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STEAM GENERATOR REPAIR REPORT

REVISION 6

January 1979

TURKEY POINT UNITS 3 & 4

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STEAM GENERATOR
REPAIR REPORT

REVISION 5

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STEAM GENERATOR

REPAIR REPORT

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STEAM GENERATOR

REPAIR REPORT

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STEAM GENERATOR

REPAIR REPORT

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STEAM GENERATOR

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1.0 INTRODUCTION, SUMMARY AND CONCLUSIONS

The steam generators at Turkey Point Units 3 and 4 have experienced corrosion related phenomena that require periodic inspection and plugging of steam generator tubes to ensure their continued safe operation. It is conceivable that continuation of the corrosion mechanism, referred to as denting, could eventually result in unacceptable inspection intervals and reductions in unit power. Thus, primarily economic considerations may require repair of the steam generators.

The Florida Power & Light Company (FPL) is currently pursuing engineering and
licensing activities to enable it to affect a timely steam generator repair 1
on Unit 3 and/or Unit 4 should the need arise.

This report discusses the safety-related aspects associated with repair of the steam generators via replacement of the lower portion of the in situ units with shop fabricated replacement lower assemblies. It does not address the in situ retubing alternate.

As discussed in Chapter 7.0, repair of the Turkey Point Steam Generators by removal/replacement of the lower assemblies through the containment equipment hatch is the preferred scheme. This report, through Revision 6, has discussed the actual implementation of this repair by removing the upper assembly by cutting the steam generator in the transition cone area and removing the lower assembly by cutting the reactor coolant piping. This method is referred to as the pipe cut approach.

However, events have occurred since the submission of Revision 6 which have led FPL to re-evaluate the pipe cut method for Turkey Point. The Surry experience has proven that the removal, refurbishment, and replacement of reactor coolant piping is time consuming and dose intensive. The total exposure experienced at Surry was 2140, substantially higher than 1301 man-rem originally estimated for each of the Turkey Point units.

Pipe alignment problems at Turkey Point can be expected to be more severe
than those at Surry. Unlike Surry, the steam generator supports at Turkey
Point are located at the base of the steam generators, thus hindering the 7
ease of movement of both the pipes and the workers during refitting of the 1
reactor coolant pipes. Taking these problems and the actual Surry experience into account, FPL recalculated the estimated occupational exposure to be incurred in making the repairs. Primarily due to the significant increase in the time required for alignment of the reactor coolant pipes, the predicted man-rems for the pipe cut method increased from 1301 man-rem to 2985 man-rem per unit.

As a result of these revised estimates, FPL investigated methods to reduce the expected man-rem exposure and the expected outage length associated with the repair. FPL has determined after extensive evaluation that an alternate method of removing the steam generator lower assembly from the reactor coolant system should result in a significantly reduced man-rem exposure and

outage length without reducing the margins of safety for the repair as shown in this report for the pipe cut method. This alternate method is referred to as the channel head cut method. FPL is now proposing to employ this method for the repair of the steam generators.

As with the pipe cut method, the channel head cut method is designed to remove the upper assembly of the steam generator from the lower assembly. However, unlike the pipe cut method, the channel head cut method calls for the separation of the lower assembly from the reactor coolant system by parting the channel head of the steam generator just below the tubesheet. This will eliminate the need to cut and refit the reactor coolant pipes. Openings in the lower assembly will then be sealed and the lower assembly will be removed from the containment as in the pipe cut method. The new lower assemblies will than be welded into the same position as were the original lower assemblies. Industry experience indicates that the lower assemblies can be welded into place without the alignment difficulties associated with the installation of the cut reactor coolant pipes. Additionally, a major part of the man-hours for the channel head cut will be spent in radiation fields which are generally an order of magnitude smaller than those for the reactor coolant pipe cuts. The lower radiation fields will produce additional man-rem savings. The estimated exposure for the entire channel cut method is 2084 man-rem per unit. 7

Revision 7 provides detailed man-rem estimates for the channel head cut method. Sections of this repair report which address expected radioactive releases during the repair, and sections which address postulated accidents associated with the repair have been re-examined, and FPL has determined that the conclusions reached for the pipe cut method are applicable to the channel head cut method.

