplement them with specialized procedures that address additional needs. Three types of procedures are planned to instill and maintain proper control. Controlled work packages, which consist of site work instructions, material data sheets, process control documents, drawings, and an ALARA checklist, are set up by work function. These work packages undergo a review cycle of all disciplines to ensure that ALARA strategies are implemented.

The second type of control procedures is the RWP, which promotes "carry through" of ALARA concerns and provides a refresher of radiation protection requirements before initiating the task. The health physics staff prepares the RWP. The proposed work is reviewed for ALARA considerations, and the ALARA requirements are noted on the permit.

The third type of procedures is the other plant health physics procedures and regulations to ensure effective minimization of exposure, such as criteria guidelines AP0501 (Radiation Protection Standards), AP0503 (Establishing and Posting Controlled Areas), AP0505 (Respiratory Protection), and AP0506 (Personnel Monitoring).

Contamination Control

Existing drywell ventilation is being used for continuous air turnover, moving air from less contaminated areas through more contaminated areas. A 16,000 scfm ventilation capability in the drywell with manifolds and multiple suction lines was installed to ventilate work areas. HEPA filter units are in use outside of the drywell to help control airborne contamination.

To control the extent of contamination, mechanical cuts rather than plasma arc cuts were planned for all major piping. The change primarily to plasma arc cuts caused few contamination problem, although it necessitated additional use of glove bags and respirators. Localized HEPA ventilation units, portable HEPA vacuum cleaners, glove bags, and tents are being used in conjunction with general and area decontamination to keep the loose surface and airborne concentrations down.

Surface contamination in the drywell is being continuously cleaned by a decontamination crew. Good housekeeping practices are also being emphasized. Operations such as grinding, buffing, and welding the inner pipe surfaces of contaminated piping were minimized. Mechanical counterbores performed on these pipes generally removed high-level fixed contamination. Catch trays to contain pipe shavings and general area drop clothes were used during mechanical cuts. End caps were tack welded over the open ends of the contaminated pipes. Shield frames were built inside pipe shipping boxes to reduce dose rates to shippable limits.

Personnel in the drywell wore one or two layers of anticontamination coveralls depending on their work location. Those working on the safe-ends or nozzles also wore respirators. Plastic outer clothing was required for work around water or mist.

A contaminated tool crib will be set up to handle small tool decontamination. This crib will be set up in the immediate proximity of the drywell. Shielding will be provided as necessary. Containment areas will also be established outside of the drywell to supplement the decontamination operation inside the drywell. Smears for alpha and beta contamination will be performed on cut pipe surfaces.

Worker Training

General employee training is received by all personnel working in the radiological controlled areas at Vermont Yankee. This training includes instructions regarding the procedures and policies governing safe performance in a radiological environment. During this training, the concept of ALARA is emphasize and stressed as being ultimately the responsibility of each individual.

Morrison-Knudsen (M-K) provides training for the replacement project, which covers project orientation, safety, security, work rules, quality program indoctrination, and health physics. To help accustom contractor personnel to the equipment and working conditions inside the drywell, extensive mockup training precedes actual operations. This not only acquaints the employees to working while in protective clothing, but also is used to correct mistakes in technique and to allow workers to practice ways to reduce exposure.

Mockups to be used in the training may include a multiple nozzle setup to teach installation, use, and removal of remotely operated cutting and weld preparation equipment; a riser to learn the templating method; and pump and valve casings for practice in remotely operated cutting, weld preparation, and automatic welding.

The training is designed to be thorough enough so that workers completing the program will be qualified to perform their activities efficiently, thus minimizing exposure.

The mockup training program is also beneficial to supervisory staff in a variety of ways. Through use of mockups, techniques and equipment can be tested to verify efficacy; parameters such as cutting speeds and welding machine settings can be determined; and time schedules can be developed or verified based on trials under simulated conditions.

6.1.6 Browns Ferry Unit 1

Although it is obvious from the Browns Ferry pipe replacement planning document that TVA had given considerable thought to dose reduction and ALARA techniques in their impending pipe replacement, little is documented as to their organization and procedures. It was intended that HPs from TVA and contractor organizations would work together and interact with plant management, support services and contractor organizations to keep doses ALARA.

The plant health physics organization anticipated responsibility for ALARA planning and work package review in addition to its more routine duties. No major modifications of health physics practices and procedures were expected. It was planned that an outage group would provide daily health physics coverage specific to the project. This group would report to the plant HP. An ALARA engineer along with several HPTs would be assigned to evaluate tasks with significant potential for exposure and recommend methods for exposure reduction. Bartlett Nuclear would provide additional HPT support.

A project HP assigned to the pipe replacement project organization assisted in planning the intended replacement. He also reviewed all work packages and was expected to make recommendations on good radiation protection practices.

M-K also planned to have an health physics group onsite. This group's responsibilities appeared to overlap considerably with the coverage and review provided by plant staff and it is assumed that they would provide assistance to the plant HPs.

The Browns Ferry site has a large, new training facility to accommodate the influx of workers during major repair outages. Browns Ferry has mockups of the control rod drive module and a jet pump nozzle used in training. They were planning to use recirculating suction and discharge nozzle mockups for specific pipe cutting and welding training. Structures simulating the bio-shield, reactor vessel and some of the associated equipment were being considered to provide more awareness of pipe and nozzle positions in the drywell. Mockups for practice installing lead bricks into metal frames had also been planned. It was anticipated that practice with the mockups and review of techniques at the prejob ALARA meetings would reduce the time workers spent in containment.

Several procedures specifically intended to keep personnel doses ALARA included the extensive use of video cameras and radio communications, which would reduce the number of people needed and perhaps the time required in the drywell to perform various task An elaborate ventilation system with coolers was designed to reduce h t stress and fatigue in the drywell during the hot summer. Frequent meetings to discuss task status, track personnel doses, and develop strategies were also planned to help keep doses ALARA.

Contamination Control

An ongoing drywell decontamination task was anticipated to keep the extent of contaminants as low as possible. Intermittent wiping and HEPA filter vacuuming were planned. Procedures would be required to clean, wrap, and store equipment removed from the drywell.

The use of tents to reduce airborne radioactivity during cutting, grinding, and hydrolasing procedures would be initiated as a precautionary measure. If tents were found to be unnecessary after the initial cuts, their use would be eliminated unless specific conditions warranted them. Because Unit 1 has a history of fuel leaks, air samples would be monitored for alpha emitters. Continuous air monitors in the drywell would monitor alpha and beta emissions.

To help reduce the amount of time workers would be required to wear masks, a temporary ventilation system was designed to remove airborne radioactive material as quickly as possible. The system (Figure 6.4) would have a capacity of 26,000 cfm. A 9,000-cfm cooling unit would also be available to make the drywell temperatures more comfortable and therefore make work more efficient during the summer months. All air movers would be outfitted with HEPA and possibly carbon filters upstream to clean the air and prevent contamination of the fans.

Browns Ferry planned to use its regular protective clothing guidelines for personnel working in the drywell. One pair of anticontamination coveralls are used for contamination levels above 1000 dpm/100 cm^2. Two layers of coveralls are used as necessary at the discretion of the HPs. Plastic outer clothing is required when working around potentially contaminated mist or liquid.

To avoid spread of any remaining contamination of the removed pipe, plastic caps were to be placed over the ends, and the pipe was to be wrapped in plastic and placed in boxes outside the drywell. Surveying, sealing, and removing the pipe would be expedited to avoid increasing dose rates in the reactor building.