1.1 SUMMARY OF STEAM GENERATOR REPAIR EVALUATION PROGRAM

1.1.1 Containment Entry and Exit of Steam Generator Lower Assemblies

A 1/2" - 1'0" scale model of the Unit 4 containment and internals and the Unit 3 containment in the area of the equipment hatch has been constructed to evaluate the feasibility of various pathways for removal of lower assemblies from containment and return of the shop fabricated lower assemblies to containment. Three pathways were evaluated in detail, viz., removal and entry via:

a. the equipment hatch,

b. an opening in the containment dome and

c. an opening in the containment wall.

Schedule, economic and safety evaluations indicated that use of the equipment hatch is the optimum pathway. Therefore, construction-related evaluations addressed in this report are for the equipment hatch pathway only.

1.1.2 Steam Generator Lower Assembly Characteristics

The existing steam generators will be parted in the upper section of the shell and at the channel head. The steam dome assemblies (upper portion of steam generator) will be removed and stored within containment. Subsequent to completion of the installation of the new lower assemblies, the original steam dome assemblies will be welded to the new lower assemblies to complete the repair. 7

The shop-fabricated lower assemblies (see Figure 2.2-1) will be equivalent to the lower assemblies they replace. They will be designed to meet existing plant mechanical and performance characteristics, and safety-related parameters will remain consistent with those utilized in the FSAR analyses.

Features to mitigate the effects of corrosion-related phenomena will be incorporated in the design. These features will not adversely alter mechanical, performance or FSAR-related characteristics. In addition, the shop-fabricated lower assemblies will be designed and manufactured to current codes and manufacturing techniques. Thus, the replacement assemblies will reflect current technology. They will satisfy the legal requirement of being equivalent to the units they replace (which were manufactured to the 1965 edition of Section III, ASME Boiler and Pressure Vessel Code).

1.1.3 Safety-Related Considerations

The potential impact of the repaired units on each appropriate accident analyzed in the FSAR has been evaluated. Because of the essential duplication of safety-related parameters, qualitative discussion is sufficient to demonstrate the appropriateness of the repaired steam generators to accommodate FSAR accidents.

Onsite transportation and handling of lower assemblies have been evaluated. Even though administrative procedures and temporary protection to buried pipes and conduits are considered sufficient to preclude occurrence of these types of mishaps, construction incidents are postulated to occur. The ability of the plant to accommodate these events is demonstrated.

The units will be repaired in series, i.e., one unit will be conducting normal power operations while the second is undergoing steam generator repair. To obviate the need to evaluate construction incidents within the containment undergoing repair, the reactor core will be off-loaded and transferred to the
fuel storage building prior to commencement of major repair activities within 1
containment.

1.1.4 ALARA Considerations

Estimates have been made of the exposure to personnel involved in the repair activity. This evaluation indicates that the reduction in man-rem exposures currently being incurred during tube inspection and plugging operations will offset, in a reasonable time period, the man-rem exposure incurred during the steam generator repair.

1.1.5 Offsite Radiological Considerations

Radiological evaluations of the gaseous and liquid releases attributable to the steam generator repair have been conducted. The resulting releases are a
fraction of the total releases during the period July 1978 through June 1979, 7
i.e., those associated with normal operation of the facility. The steam generator repair will in effect provide a reduction in future radioactive effluents from the plant because the repair will result in enhanced steam
generator integrity, thus reducing primary to secondary system leakage. 1

1.1.6 Unique Aspects of the Program

The shop fabrication of the lower assemblies will be conducted in accordance with standard practices. Welding of the steam dome assembly to the lower assembly in the field was utilized in the installation of the existing steam generators, which were shipped in two sections. Concrete removal and replacement will be accomplished utilizing standard construction practices. Transport and lifts of heavy vessels well in excess of the approximately 186-ton 7
lower assemblies are commonplace. Movement of heavy loads across buried pipe and electrical conduit is commonplace during construction of power plants, e.g., during construction of Units 3 and 4 the steam generators and reactor vessel were transported over the same pipe and conduit that the lower assemblies will pass over, without incident. In summary, the repair program will utilize tried and proven manufacturing and construction practices.

1.1.7 Steam Generator Disposal

The repair activity and ultimate disposal of the existing lower assemblies are separable issues. This report discusses the various means by which the steam generators can be disposed of to demonstrate the feasibility of disposal. The evaluation indicates that economic consideration will likely determine the method to be utilized. Depending on the method chosen, during the time between removal from containment and ultimate disposal, the lower assemblies will either be stored onsite in a temporary storage facility, or 7
placed in a laydown area to await immediate barge shipment to a burial facility.

1.2 IDENTIFICATION OF PRINCIPAL AGENTS AND CONTRACTORS

The Florida Power & Light Company, hereinafter FPL, is a public utility corporation duly authorized and existing under the laws of the state of Florida. FPL is the sole owner and operator of the Turkey Point Plant.

FPL has been actively engaged in nuclear power for a number of years and currently operates 3 nuclear power units: Turkey Point Units 3 and 4 and St. Lucie Unit 1. This represents a total operating experience of approximately 11 reactor years. Construction of St. Lucie Unit 2 is in progress. FPL will have overall responsibility for the steam generator repair program.

Bechtel Power Corporation has been retained by FPL as the Architect Engineer. Bechtel has been continuously engaged in construction and engineering activities since 1898. Since the close of World War II, Bechtel has placed strong emphasis on electrical power generation projects. During this period, Bechtel has been responsible for the design and/or construction of more than 237 thermal generating units, representing more than 118,000 MW_e of new generating capacity. Of this number, a nuclear capacity of more than 69,000 MW_e has been or is currently being engineered.

For more than 20 years, Bechtel has been actively working on nuclear power plants. Responsibiliies have covered design, construction, startup, site surveys, license applications, feasibility studies, and equipment procurement. The ratings of thermal generating plants designed by Bechtel range to 1,300 MW_e per unit and include most types of station design and arrangement. The majority of contracts for these facilities provided Bechtel with complete responsibility for engineering, construction, and startup.

Westinghouse Electric Corporation manufactured the existing steam generators. Westinghouse experience in nuclear plants for the electric utility industry is demonstrated by the pressurized water reactor plants that Westinghouse has designed, developed, and manufactured. There are 105 Westinghouse pressurized water reactor plants to date, including those plants in operation, currently under construction or on order.

Westinghouse Electric Corporation has long held a position of leadership in the electrical manufacturing industry. Traditionally this leadership has been based on technological development of both standard and new products, reliability and product quality. Through early participation in basic research and basic engineering development, Westinghouse has established a broad technological foundation in nuclear power application. This has been followed by a continuing program of sound technological development which enables Westinghouse to offer to the electric utility industry a reliable and safe source of power from the atom and any services related to the maintenance of nuclear power plants.

1.3 OTHER CONSIDERATIONS

Repair or replacement of equipment at a power plant, performed in accordance with appropriate procedures, is a maintenance activity that is routinely conducted. Because of the scope of the steam generator repair, it was considered prudent to evaluate this activity to determine:

a. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c. If the margin of safety as defined in the basis for any technical specification is reduced.

Each FSAR accident analysis has been evaluated to determine if the parameters of the repaired steam generators would alter the conclusions reached in the FSAR. Additionally, the construction incident potential has been evaluated to determine the presence of any new or unique accidents, the potential impact on the ability to shut down the operating unit and maintain it in a safe shutdown configuration, and the potential impact on cooling spent fuel. The evaluation of items a, b and c above indicates that the repair activity does not involve an unreviewed safety question, nor the need for additional or modified Technical Specifications.