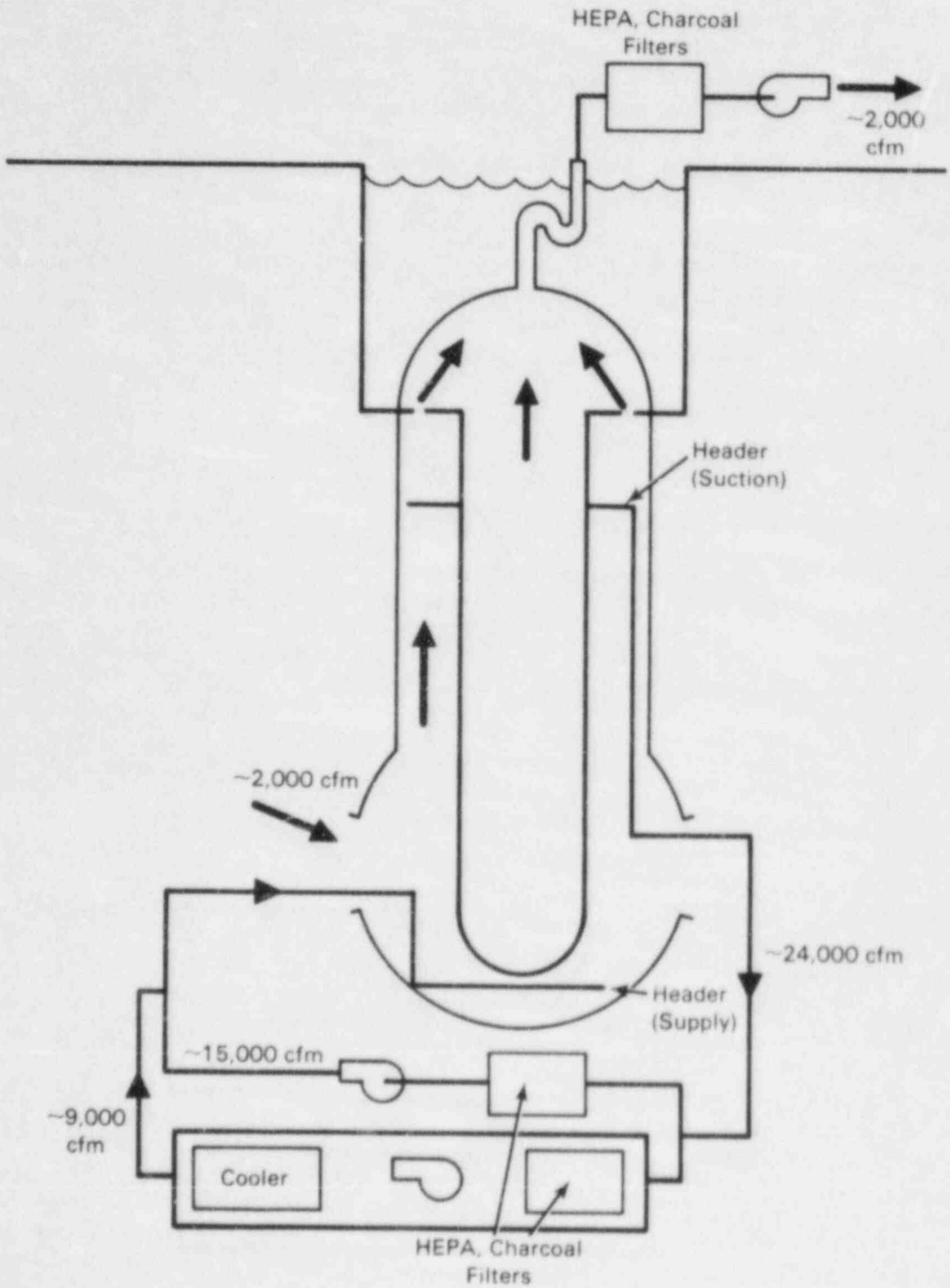

FIGURE 6.4. Browns Ferry Temporary Drywell Ventilation

6.2 TECHNIQUES FOR MONITORING EXPOSURES

Proper running of an ALARA program can only be accomplished if the measurement and tracking of occupational dose is effective. Administrative dose limits differ by plant as do the types of personnel dosimetry. The dose limits, personnel dosimetry, and frequency of readout for each plant are discussed below.

6.2.1 Nine Mile Point Unit 1

The exposure limitations program at NMP-1 consisted of procedures establishing policy for the licensee's external exposure guides to meet the concepts of ALARA. These exposure guides were:

- 100 mrem/week
- 1,000 mrem/calendar quarter
- 4,000 mrem/calendar year.

However, to meet work demands, there were provisions to allow staff members to exceed the radiation exposure guides with the approval of the 1) employee's supervisor, 2) Supervisor, Chemistry and Radiation Protection, and 3) Station Superintendent. Only oral approval from the employee's supervisor was required to exceed the weekly guide. Written approval was necessary to allow workers to exceed the quarterly or yearly limits. Contractors were not allowed to exceed 1,250 mrem/calender quarter without a current NRC-4 form on file. The external exposure limitations program was established to set guidelines for personnel safety. External dosimetry was used to track personnel exposures and to ensure that the exposure guidelines were not exceeded. The same limits were used for all NMPC staff and contractors.

External Dosimetry

The control of personnel external exposure at NMP-1 during this outage was accomplished by the use of exposure measuring devices and a records system for documenting doses. The external exposure measuring devices used by the licensee were a vendor-processed film badge, licensee-processed thermoluminescent dosimeters (TLDs) both for whole-body and extremity measurements, and pocket ionization chambers (pencil dosimeters).

The film badge used was supplied and processed by Landauer, a dosimetry vendor, on a semimonthly exposure period rotation and was designed to measure gamma, x-ray, and beta radiation. An additional film package supplied by the vendor was used when neutron dose rates exceeded 2 mrem/hr or the total dose was expected to be greater than 5 mrem. The licensee's policy was to provide beta-gamma film badges for all personnel entering the restricted areas, and to provide neutron film badges for all radiation protection personnel and other personnel as needed when neutron radiation was present. Obviously, neutron badges were not used during the outage.

The licensee's TLDs consisted of a clip-on holder with three positions for LiF TLD chips, one of which had a thin window for beta measurement. Processing and recording of TLD data was performed by radiation protection technicians in conjunction with dosimetry clerks. It was the policy of NMP-1 to use the TLD as a backup device to the film badge, the latter being used as the legal record. TLDs were required to be worn in high-radiation areas or as required by the Chemistry and Radiation Protection Department. In addition, TLDs were required to be worn when personnel exceed the following whole-body doses: 2000 mrem per quarter and/or 4000 mrem per year. This provided a mechanism for rapid process turnover in evaluating personnel exposure.

Self-reading pocket (pencil) dosimeters were also used in conjunction with the film badge and TLD. It was NMPC policy to require the wearing of pencil dosimeters for entries into restricted areas. The pencil dosimeter served as backup mechanism when TLDs were not used. If both TLDs and pencil dosimeters were used, the higher reading was recorded for the person's exposure file until the film badge could be processed. The responsibility for reading, rezeroing, and recording pencil dosimeter results was left up to each staff member.

The use of three separate systems in conjunction, and as back-ups to each other, provided an excellent tracking of personnel man-rem exposures. This multiple dosimetry data and the dose rate measurement data collected from survey instruments and area monitors allowed the Radiation Protection Department and the ALARA committee to control exposures to ALARA. The man-rem estimates and actual values presented in Section 6.3 of this document are based entirely on the dosimetry and instrument survey data in conjunction with man-hour estimates for tasks where applicable.

Dose Tracking

All personnel entering the restricted area were issued pencil dosimeters to measure external radiation doses. The pencil dosimeters were used to aid the workers while at the job location and were read and the readings recorded each time a worker exited the drywell. A TLD was issued to workers entering high-radiation areas and for other specialized work environments as directed by radiation protection personnel. The processing of TLDs was performed onsite with around-the-clock coverage. Day-to-day dose tracking was achieved by entering the pencil dosimeter readings into a computer, which summarized daily personnel exposures by individual and task. This summary report was distributed to site supervisors twice a day for their review. Reports were issued before the next operating shift (12-hour shifts) to ensure close monitoring of radiation exposure. The reports also included the recirculation loop number, RWP number, total exposure, and total man-hours for each daily individual and task exposure listing.

In addition to the dose summary reports, tracking of dose was accomplished by the radiation protection technicians ensuring that ALARA practices were followed for work activities in the drywell. In this way, improper ALARA practices could be corrected before they became exposure problems. Job performance in terms of ALARA was based on dose tracking by job. Every job listed in the man-rem estimate was tracked with the computer, which was able to compute the man-rem accumulated per job as well as compare the actual man-rem totals with the projected man-rem estimate. By comparing a job's man-rem total to its estimate, its performance from an ALARA standpoint could be instantly assessed. The jobs were further divided into tasks. Each task in a job was tracked for man-rem and man-hour accumulations. If a job began to exceed the estimated man-rem, the tasks were evaluated to determine the reasons for the discrepancy.

6.2.2 Monticello

The radiation workers in the drywell at Monticello were equipped with low-range and high-range pocket dosimeters. The high-range dosimeters had a range of 0-1000 mR. For high-dose-rate areas, the workers had Xetex alarming radio-transmitters. Three receiving units were procured, and each unit could track five dosimeters. The receiving units were stationed at the health physics table. There were some problems using these dosimeters. First, the radio transmitters were not very rugged and required considerable maintenance. At the end of the outage, 5 of the 15 transmitters were not repairable. Second, the frequency of the Xetex instruments was similar to that used by the guards for their communications, and occasionally the readings were influenced by transmissions sent by the guards. Despite the disadvantages, it was felt that the Xetex system saved considerable dose that would normally have been spent on radiation protection coverage.

Extremity dosimeters were placed on personnel working on the nozzles and in high-radiation fields such as the recirculation pumps and the feedwater spargers. Extremity evaluation was required when the uncorrected contact reading was three times the whole-body reading.

Dose tracking was performed with the pocket dosimeters. The doses were updated every 12 hours. An Eberline representative was present during the outage to read the TLDs that were used for the permanent record. The TLDs were read on a monthly basis, or as needed. An IBM computer system was used to keep track of the doses.

Monticello Nuclear Generating Plant has a dose limit of 1000 mrem/quarter. To exceed this limit, a NRC Form 4 must be completed and the employee can then receive up to 2000 mrem/quarter. Exposure requests may be made with management approval to increase the dose to 2500 mrem/quarter, and again to 2800 mrem/quarter. The annual dose limit for NSP facilities is 4500 mrem/yr with a limit of 5000 mrem/yr including dose incurred at non-NSP facilities. Women are limited to a dose of 120 mrem/quarter or 200 mrem/yr unless they sign a waiver.

6.2.3 Cooper

Cooper's whole body exposure quarterly administrative limit is 1,000 mrem. Exposure above this limit to CNS personnel requires written approval of the department supervisor and the CNS chemistry and HP supervisor. For contractor personnel, exposures in excess of 1,000 mrem/quarter requires mutual agreement in writing between NPPD and the worker's employer provided that the worker has an up-to-date NRC Form 4s and has exposure left under the 5(n-18) rule (10CFR 20). A limited number of key individuals received up to 2.75 rem/quarter during the pipe replacement outage.

Each of the TWPs set up specifically for the pipe replacement was carefully tracked for dose incurred. All individuals entering the drywell signed in under the appropriate SWP. Their pocket dosimeters were read after they left containment and their doses were recorded in a real-time computer system. Their total authorized and remaining exposure could then be called up before their next entry.

All plant personnel wore 0-200 mR pocket dosimeters. Everyone entering the drywell also wore a 0-1 R dosimeter during entry. Additionally, all workers recorded their total daily dose as a double check. The computer flagged the names of people whose TLDs and pocket dosimeters were not in agreement, and these exposures were further evaluated. The LiF TLDs (beta-gamma and gamma sensitive) were changed each month. A TLD reader and operator were on site to process TLDs as quickly as possible.

Special dosimeters (high-range dosimeters, alarming dosimeters, etc., not usually worn by personnel) were issued and controlled at the access point.

6.2.4 Peach Bottom Unit 2

An IBM-PC/XT computer was used as an aid in tracking doses at Peach Bottom. The computer kept a running total of all exposures for personnel, job code, RWP, and craft as well as man-rem estimates for the job. Self-reading pocket dosimeters (500 and 1000 mR) were used to estimate exposure for RWP purposes. Harshaw TLD dosimeters that are easily read out were used for daily dose tracking records on the computer. An Eberline TLD badge was used for permanent exposure records. The high-range pocket dosimeters (1000 mR pencils), Eberline finger rings, and alarming digital dosimeters supplemented dosimetry for individuals entering high radiation areas such as near recirculatory pumps and discharge valves.

Administrative dose limits at Peach Bottom were 300 mR/day, 2500 mR/quarter, and 4500 mR/year.

6.2.5 Vermont Yankee

Administrative limits for the Vermont Yankee plant are 600 mR/week and 1000 mR/quarter for employees who do not have NRC Form 4 paperwork in order, or 2000 mR/quarter with an up-to-date Form 4.

Vermont Yankee badges their radiation workers with Harshaw LiF dosimeters for the official dose measurements and 0-500 mR self-reading pocket dosimeters for the daily records. During the replacement outage, personnel entering the drywell were also issued a high-range 0-1000 mR self-reading dosimeter. Extremity badges were used at the discretion of the HPs, particularly if 25% of the quarterly limit was likely to be incurred by the extremities. Teledose dosimeters were also used in the drywell. These dosimeters were usually placed on the person most likely to receive the most exposure. The TLDs were supplied and read out by the Yankee Atomic Environmental Laboratory. Employees approaching 80% of the administrative limit as calculated from pocket dosimeters had their TLDs read out at the site to determine their official dose.

6.2.6 Browns Ferry Unit 1

The administrative exposure limits for TVA employees were 3 rem/quarter and 4 rem/year. Based on the contract with M-K, limits for contracted workers were 3 rem/quarter and 5 rem/year for exposure incurred at TVA plants.

Dose tracking during the pipe replacement would be conducted similar to routine personnel monitoring except that tracking of individual tasks was planned so that man-rem by task based on self-reading dosimeter values could be updated every 24 hours. Personnel would wear panasonic TLDs, multiple badging when necessary, self-reading dosimeters, and alarming dosimeters during the project. Area monitors would also be used to help monitor conditions in the drywell. Official doses would be based on results of the TLDs, which would be processed monthly or as necessary.

6.3 RADIATION DOSES TO WORKERS

In keeping with the ALARA philosophy, all six plants planned ways to minimize the radiation dose received by their employees during pipe replacement and associated tasks. Presented here are summaries of the ALARA plans, the estimates of work time and dose for tasks, the actual time and measured dose, and the estimated man-rem saved by ALARA practices.

6.3.1 Nine Mile Point Unit 1

Minimizing personnel exposure to ALARA was considered to be a major priority for the recirculation system repair/replacement efforts at NMP-1. The control of exposures was accomplished by first estimating the exposures required to perform individual tasks, then applying effective dose reduction techniques (considering both technical and cost restrictions), and finally measuring the actual personnel exposures to determine the man-rem savings. Table 6.1 gives the collective occupational man-rem doses received by staff at NMP-1 for the years 1970-1981. As a comparison, 1465 man-rem were received by NMPC personnel and its contractors during the 10-month recirculation system outage during 1982 and 1983.

The man-rem dose estimates for the repair/replacement of the entire recirculation system at NMP-1 were calculated by NMPC personnel. These estimates, for the entire project and individual tasks, were continuously updated as new information became available. The exposure estimates were obtained by estimating the man-hours required to perform a job in a radiation field and multiplying by measured or "calculated" exposure rates at the work location. (Exposure rates measured during previous outages or drywell entries were corrected for increased operating history, low water level in the core, etc., to yield the "calculated" exposure rates.) Estimates of the radiation

TABLE 6.1. Annual Collective Occupational Exposure at NMP-1

Year	Occupational Exposure man-rem
1970	44
1971	195
1972	285
1973	567
1974	824
1975	681
1976	428
1977	1383
1978	314
1979	1497
1980	591
1981	1592

exposure were made to establish workforce needs in crucial areas of high radiation levels where skilled personnel must work. The estimates also assisted management in determining areas where temporary shielding could be used.

Two man-rem estimates for the entire repair/replacement efforts were performed by NMPC and submitted to the NRC on June 1, 1982 (Lempges 1982b). The first estimate did not consider the chemical decontamination of the recirculation system, whereas the second estimate took this into account. A total of 5,648 man-rem was estimated without decontamination; and 2,900 man-rem was estimated with decontamination for an estimated savings of 2,742 man-rem (49%) savings over the first estimate. Separate estimates were given for the suction and discharge sides of the recirculation system, and these were further broken down into four major tasks and several subtasks. A summary of this data is given in Tables 6.2 and 6.3. In addition to decontamination, new exposure rate measurement data and the effects of leaving water in the control rod guide tubes contributed to this savings.

Throughout the duration of the outage, new dose estimates were calculated routinely. On April 8, 1983, near the completion of the outage, an estimate was submitted to the NRC of the doses for the total outage. It was generated from the actual personnel doses received (dosimetry data) as of that time and an estimate of doses required to finish the outage. This data is summarized in Table 6.4, and a comparison is made to the June 1, 1982 estimate (decontamination taken into account). The new estimate is about 50% of the 1982 estimate.

This can be attributed to the use of effective dose reduction techniques and partly to more accurate dosimetry measurements (1982 estimates were conservative). Table 6.4 shows that the estimated dose (April 8) and the actual dose received (July 26) were in fairly good agreement (107%). Between the period of April 8 to July 26, 1983, NMPC continued to generate exposure estimates (Mangan 1983). However, these were really generated as an administrative requirement, since the estimated and actual values were almost identical and no new information was made available. Although the difference between the estimates and actual totals was insignificant, there were some disparities between individual task estimates and actual task man-rem and/or man-hour values as shown in Tables 6.2 through 6.5.

Table 6.5 gives a summary of the estimates and the actual man-hours worked for the recirculation system repair/replacement efforts. These data were included with the man-rem exposure reports given to the NRC.