1.4 CONCLUSIONS

The fundamental conclusions reached are that the steam generator repair can be conducted utilizing proven manufacturing and construction techniques and that the repair program does not result in any adverse impact on the ability

to shut down the operating unit, maintain a safe configuration, and cool stored spent fuel. Additionally, current FSAR safety analyses are applicable to the repaired steam generators. The detailed bases supporting these conclusions are provided in the report that follows.

2.0 REPLACEMENT COMPONENT DESIGN

Westinghouse will shop fabricate new steam generator lower assemblies as illustrated by Figure 2.2-1. The design of the lower assemblies will match the design performance of the lower assemblies being replaced. However, several design improvements that do not alter mechanical, performance and FSAR parameters are included in the design. These design features will improve flow distribution, improve tube bundle access and reduce secondary side corrosion. This section discusses the design and manufacture of the lower assemblies.

2.1 COMPARISON WITH EXISTING COMPONENT DESIGN

2.1.1 Parametric Comparison

The steam generators for the Turkey Point plants, upon completion of the repair, will have physical, mechanical and thermal characteristics consistent with the original design and safety analysis as currently documented in the FSAR. The existing steam generators were built to the 1965 edition of the ASME Boiler and Pressure Vessel Code (ASME Code); the new component parts of the steam generators will be designed and fabricated based upon the 1974 edition of the ASME Code, including all addenda through Winter 1976. The Stress Report will be based upon the 1965 edition of the ASME Code, including all addenda through Summer 1965. The replacement lower assemblies will be fabricated and analyzed to standards equivalent to the original units.

The replacement lower assembly will incorporate a number of refinements in design which are discussed in Section 2.2. During 1975 several modifications were made to the installed steam generators to increase performance and promote reliability. These modifications (described and noted in the text) will be retained or improved with the replacement lower assemblies. The modifications accomplished at that time consisted of removing the downcomer resistance plate, improving the moisture separators, modifying the blowdown arrangement inside the steam generators, installing tube lane blocking devices and modifications to the feedring to improve performance. These modifications increased the circulation ratio and improved the units' ability to resist sludge buildup.

Design data for the steam generators is presented in Table 2.1-1 allowing comparison between the present steam generators and the repaired units. Improvements have been made for increased access to the secondary side of the steam generators (currently six 6-inch hand holes around the bundle in the tubesheet area are planned). The thermal data for each steam generator will remain the same as the original steam generators.

Since the replacement lower assemblies have been designed to incorporate changes based on field experience, a number of minor changes in specific components have been made which could affect the thermal hydraulic performance of the unit. In order to maintain the original thermal and hydraulic conditions, adjustment of heat transfer surface parameters was necessary; changes in the

support plate configuration and desire to improve the circulation ratio resulted in a decrease in the number of tubes. These modifications resulted in the reactor coolant water volume in the steam generator being reduced slightly, the secondary side volume being increased slightly, a slight decrease in the amount of heat transfer surface area, as well as a slight increase in the heat transfer coefficient. Imposing closer manufacturing tolerances on the tube wall thickness results in an increase in the overall heat transfer coefficient (approximately 2.5%) for the repaired units. This increase in heat transfer coefficient offsets the decrease in heat transfer area (approximately 2.2%) so that steam generator heat transfer remains essentially unchanged. 5

Materials used in the fabrication of the replacement lower assemblies will be procured to the requirements of the 1974 edition of the ASME Code, including all addenda through Winter 1976. These materials will be identical to those used in the original steam generators except where specific design changes have been recently incorporated or fabrication practice has changed. Specific examples of these occurrences are enumerated as follows: plate material used in the secondary shell formation has been changed to SA-533 Grade A Class 2 from SA302 Grade B Class 1 as a result of fabrication practices; support plate material has been changed to SA-240 Type 405 from SA-285 Grade C as a result of design changes to prevent corrosion. Material changes due to design improvements do not degrade the physical, mechanical and thermal properties of the steam generators. Further discussion is provided in Section 2.2 and Table 2.1-2 enumerates past and present applications of materials.