The exposure reports sent to the NRC by NMP-1 provided a detailed breakdown of the tasks to be performed, the measured or estimated exposure rates in the areas where work was performed, the projected man-hours for each task, and the estimated man-rem for each task. Members of the ALARA committee reviewed the man-hour and man-rem estimates and the actual exposure data before they were sent to the NRC. This review was also performed to determine if further dose reduction techniques could be employed beneficially. The site refined the estimates as the work progressed to incorporate dose reductions based on experiences gained during the first loop repair/replacement. This experience was then applied to subsequent work on the remaining four loop modifications, which resulted in further dose reductions.

The actual man-rem doses received by workers during this outage and reported to the NRC were based on actual dose measurements using dosimetry devices issued to personnel. NMPC used three dosimetry systems (TLDs, film badges, and pencil dosimeters) in conjunction for personnel dose measurements. The ratio of film badge to recorded pencil dosimeter measurements were found to be 0.82 during the repair/replacement efforts. Total accumulated pencil dosimeter man-rem data for the outage were then corrected by applying the ratio of 0.82. This multiple backup dosimetry system provided good measurement data. In situations or locations where dosimetry data were hard to obtain or were uncertain, dose rates as measured by survey instruments and actual man-hours worked were used to calculate personnel exposure. The drywell coordinators from NNIC were responsible for keeping track of actual man-hours worked at each job location. This information was logged on each exit from the drywell. The actual man-rem doses received and man-hours worked for the outage are summarized in Tables 6.4 and 6.5, respectively.

A significant amount of dose was saved during this outage. The savings can be classified into two groups. The first category of exposure savings was due to the use of a chemical decontamination of the recirculation system piping before the piping removal efforts began. The remainder of the savings was achieved as a result of the dose reduction techniques employed during the actual removal and replacement of the piping. Such techniques include the use of 1) temporary shielding, 2) planning, 3) training, 4) mockups, 5) remote equipment/monitoring, and 6) contamination control.

TABLE 6.2. NMP-1 Man-Rem Dose Estimates - Discharge Side

Type of Work	Estimates Before June 1, 1982*	Estimates of June 1, 1982**	Savings
A. Preliminary Drywell Work			
1. Shielding installation	160.0	160.0	0.0
2. Pipe preparation	46.0	23.0	23.0
Subtotal	206.0	183.0	23.0
B. Removal of Existing Safe-Ends and Elbows			
1. Cutting setup	841.8	265.8	576.0
2. Pipe cutting operations	235.0	120.0	115.0
3. Shielding installation	600.0	420.0	180.0
4. Welding preparation	130.0	102.2	27.8
5. Equipment/pipe removal	171.0	88.0	83.0
Subtotal	1,977.8	996.0	981.8
C. Installation of New Safe-Ends and Elbows			
1. Welding setup	570.4	357.3	213.1
2. Pipe welding operations	872.5	539.5	333.0
3. Weld inspection	137.5	87.5	50.0
Subtotal	1,580.4	984.3	596.1
D. Postrepair Drywell Work			
1. Shielding removal	168.4	104.8	63.6
2. Drywell restoration***	48.8	43.4	5.4
Subtotal	217.2	148.2	69.0
TOTAL	3,981.4	2,311.5	1,669.9

* Decontamination of recirculation system not considered.
** Decontamination of recirculation system taken into account.
*** Discharge and suction sides of recirculation system.

TABLE 6.3. NMP-1 Man-Rem Dose Estimates - Suction Side

Type of Work	Estimates Prior to June 1, 1982*	Estimates of June 1, 1982**	Savings
A. Preliminary Drywell Work			
1. Work area setup***	66.6	45.4	21.2
2. Shielding installation	244.8	44.8	200.0
3. Pipe preparation	158.4	63.8	94.6
Subtotal	469.8	154.0	315.8
B. Removal of Existing Safe-Ends and Elbows			
1. Cutting setup	170.9	110.9	60.0
2. Pipe cutting operations	41.0	35.0	6.0
3. Shielding installation	100.0	12.2	87.8
4. Welding preparation	73.5	45.0	28.5
5. Equipment/pipe removal	26.0	16.0	10.0
Subtotal	411.4	219.1	192.3
C. Installation of New Safe-Ends and Elbows			
1. Welding setup	237.8	55.5	182.3
2. Pipe welding operations	381.0	115.5	265.5
3. Weld inspection	64.5	39.0	25.5
4. Equipment removal	4.0	1.5	2.5
Subtotal	687.3	211.5	475.8
D. Postrepair Drywell Work			
1. Shielding removal	99.2	9.6	89.6
2. Miscellaneous	0.5	0.5	0.0
Subtotal	99.7	10.1	89.6
TOTAL	1,668.2	594.7	1,073.5

* Decontamination of recirculation system not considered.
** Decontamination of recirculation system taken into account.
*** Discharge and suction sides of recirculation system.

TABLE 6.4. Comparison of Man-Rem Dose Estimates and Actual Exposures Received at NMP-1

Type of Work	Estimate of June 1, 1982*	Estimate of April 8, 1983	Actual July 26, 1983
A. Preliminary Drywell Work	337.0	571.2	514.8
B. Removal of Recirculation Loop - Suction Side	219.1	174.8	175.1
C. Removal of Recirculation Loop - Discharge Side	996.0	266.6	263.6
D. Installation of New Safe-Ends, Elbows, and Piping - Suction Side	211.5	228.5	215.4
E. Installation of New Safe-Ends, Elbows, and Piping - Discharge Side	984.3	165.1	159.5
F. Postrepair Drywell Work	158.3	63.7	53.0
G. Indirect Recirculation System Work	-	90.9	82.7
TOTAL	2,906.2	1,560.8	1,464.1

* Decontamination of recirculation system taken into account.

Tables 6.2 and 6.3 show that there was a total "estimated" decontamination dose savings of approximately 2,743 man-rem (49%). The use of temporary shielding can either be a net benefit or a disadvantage. At NMP-1, the shielding was a benefit. In contrast, it could be a disadvantage if the installation and removal required more personnel exposure than the dose saved due from the reduction of dose rates at the work locations. In addition to the reduction of exposure by the use of temporary shielding, NMP-1 found that chemical decontamination also reduced the estimated exposure needed to place and remove the shielding. A total of approximately 621 man-rem (45%) was saved (summarized from Tables 6.2 and 6.3) during the shielding operations because of the use of chemical decontamination.

The use of dose reduction techniques during the repair/replacement efforts at NMP-1 produced a net man-rem and man-hour savings. Comparing the actual values to the estimates (Tables 6.4 and 6.5) gives an indication of the effectiveness of the work practices used.

TABLE 6.5. Comparison of Man-Hour Estimates and Actual Man-Hours Worked at NMP-1

Type of Work	Estimate of June 1, 1982*	Actual July 26, 1983
A. Preliminary Drywell Work	3,400	57,753
B. Removal of Recirculation Loop - Suction Side	1,100	4,010
C. Removal of Recirculation Loop - Discharge Side	1,714	5,348
D. Installation of New Safe-Ends, Elbows, and Piping - Suction Side	2,108	9,756
E. Installation of New Safe-Ends, Elbows, and Piping - Discharge Side	3,608	8,043
F. Postrepair Drywell Work	1,596	3,983
G. Indirect Recirculation System Work	-	6,246
TOTAL	13,526	95,089

* Decontamination of recirculation system taken into account.

In comparing the dose values from Table 6.4, it is necessary to consider the estimate of June 1, 1982. The April 8, 1983, estimate, submitted by NMPC to the NRC, as discussed earlier, is based on actual dosimetry data (near completion of the outage). Therefore, it should not be used as the estimate to compare to the actual values for determining dose savings or the effectiveness of dose reduction techniques. The net exposure savings from the data given in Table 6.4 was 1,442.2 man-rem, or approximately a 50% savings over the June 1, 1982, estimate.

The comparison of the man-hour estimates and actual values from Table 6.5 shows that there was a net increase (approximately 600%) in the total required man-hours. In fact, the final level of 95,089 man-hours was a substantial increase (approximately 600%) over the estimated value. This would not seem consistent with an overall 50% reduction in personnel exposure. However, in reporting to the NRC on July 26, 1983, NMPC included the actual man-hours received for all personnel entering and leaving the NMP-1 drywell. A significant portion of these entries involved personnel supporting the major repair/replacement tasks. In most cases, this involved low-dose-rate areas, and little exposure was received even though many man-hours were required to perform the work. The original estimates (June 1, 1982) given to the NRC by

NMPC did not consider the hours required for these support functions. These included activities such as inspections, fire watches, photography, and security.

A summary of the total man-rem and man-hour savings is given in Table 6.6. This data is based upon the data in Tables 6.2 through 6.5 plus reports given to the NRC. In addition to the overview data given in these tables, there were several specific cases that should be mentioned that are not summarized above.

There was a significant reduction in the actual man-hours worked for the removal of shielding and scaffolding from the drywell. Easily accessible storage areas were established in the drywell near the equipment hatch and removal of scaffolding was restricted to specific times so that the total man-hours were kept to a minimum. Only 115 of an estimated 775 man-hours were used for these tasks with a savings of 12 man-rem.

The estimates for general inspections were based on the average daily man-hours needed throughout the outage. However, as the outage drew to a close, less inspection was required. Drywell inspection requirements resulted in an overall reduction of 21 man-rem and 326 man-hours over the estimated values. The actual recirculation piping insulation reinstallation exceeded its estimate by over 1,222 man-hours and 8 man-rem.

The site policy at NMP-1 required that a guard be posted on duty at the drywell entrance during the outage. About 18,400 man-hours were used for this task, using approximately 32 man-rem. The security required for this outage exceeded the estimated man-hours by 380, with no appreciable increase in man-rem.

6.3.2 Monticello

During the pipe replacement outage at Monticello, the highest doses received were between 2600 and 2700 mrem. The personnel who accumulated the greatest dose were the welders, maintenance personnel, and pipe fitters. The original dose and time estimates and the final totals, broken down by task, are shown in Table 6.7. A considerable amount of the estimated dose was not

TABLE 6.6. Summary of Total Man-Rem and Man-Hour Savings at NMP-1

	Savings*			
	Exposure (man-rem)		Man-Hours	
Type of Work	Total	%	Total	%
Chemical Decontamination	2,743	49	-	-
Repair/Replacement Efforts	1,442	50	-81,563	-600
Total	4,185	74	-81,563	-600

* A positive value indicates a net savings and a negative value indicates a net increase.

TABLE 6.7. Final Dose and Initial Dose Estimates for the Recirculation Piping Replacement Project at Monticello

Task Description	Original Estimate February 5, 1985 man-hr	Original Estimate February 5, 1985 man-rem	Final Total January 17, 1985 man-hr	Final Total January 17, 1985 man-rem
A. Piping System Removal				
1. Drywell preparation	11,370	633.2	7,659	234.6
2. Loop A system removal	2,078	83.1	1,164	18.2
3. Loop B system removal	1,960	68.9	1,044	20.7
4. RHR Piping Removal	-	-	1,214	72.3
B. Piping System Installation				
1. Loop A system installation	6,299	145.3	6,911	68.1
2. Loop B system installation	4,814	110.0	6,379	62.9
3. Drywell restoration	6,880	142.7	13,813	104.5
4. RHR piping installation	-	-	3,460	42.1
C. Safe-End Replacement				
1. Loop A discharge S/E replacement	3,790	127.5	3,410	199.1
2. Loop B discharge S/E replacement	4,489	184.4	3,410	199.1
3. Loop A suction S/E replacement	483	22.7	757	13.4
4. Loop B suction S/E replacement	520	23.3	757	13.4
5. Jet Pump inst S/E replacement	0	0	1,507	25.0
6. SBLC S/E replacement	0	0	332	2.6
D. Site Support				
1. Materials and equipment handling	4,300	4.3	369	10.2
2. Q/A inspection and radiography	63	4.9	3,614	53.0
3. General supervision	100	5.9	1,087	16.3
4. Security	5,232	5.2	0	1.8
5. General laborer support work	-	-	10,628	93.4
6. Waste handling	-	-	196	16.6
E. Separate Contracts				
1. Health physics support	4,600	46.0	6,643	82.3
2. Induction heat stress improvement	-	-	5,456	62.5
F. Auxiliary Workscopes				
1. Small-bore piping replacement	4,530	97.3	7,296	68.5
2. Hanger & restraint work	4,370	182.5	6,529	51.6
3. Refuel floor work	2,456	34.5	1,272	50.9
	68,334	1,921.7	94,919	1,583.1

used during the cutting and removal of the pipes, which was accomplished in 9 days and took approximately 72% of the estimated time and 44% of the estimated dose. This included the time and dose required to remove the RHR piping, which was not in the original estimate.

The piping installation required substantially more time than originally estimated, although less dose was required. The replacement of the safe-ends took longer and required more dose than was originally estimated. This was attributed to the difficulties encountered during installation of the safe-ends and alignment of the thermal sleeves.

Tasks such as the RHR piping decontamination, removal and replacement, and the replacement of the jet pump instrument safe-end and the SBLC safe-end had not been expected early in the outage and so no estimate of dose or time were provided. The scope of some tasks such as supervision and radiography had originally been underestimated.

Specific activities that were credited with reducing dose include decontamination, hydrolasing the return nozzles (especially the gap between the thermal sleeves and the safe ends), removal and replacement of the RHR piping, removal of the peripheral control rods and insertion of the remainder of the control rods about a third of the way in, and use of remote TV monitors, remote dosimeters and communication headsets (which eliminated some of the dose which would otherwise have been used on supervision and radiation monitoring).

It was estimated that the chemical decontamination saved 827 man-rem, while the removal of the RHR piping resulted in an estimated dose savings of 900 man-rem. Although the man-hours required for the pipe replacement were higher than estimated, the actual doses received during the entire pipe replacement were lower than estimated. The original estimate had been 1,922 man-rem. The actual dose from the recirculation pipe replacement was 1,583 man-rem (despite the added project scope).

6.3.3 Cooper

The initial man-hour and man-rem estimates for the pipe replacement program at Cooper was broken down into 144 separate tasks. The man-hour estimates totaled 67,043 for the overall program. The estimates were based on projected working area dose rates and workforce levels for the individual tasks.

As the pipe replacement program at Cooper progressed, a need to revise the initial estimates became apparent. The two major reasons for the first revision were underestimation of man-hour requirements and overestimation of working area dose rates. Man-hour requirements were underestimated because of contractor underestimates, unanticipated problems, and cramped conditions interfering with work progress. Working area dose rates were overestimated because of the significantly improved decontamination factor achieved and the underestimation of the effectiveness of temporary shielding. Fortunately the increase in man-hour requirements and the decrease in working area dose rates counteracted each other and resulted in only a 1.4% increase in the overall man-rem estimate. The revised estimate predicted 96,295 man-hours and 1,435.7 man-rem for the overall pipe replacement program.

A second revision to the estimates was required when the work scope was expanded and the outage extended. Most of the additional man-hours in the drywell and subsequent additional man-rem are from tasks that are expected to reduce the potential for and aid in the detection of future IGSCC and subsequently reduce future radiation doses. These tasks were corrosion resistant cladding (CRC), induction heating stress improvement (IHSI), and weld crown reduction (WCR) allowing automated in-service inspection (ISI) examination. The remainder of the increased drywell work involved concerns regarding accurate pipe alignment, upgrades and changes in plant design, and various other maintenance tasks. This second revision estimated a total of

123,810 man-hours and 1,750.6 man-rem as shown in Table 6.8. The actual completed work required a total of almost 187,000 man-hours and 1,636 man-rem based on the official TLD readouts.

TABLE 6.8. Man-Hour and Man-Rem Estimates for the Pipe Replacement Program at Cooper

	Man-hours	Man-rem
Initial Estimate (8/15/84)	67,043	1,415.7
Revision 1 Estimate (4/3/85)	96,295	1,435.7
Revision 2 Estimate (4/29/85)	123,810	1,750.6
Actual (12/27/85)	186,999	1,636.0

Table 6.9 is a summary of the 144 separate tasks included in the initial man-rem estimate for the IGSCC project at Cooper Nuclear Station as approved by NPPD.

TABLE 6.9. Summary of Man-Rem Estimates by Task Type for the Pipe Replacement Program at Cooper

	Initial Estimates	
Task	Man-Hours	Man-Rem
Supervision	14,200	176.3
Decontamination	1,159	79.4
In-Core Work	1,649	21.5
Pipe Dimensioning	91	12.3
CNS Support	5,340	66.3
Equipment Transport	863	9.0
Small-Bore Piping	2,034	31.7
Shielding	1,926	104.2
Housekeeping and Area Decontamination	4,980	74.1
Lighting and Power	350	10.1
Insulation	1,050	30.3
Supports and Protection	5,298	83.4
Ductwork	2,286	31.9
Electrical	4,132	46.9
Rigging	872	24.9
Scaffolding	452	12.7
Pipe Packaging	89	1.3
Pipe Cut-Out	4,234	157.7
Auxiliary System	2,968	167.1
Weld Preparation	854	16.6
Pipe Installation	8,254	234.8
Tool Decontamination	480	3.8
Miscellaneous	482	16.4
Preoperation Testing	3,000	3.0
TOTAL	67,043	1,415.7

The final values based on TLD readings for the recirculation pipe replacement are presented in Table 5.10. These numbers do not include the following repair or replacement tasks: refueling floor work, core spray, RWCU, jet pump at SLC nozzle, or IHSI. The total man-hours including these tasks amounted to 186,999 man-hr and 1,636 man-rem. As is the case with most sites, the initial man-hour estimate did not account for personnel frequently entering the drywell with materials or equipment who were not likely to incur dose. Therefore the man-hour estimates were too low while the dose did not change much from estimates.