2.1.2 Physical Compatibility With Existing Steam Generators and Systems

New steam generator lower assemblies (see Figure 2.2-1) will be provided. These lower assemblies are designed to be identical physical replacements for the existing units. Outside overall dimensions will be the same as will be the location of nozzles and support attachments. Interfaces between the steam generators and plant components and systems are maintained. Dry and wet weights of the steam generators will remain approximately the same as will the center of gravity, therefore, no changes to the present supports or their configuration are necessary.

2.1.3 ASME Code Application

The present operating steam generators were designed and constructed to the requirements of the 1965 edition of the ASME Code, Section III, Summer 1965 addenda. The replacement assemblies will be fabricated to the requirements of the 1974 edition of the ASME Code including all addenda through Winter 1976. Design of the steam generators will be consistent with the original design of the reactor coolant system as well as the upper shell assembly of the steam generators which will not be replaced. Materials to be used in fabrication will be procured to the requirements of the current codes to facilitate construction. All material certification tests will be performed and recorded as required by current versions of the code. None of the requirements imposed on the replacement assemblies will inhibit the capability of the steam generators to meet performance and FSAR safety requirements.

2.1.4 Regulatory Guide Application

The compilation below addresses Regulatory Guides considered applicable to the fabrication of the replacement lower assemblies. It must be noted that these guides were issued subsequent to construction and operation of this facility.

The intent is to accommodate, consistent with facility design and repair program objectives, the guidance provided by these regulatory guidelines.

1.26 Quality Group Classifications and Standards for Water, Steam and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 3, February 1976)

Westinghouse utilizes the classification system of ANSI N18.2A-1975 for water and steam containing components. This classification method assigns safety-related components to safety classes. Assignment of the primary side of the steam generator to Safety Class 1 and the secondary side to Safety Class 2 is consistent with the quality groupings which would result from this regulatory guide and 10 CFR 50.55a.

1.28 Quality Assurance Program Requirements (Design and Construction) (Safety Guide 28, June 1972)

The Westinghouse position on Regulatory Guide 1.28 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A. 3

1.31 Control of Stainless Steel Welding (Rev. 1, June 1973)

The Westinghouse production weld verification program, as described in WCAP-8324-A, was approved by the NRC as a satisfactory substitute for following the recommendations of the NRC Interim Position on Regulatory Guide 1.31 (4/74). The results of the verification program support the hypothesis presented in WCAP-8324-A; these results have been summarized and documented in WCAP-8693, which has been submitted to the NRC for information.

1.34 Control of Electroslag Weld Properties (December 28, 1972)

Where electroslag welding is used, Westinghouse requires its suppliers to follow the recommendations of this guide.

1.37 Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March 16, 1973)

The Westinghouse position on Regulatory Guide 1.37 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A. 3

1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling for Nuclear Power Plants (Rev. 2, May 1977) 3

The Westinghouse position on Regulatory Guide 1.38 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occured during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.43 Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (May 1973)

The Westinghouse Tampa Division will use materials made to fine-grain practice or which are not susceptible to underclad cracking. These materials do not require the controls listed in the guide.

1.44 Control of the Use of Sensitized Stainless Steel (May 1973)

All of the unstabilized austenitic stainless steels used for component parts of the reactor coolant pressure boundary are utilized in the final heat treated condition required by the respective ASME Code, Section II, material specification for the particular type or grade of alloy. Processing and fabrication are performed using established methods and techniques to avoid sensitization. Westinghouse has verified that these practices will prevent sensitization by performing corrosion tests on as-received wrought material, as well as on production and qualification weldments. In addition, the water chemistry in the reactor coolant system is controlled to prevent intergranular attack of unstabilized stainless steels; the effectiveness of these controls has been demonstrated by both laboratory tests and operating experience.

1.48 Design Limits and Loading Combinations for Seismic Category I Fluid System Components (May 1973)

Westinghouse meets and will continue to meet the requirements of General Design Criterion 2 and will thereby satisfy the concerns of Regulatory Guide 1.48. The loading combinations and design limits used in the code stress analysis of the steam generator will be the same as those in the Turkey Point FSAR.