6.3.4 Peach Bottom Unit 2

PECo has not completed a task-by-task analysis of the pipe replacement doses for all 370 individual tasks but has summarized doses for the entire project into the five major categories shown in Table 6.11.

TABLE 6.10. Totals for Cooper Job Categories

Job Performed	Actual Totals Man-Hours	Actual Totals Man-Rem
Chemical Decontamination	864.7	12.070
Mobilization	11,166.31	294.921
Pipe Removal	17,910.77	186.049
Pipe Replacement	63,016.79	441.571
Remove & Replace N1	3,454.82	40.959
Remove & Replace N2	14,310.87	192.251
Restoration	34,229.56	157.793
HP Support	6,382.53	40.802
Miscellaneous Support	5,490.95	45.813
ISI Inspection	919.69	9.495
TOTAL	157,746.93	1,421.724

TABLE 6.11. Major Task Categories with Estimated and Actual Man-Rem Doses at Peach Bottom

Task Category	Estimated Man-Rem	Actual Man-Hour	Actual Man-Rem
Predecontamination	445	10,446	360
Pipe Decontamination	38	2,218	38
Pipe Removal	516	12,729	304
Pipe Replacement	887	70,580	1,060
Drywell Restoration	59	12,909	134
TOTAL	1,945	121,180	1,895

The dose estimates were based on replacement of the recirculation and RHR piping, head spray, and the RWCU penetration. The dose estimates covered only the inspections of the recirculation safe-ends and the jet pump instrument safe-ends and seals. The inspection revealed that replacement was necessary. The dose incurred from replacing these components was substantial and is included in the actual man-rem values. Replacing the seals and safe-ends required 8,582 man-hours at a dose of 211 man-rem.

A second scope change causing an increase in effort and dose was in decontaminating the bowl of a recirculation pump, a process separate from the pipe decontamination. Hot spots of 700-800 R/hr in the recirculating pump dictated the need for a dose reduction strategy. Hydrolancing was not effective in removing the high-exposure fields, so glass-bead blasting was successfully used. This reduced exposure rates an average of 600 mR/hr with hot spots to about 2 R/hr. The pipe disassembly and decontamination required 3,716 man-hours and incurred a dose of 93 man-rem.

The effectiveness of the actions taken to reduce exposure was analyzed, and most of those actions were judged to have saved personnel exposure (see Table 6.12). Chemical decontamination, the largest dose savings measure, was

TABLE 6.12. Effectiveness of ALARA Mitigating Measures at Peach Bottom Unit 2

ALARA Technique	Dose Savings Man-Rem (estimated)	Dose Savings Man-Rem (actual)
Control Rod Position	175 - 200	175 - 200
Hydrolance Crevice in Discharge Nozzles	30	100
In-Vessel Shielding of Suction Nozzles	175 - 250	175 - 250
Water in Vessel Using Jet Pump, Slip Joint, and Nozzle Plugs	50 - 55	0
Chemical Decontamination	3200 - 3600	1200 - 1250
Suction and Discharge Nozzle Safe-End Shielding	180 - 210	120
Hot Spot Shielding in the Drywell	140	150 - 170
Mockup Training	50 - 200	50 - 200
Valve Weld Prep and Rebuild Location	68	-30

less effective than expected because of the difficulties arising from decontaminating the pumps and valves. Jet pump nozzle plugs and slip joint plugs did not seal as well as expected, and it was determined that water in the vessel did not provide as much shielding as anticipated. Performing valve weld preparation in the drywell was expected to incur less dose than removing the valves and completing the work in low-background areas. However, because of the high exposure rates at the drywell site, the dose for maintaining and weld prepping these valves in place was higher than the estimates for removing them to a low-background area.

6.3.5 Vermont Yankee

An estimate of the dose likely to be incurred as a result of the pipe replacement project was calculated by the recirculation project team using historical data of Vermont Yankee dose rates, projections of decontamination effectiveness, and the results of other utilities' replacement outages. Refinement and revision of the estimates will proceed if ongoing analysis reflects significant variation in actual doses over estimates. Additionally, revised dose estimates will be made based on actual rather than historical dose rates.

Baseline levels for dose rate projections were based on the 1984 refueling outage during which extensive gamma surveys were conducted. Measurements were made on contact and at 18 inches from primary piping and components, and general area exposure rates were taken. Survey results were increased by a projected average buildup factor. To facilitate dose projections, the drywell was divided into zones. Major, predictable changes in dose rates were estimated for each zone and for each major change of condition.

The first of the expected dose rate changes resulted from the in-situ chemical decontamination of the primary reactor recirculation piping. General area decontamination factors (DF) were used in this calculation. Previous decontamination has resulted in DFs between 2 and 15. A DF of 5 was chosen as a reasonable but conservative figure for most drywell locations. A DF of 2 was applied to the nozzle regions, reflecting the difficulties in dose reduction techniques in these areas.

Other estimates considered changes to dose rates from such activities as pipe removal and installation of shielding.

The doses originally estimated to be incurred during the pipe replacement outage as calculated by Morrison-Knudsen and reviewed by Vermont Yankee are listed in Table 6.13.

A December 1985 update of the projected doses estimates a total dose of 1,647 man-rem. This includes several tasks (bottom head drain, core spray nozzle, IHSI) not specifically related to the pipe cutting and removal. Task categories from the original estimate along with the new estimate are presented in Table 14.

TABLE 6.13. Tasks and Estimated Doses for Pipe Replacement at Vermont Yankee

Task Description	Original Dose Estimate (man-rem)
A. Replacement Preparations	
Whip restraint removal, sides A & B	22.3
Piping insulation removal	34.2
Drywell shielding	14.6
Establish drywell access	5.1
Interference removal	59.0
Drywell decontamination and cleanup (ongoing)	109.6
Chemical decontamination of pipe	7.2
B. Piping System Removal	
Remove suction pipe side A	37.1
Remove suction pipe side B	29.9
Remove discharge pipe side A	54.6
Remove discharge pipe side B	42.3
Remove spring hangers RCR system	7.2
Machine cut thermal sleeves and safe-ends	60.5
Remove RHR 30	11.8
Remove RHR 31	5.6
Remove RHR 32	8.8
Machine prep A - suction nozzle (N1)	14.4
Machine prep B - suction nozzle (N1)	14.2
Machine prep recirculation pump P-1B-1A	2.7
Machine prep recirculation pump P-1B-1B	1.7
Machine prep valves	26.0
Machine prep existing RHR-30	4.8
Machine prep existing RHR-31	2.6
C. New Pipe Installation	
Install recirculation suction pipe A	66.3
Install recirculation suction pipe B	56.2
Install discharge pipe A	58.0
Install discharge pipe B	48.3
Install spring hanger RCR system	5.2
Install sleeve and safe-ends	82.9
Install RHR-30	12.5
Install RHR-31	9.5
Install RHR-32	14.9
Install spring HGRS RHR system	19.1

Table 6.13 (contd)

Task Description	Original Dose Estimate (man-rem)
D. Postinstallation Tasks	
Install piping insulation	27.7
Install whip restraints - A	7.7
Install whip restraints - B	7.7
Clearance inspection	0.8
IHSI	48.9
System hydrostatic testing	14.4
Pipe disposal	49.6
E. Site Support	
General inspection/supervision	127.6
Elec/communication maintenance	28.8
HP inspection/control point	48.0
Drywell security	7.2
Fire watch/inspection	74.4
TOTAL	1,391.9

TABLE 6.14. Original and Revised Dose Estimates by Job Category

Task Category	Original Dose Estimate man-rem	Revised Dose Estimate man-rem
Replacement Preparations	252	381
Piping System Removal	324	346
New Pipe Installation	373	567
Postinsulation Tasks	157	136
Site Support	286	217
TOTAL	1,392	1,647

The most critical and dose-intensive activities involved in the project include the cutting, machining, and welding of the thermal sleeves to the jet pump risers. Other tasks expected to incur significant doses include drywell decontamination and cleanup, cutting and removing thermal sleeves and safe ends, installing the new suction and discharge pipe, fire watch inspection, and supervision and general inspection. Although this last item incurs a high dose, it has been minimized through using TV cameras inside the drywell with monitors outside the drywell; this allows supervisory personnel and health physics staff to observe progress outside the high-dose-rate areas.

6.3.6 Browns Ferry Unit 1

The total dose estimate for the pipe replacement at Browns Ferry was 1780 man-rem. Previous area exposure rates and doses from previous outages were taken into consideration in the dose estimates as were the expected exposure reduction rates resulting from the chemical decontamination. The effectiveness of shielding and the exposure rate gradients from actual time spent in various dose rate areas were estimated. The man-hour estimates were multiplied by the dose rate estimates to calculate anticipated man-rem.

Table 6.15 lists the estimated man-hours by task and the estimated doses with and without pipe decontamination. The estimated dose savings from the planned chemical decontamination was greater than 1100 man-rem and easily justifies the estimated 10 rem dose for the decontamination process. The total estimated dose was 1779.8 man-rem.

The most dose-intensive tasks in the replacement were expected to be installation of the recirculation piping and core spray system. Welding and ISI prepping of the joints between the piping and reactor vessel would have contributed heavily to these task doses. General drywell support, which includes crafts support and supervision, was also expected to incur large doses.

TABLE 6.15. Estimated Time and Doses for Pipe Replacement at Brown Ferry

Task Description	Man-Hours	Man-Rem Without Decontamination	Man-Rem With Decontamination
1. Loop A Recirc System Removal	2,571	233.8	100.3
2. Loop B Recirc System Removal	2,529	234.9	100.5
3. Loop A Recirc System Installation	9,055	243.4	243.4
4. Loop B Recirc System Installation	8,883	232.5	232.5
5. RHR Piping Removal	1,049	47.6	15.8
6. RHR Piping Installation	5,661	41.4	41.4
7. RHR Head Spray Removal (not to be replaced)	487	2.7	2.7
8. A Core Spray Removal	589	31.9	31.9
9. B Core Spray Removal	589	31.9	31.9
10. A Core Spray Installation	2,285	105.9	105.9
11. B Core Spray Installation	2,285	105.9	105.9
12. RWCU Piping Removal	154	28.0	9.2
13. RWCU Piping Installation	1,250	8.5	8.5
14. Support Tasks			
General area shielding and decontamination work	1,004	63.7	23.6
General interference removal	2,019	242.3	80.8
General drywell support	22,688	673.3	295.0
QC/QA inspection and NDE work	2,395	16.7	16.7
Drywell restoration	5,953	41.7	4[illegible].7
Engineering field support	2,400	107.2	43.2
IHSI	582	14.7	14.7
Electric	7,070	347.8	184.2
Health physics		60.0	40.0
Chemical decontamination		*	10.0
TOTAL	81,718	2,915.8	1,779.8

7.0 RADIOACTIVE WASTE MANAGEMENT

Two of the significant sources of radioactive waste generated during the pipe replacement outages are the radioactive resins from the decontamination process and the old recirculation piping cut out of the system. The remaining waste includes routine items such as contaminated clothing. In most of the six cases reviewed here, vendors removed the resins and pipe. The rest of the waste was handled routinely at the plants.

7.1 NINE MILE POINT UNIT 1

To reduce occupational exposures in the drywell of NMP-1 during the recirculation system repair/replacement efforts, a chemical decontamination was performed with London Nuclear's Can-decon® process before the repair/replacement activities. The chemical piping decontamination resulted in the generation of radioactive resin from each of the decontamination system ion exchange columns. After the decontamination, a resin slurry was formed that allowed discharge of the resin into resin liners and shielded casks for processing. The resin waste was solidified, stored, and shipped for disposal pursuant to the criteria of 10 CFR 61. The resin waste was stored onsite temporarily in shielded areas of the waste building.

The old piping, elbows, and safe-ends removed from the recirculation system were packaged and stored for shipment and disposal. A special area was set up for temporary storage of that material.

7.2 MONTICELLO

The radwaste generated during the decontamination was removed by Quadrex. Quadrex in turn subcontracted to Chem Nuclear Systems Inc. to solidify the resins. Chem Nuclear also extracted and disposed of the solid radwaste. The pipe was sent to Battelle, Pacific Northwest Laboratories for examination.

The insulation that was removed during the outage was reused instead of being disposed of as waste. The insulation had just been replaced during the previous outage.

7.3 COOPER

Nebraska Public Power District built a facility to process and decontaminate waste during the pipe replacement. This facility will be converted to a low-level waste storage facility. Chem Nuclear extracted and disposed of the solid waste from the decontamination operation. The rest of the radwaste generated during the pipe replacement was compacted into drums or crates and sent to Beatty, Nevada. U.S. Ecology was responsible for transporting the waste. NPPD was responsible for the disposal and removal of radioactive waste from the site. Quadrex accepted the old pipe.

7.4 PEACH BOTTOM UNIT 2

Radioactive resin waste generated during the Can-decon® pipe decontamination was solidified using a cement solidification process and disposed of in accordance with 10 CFR 61. Other generated radioactive wastes were wrapped, tagged, and transferred to a radwaste handling facility adjacent to the Unit 2 reactor building before being shipped offsite for disposal. Chem Nuclear disposed of decontamination resins at the Barnwell, South Carolina disposal site. The pipe was shipped to Quadrex and Oak Ridge National Laboratory for further decontamination and recovery as scrap metal.

7.5 VERMONT YANKEE

Resins from the Citrox decontamination process were shipped to the Barnwell, South Carolina disposal site by Chem Nuclear. The pipe ends were tack welded with stainless steel covers and loaded into shipping boxes for disposal. The shipping boxes were lined with shielding to drop the dose rates to below shippable limits. Some of the pipe was shipped to the Hanford, Washington, disposal site. Brookhaven National Laboratory and Battelle, Pacific Northwest Laboratories received some of the pipe for examination and analysis. Fitzpatrick Nuclear Plant received the weld overlays for further study.

7.6 BROWNS FERRY UNIT 1

Normal radwaste resulting from the pipe replacement outage would be sent to the Barnwell, South Carolina disposal site. Resins from the pipe decontamination were London Nuclear's responsibility and would ultimately be shipped to Barnwell. The removed pipe was to be sent to Battelle, Pacific Northwest Laboratories for examination.

8.0 ACCUMULATED EXPERIENCE FOR FUTURE APPLICATION

In each case where pipe replacement has been undertaken, the utility staffs emphasized the importance of extensive and thorough planning. With sufficient planning and lead time, problems may be anticipated and avoided; equipment may be designed to simplify difficult tasks; and specific dose reduction strategies may be developed.

As the first site to perform major pipe replacement, NMP-1 passed much of its experience to other utilities to aid their replacements. General practices by NMP-1 and at most other sites that provided significant minimizing of exposure during the replacement outage included:

- use of chemical decontamination before the repair/replacement efforts
- filling of the control rod guide tubes with water for shielding
- use of temporary shielding, such as shield curtains, nozzle plugs, and personnel waiting areas
- use of contractors with specialized expertise in health physics pipe repair/replacement, etc.
- use of an multielement external dosimetry system (TLDs, film badges, pencil dosimeters) and specific exposure limits
- use of ALARA procedures and planning
- use of full-sized mockups and appropriate training (maintenance and health physics)
- review of exposure data by supervisors, radiation protection personnel, and the ALARA committee and use of a formalized computer-controlled dose tracking system
- decontamination of tools and equipment, and control of surface/airborne contamination
- use of protective clothing, respiratory protection, and portable ventilation
- use of remotely operated welding and cutting equipment
- use of shielded storage areas for contaminated equipment.

Some of the site-specific lessons learned or planned and other actions the sites found particularly beneficial are discussed by plant. Vermont Yankee, which is still in their replacement outage at this time, and Browns Ferry, which postponed their replacement, are not discussed here. Both utilities have made use of previous experience at earlier replacements to facilitate their respective pipe replacement programs. In spite of extensive planning, unexpected situations do arise and require unique and sometimes ingenious procedures to handle them. Questions on the efficacy of various types of shielding also arise, and where measurements have been made to address such questions, answers are provided.

8.1 NINE MILE POINT UNIT 1

Many lessons were learned by NMPC and its contractors before and during the recirculation system repair/replacement outage at NMP-1. The major items identified are listed below.

- Detailed planning (beginning in late 1978) saved man-rem and man-hours. Three years of preplanning allowed NMP-1 to develop a formal repair/replacement program that received both extensive internal and external peer reviews. The end result of this was a smooth-running repair outage with only minor problems. However, NMP-1 did not benefit from previous recirculation system piping replacements at other BWRs because theirs was the first.

- Planning allowed NMP-1 to identify, design, purchase, and store long-lead-time items such as materials, tools, and equipment. This included safe-ends, piping, elbows, shielding, and welding material. Automatic cutting, pipe prepping, and welding equipment was designed, built, and delivered to NMP-1 well in advance of the outage. The design of some of this equipment was based on the experience gained with such equipment for shipyard work on merchant and naval vessels.