1.50 Control of Preheat Temperature for Welding of Low-Alloy Steel (May 1973)

Westinghouse practices are in agreement with Regulatory Positions C.1.a, C.3 and C.4. For Regulatory Position C.1.b, Westinghouse qualifies welding procedures within the preheat temperature ranges required by Section IX of the ASME Code. For Regulatory Position C.2, Westinghouse uses the methods documented in WCAP-8577-A, which has been accepted by the NRC.

1.58 Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (August 1973)

The Westinghouse position on Regulatory Guide 1.58 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A. 3

1.64 Quality Assurance Requirements for the Design of Nuclear Power Plants (Rev. 1, February 1975)

The Westinghouse position on Regulatory Guide 1.64 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A. 3

1.66 Nondestructive Examination of Tubular Products (October 1973)

Steam generator nozzles are either radiographed or ultrasonically tested in the circumferential and axial directions in accordance with the guides' positions. Steam generator tubing receives eddy current, circumferential ultrasonic testing, and hydrostatic testing to satisfy the guides' recommendations.

1.71 Welder Qualification for Areas of Limited Accessibility (December 1973)

Westinghouse practice does not require qualification of welders for areas of limited accessibility. Shop welds are repetitive and closely supervised and the ASME Code, Sections III and IX requirements are followed.

1.83 Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)

Westinghouse steam generators are designed to permit access to tubes for inspection and plugging. A preservice inspection of the steam generators will be conducted to establish baseline conditions.

1.84 Code Case Acceptability-ASME Section III Design and Fabrication (June 1974) (Rev. 1, April 1975) (Rev. 2, June 1975) (Rev. 3, September 1975) (Rev. 4, November 1975) (Rev. 5, February 1976) (Rev. 6, May 1976) (Rev. 7, August 1976) (Rev. 8, November 1976) (Rev. 9, March 1977)

1.85 Code Case Acceptability-ASME Section III Materials (June 1974) (Rev. 1, April 1975) (Rev. 2, June 1975) (Rev. 3, September 1975) (Rev. 4, November 1975) (Rev. 5, February 1976) (Rev. 6, May 1976) (Rev. 7, August 1976) (Rev. 8, November 1976) (Rev. 9, March 1977)

1. Westinghouse controls its suppliers to:

   a. Limit the use of code cases to those listed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered, except as allowed in item 2 below

   b. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered, where use of such cases is needed by the supplier

   c. Allow continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended

2. Westinghouse seeks NRC permission for the use of code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered and permits supplier use only if NRC permission is obtained or is otherwise assured (e.g., a later version of the regulatory guide includes endorsement)

1.88 Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (Rev. 2, October 1976)

The Westinghouse position on Regulatory Guide 1.88 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A. 3

1.123 Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (Revision 1, July 1977)

The Westinghouse position on Regulatory Guide 1.123 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A. 3

2.2 COMPONENT DESIGN IMPROVEMENTS

As noted, the physical, thermal and hydraulic characteristics of the steam generators will essentially duplicate those of the original units. However, design changes which do not alter FSAR safety requirements have been incorporated in the design. These changes increase the operating reliability and improve resistance to corrosion of the secondary side thereby minimizing the potential for future repair efforts. Figure 2.2-1 illustrates some of these improvements. It should be noted that some of these features have been installed in the in situ units (see Section 2.1.1).

Research, development and testing have been utilized to select design parameters, material and component configurations which will prevent degradation of the repaired steam generators. Confirmatory tests in model boilers and other tests on the material and component configuration are continuing.

2.2.1 Design Refinements To Prevent And Inhibit Corrosion

2.2.1.1 Increased Circulation Ratio

Circulation ratio is defined as thc total tube bundle flow divided by the feedwater flow and is inversely proportional to the steam quality exiting the tube bundle. As the circulation ratio increases, certain parameters of the steam generator, such as lateral velocity sweeping across the tubesheet, steam quality, void fraction and number of tubes exposed to sludge, change in a favorable direction. Low steam quality in the bundle reduces tube exposure to local steam blanketing. This also reduces the number of potential areas of concentration for chemical impurities. In addition, higher circulation ratios increase the fluid velocity sweeping across the tubesheet to the center of the bundle. Specific design changes, such as the quatrefoil plates (see Subsection 2.2.1.8), modification in the tube bundle size and wrapper to shell distance, influence the circulation ratio.