- Plasma-arc cutting equipment was chosen for the recirculation system pipe cutting operations. This equipment allowed a 28-in. pipe to be cut in 8 minutes, as opposed to approximately 8 hours for a conventional mechanical machine cut. Because of the potential of fire in the drywell area from this cutting operation, plastic fire retardent curtains were setup around the equipment in addition to the use of other fire prevention techniques (sucn as portable fire extinguishers). Loss of chips into the pipe system from the cutting operations was minimized by dirt seals and splash pans that caught falling chips.

- Specially designed jigs, templates, and fixtures were used to help position each piping element to within tolerances of 3/32 inch.

- Ultrasonic testing did not indicate the presence of cracks during routine testing in 1981, but cracking was discovered by a series of tests in 1982. New weld crowns were reduced to make ultrasonic testing more sensitive and reliable in the future.

- For day-to-day decisions, a single repair/replacement project manager from NNIC, in contact with his counter part from NMPC, was in charge.

- IGSCC was found equally among shop and field welds at NMP-1. None of the contributing factors to IGSCC (chemistry, stress, sensitized material) were found to be excessive enough to expect the degree of IGSCC that occurred.

- Replacement piping used by NMPC consisted of prebent (Type-316 stainless steel), which is expected to be fully resistant to IGSCC in the lifetime of the NMP-1 plant.

- Remotely controlled television monitoring and communication systems were used. The television system consisted of cameras at each major work location, the drywell entrance, and other locations as needed. Welders observed weld progress from a distance through the use of remote fiber optics trained on the weld puddle.

8.2 MONTICELLO

One of the lessons learned during the Monticello pipe replacement was the need for extensive planning. The safe-end work required especially good controls, planning and mockups. Although planning for the outage started up to a year and a half before the outage, the schedule was still tight. Much of the procedure writing occurred as the outage progressed. This was due in part to the added project scope, limited resources, ongoing problems encountered and the amount of time involved for the selection of contractors.

The removal and replacement of equipment with high dose rates was beneficial in reducing the total amount of dose received during pipe replacement. The replacement of the RHR piping reduced the total dose by an estimated 900 man-rem despite the increase in the amount of work. On the other hand, the facility indicated that some dose was received during the replacement of a RHR suction isolation valve that could have been avoided had a new valve been procured. The decision to keep the valve was based on an analysis of the ALARA benefits versus the cost of replacement.

The water level control in the vessel was not as critical as originally expected. Dose rates changed only slightly when the water level was lowered from the top of the shroud to just below the jet pump diffusers.

8.3 COOPER

The majority of the Cooper replacement was planned out well in advance to avoid delays, increase efficiency, and prevent increased workscopes. There were, however, unanticipated problems. Weld repair was one of the most time and dose consuming. Many welds had to be redone, not because the weld was poor, but because the inspection tests showed evidence of inclusions that were really just crud. This situation emphasized the need to closely examine joints between new and old material, especially cast material, or at pump and valve interfaces. Welds between two stainless-steel pipes or between stainless and Inconel® are difficult enough, but welds between stainless and cast material (especially valves) are extremely difficult. Having to redo these welds because of crud in the joint is very labor and dose intensive. Welds of new pipe to new pipe resulted in a failure rate of 25% (not including N1 or N2

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work). Rework of the welds required about 125 man-rem, a significant fraction of the total dose.

Cooper's decision not to decontaminate the pumps probably created more problems than it solved. They had to fabricate shields for the pumps that would allow them to work on the pipe inlet and outlet and sever the pipes. The ALARA coordinator recommended reviewing the possibility of including the pumps in future decontaminations.

There was good cooperation between contract workers and plant health physicists. However, the type of contract between CNS and the general contractor caused dissension at the supervisory level. The contractor was on a fixed-cost/fixed-time/cost-overrun contract, and his priorities differed from those of the ALARA/health physics supervisors, who were trying to keep doses ALARA. Time constraints by the contract caused increased numbers of workers in the drywell to complete the work quickly and increased the total dose. Typically, union workers were willing to delay work for ALARA procedure changes but supervisors kept pushing. As a result of this constant battle, the utility personnel strongly recommend that for contracts with time and money limits, a man-rem limit should also be included to force better planning and more efficient use of the workforce.

8.4 PEACH BOTTOM UNIT 2

Numerous problems were encountered with chemical decontamination of pumps and valves at Peach Bottom. Can-decon® solution passed through the pumps but could not reduce the hot spots sufficiently. PeCo eventually had to disassemble the pumps and decontaminate them by hydroblasting with a water and glass-bead slurry. As a result of PECo's experience, they recommend that the chemical decontamination method chosen and the procedure used should be carefully chosen for maximum benefit. If it is necessary to remove the pumps, this should be done as early as possible to minimize the doses received from these highly contaminated pieces of equipment.

Plugging of jet pump slip joints, nozzles, and annulus pumps to maintain water levels and shield internal rector vessel sources was ineffective in controlling leakage and caused more dose than it saved. PECo did find that vessel sources could be effectively shielded using various forms of lead shielding.

ALARA responsibility for the pipe replacement program at Peach Bottom was given to the pipe installer (CB&I). A conflict of interest resulted whenever ALARA concerns interfered with installation tasks. ALARA responsibilities would be best served by the utility or an organization independent from the primary contractor.

PECo agreed with the other utilities that extensive planning and coordination is necessary to keep work progressing smoothly and efficiently. Coordination of tasks should be done to minimize the overall man-rem for the entire project. General area decontamination techniques and adequate ventilation should be maintained. Hot spot shielding done as early as possible may reduce drywell exposure rates even more than expected.

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10.0 ACRONYMS

AE - acoustic emission

ALARA - as low as reasonably achievable

AP - alkaline permanganate

BWR - boiling-water reactor

Can-decon® - a registered trademark of London Nuclear, Inc.

CB&I - Chicago Bridge and Iron

Citrox - a decontamination process using citric and oxalic acids

Cfm - cubic feet per minute

CNS - Cooper Nuclear Station

CRC - corrosion resistant cladding

dpm - disintegrations per minute

DF - decontamination factor

EPRI - Electric Power Research Institute

GE - General Electric

HAZ - heat affected zone

HEPA - high-efficiency particulate air

HP - health physicist

HPT - health physics technician

HSW - heat sink welding

H_2WC - hydrogen water chemistry

IGSCC - intergranular stress corrosion cracking

IHSI - induction heating stress improvement

IRM - Institute for Resource Management

ISI - in-service inspection

JAJ - J. A. Jones

JPI - jet pump instrumentation

LOMI - low-oxidation metal ion, a decontamination process

LPHSW - last-pass heat-sink welding

M-K - Morrison-Knudsen

MNGP - Monticello Nuclear Generating Plant

MPC - maximum permissible concentration

NDE - nondestructive evaluation

NE&C - Nuclear Engineering and Construction (NSP)

NG - nuclear grade

NMP-1 - Nine Mile Point Unit 1

NMPC - Niagara Mohawk Power Corporation

NNIC - Newport News Industrial Corporation

NPPD - Nebraska Public Power District

NRC - Nuclear Regulatory Commission

NSP - Northern States Power Company

PECo - Philadelphia Electric Company

PWR - pressurized water reactor

RCP - reactor coolant pipe

RCW - reactor coolant water

RHR - residual heat removal

RPV - reactor pressure vessel

RWCU - reactor water cleanup

RWP - radiation work permit

RWRS - reactor water recirculation system

SAFT-UT - synthetic aperture technique of UT

SBLC - standby liquid control

scfm - standard cubic feet per minute

SHT - solution heat treating

SWP - special work permit

TLD - thermoluminescent dosimeter

TVA - Tennessee Valley Authority

TWP - task work package

UT - ultrasonic testing

WCR - weld crown reduction

WI - work instruction

WOR - weld overlay repair

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16. ABSTRACT (200 words or less)

Replacement of primary recirculating coolant pipe in BWRs is a major effort that has been carried out at a number of nuclear generating stations. This report reviews the planned or actual pipe replacement projects at six sites: Nine Mile Point-1, Monticello, Cooper, Peach Bottom-2, Vermont Yankee, and Browns Ferry-1. It covers the radiological issues of the pipe replacement, measures taken to reduce doses to ALARA, estimated and actual occupational doses, and lessons learned during the various replacements. The basis for the decisions to replace the pipes, the methods used for preparation and decontamination, the removal of old pipe, and the installation of the new pipe are briefly described. Methods for reducing occupational radiation dose during pipe repairs/replacements are recommended.

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