2.2.1.2 Flow Distribution Baffle

A flow distribution baffle has been provided 18 inches above the tubesheet. This baffle has a cut out center section and oversized drilled tube holes. The increased circulation ratio provides a greater lateral flow across the tubesheet surface. The baffle plate will assist in directing this flow across the tubesheet then up the center of the bundle through the center cutout. The design is sized to minimize the number of tubes exposed to sludge. Consistent with this purpose, the design causes the sludge to deposit in and near the center of the bundle at the blowdown intake. The flow distribution baffle plate material is ferritic stainless steel. Figure 2.2-2 illustrates the flow distribution baffle.

While the baffle will direct flow toward the center of the bundle, the average velocity around the tubes will be sufficient to prevent sludge from settling. In addition, as noted, access holes have been provided to allow sludge lancing of the baffle plate.

2.2.1.3 Improved Internal Blowdown Design

Each steam generator will be designed to have two 2-inch schedule 40 Inconel internal blowdown pipes. The blowdown rate from the steam generators is varied as required by chemistry conditions in the feedwater and as monitored in the blowdown. Maintenance of the steam side water chemistry is assisted through the use of the blowdown system. Continuous blowdown of the steam generator provides a dynamic system which is constantly removing impurities from the steam generator. During hot standby and hot functional testing, blowdown is employed, as needed, to maintain the steam generator chemistry within specification. The blowdown intake location is coordinated with the baffle plate design so that the maximum intake is located where the greatest amount of sludge is expected to deposit. The improved blowdown system will allow higher capacity blowdown in comparison with the present blowdown arrangement.

2.2.1.4 Tube Expansion in Tubesheet

Following insertion into the tubesheet hole, tack rolling, welding and gas leak testing, the tubes are expanded to the full depth of the tubesheet hole. Full-depth expansion prevents crevice boiling. In addition it prevents a buildup of impurities from forming in the crevice region. The present steam generator tubes were only partially expanded in the tubesheet.

2.2.1.5 Thermally Treated Inconel 600 Tubing

Research by Westinghouse has determined that significant improvement in the stress corrosion resistance of Inconel 600 tubing can be achieved by modification of the metallurgical structure through thermal treatment. The primary objective of this treatment is to develop an improved metallurgical structure, associated with grain boundary precipitate morphology, which provides increased margin with respect to stress corrosion performance. Several benefits result from this treatment such as improved resistance to stress corrosion cracking in NaOH, resistance to intergranular attack in oxygenated environments, resistance to intergranular attack in sulphur-containing species and reduction of residual stress imparted by tube processing.

Studies conducted at Westinghouse and elsewhere have indicated that certain heat treatments can improve caustic stress corrosion resistance but result in a chromium-depleted grain boundary layer (sensitization) which is not as resistant to off-chemistry environments, should they be experienced. However, analysis of available data also indicates that there is a broad band of temperature and time within the typical sensitization range for Inconel 600 which provides improved resistance to stress corrosion cracking in both caustic and pure water environments. Thermal treatment in this time-temperature band avoids formation of the chromium depleted grain boundary layer. The thermal treatment to be used will be within this time-temperature band.

2.2.1.6 Offset Feedwater Distribution

Previously, feedwater flow within the steam generators was modified so that 80 percent of the flow is directed to the hot leg side of the bundle and 20 percent of the flow is directed to the cold leg side of the bundle. This reduces the steam quality in the hot leg side of the bundle and raises the steam quality in the cold leg side of the bundle. The effect of these changes in steam quality is to shift the point of highest steam quality at the tubesheet elevation toward the center of the bundle. The point of highest steam quality has the lowest density and is, therefore, a likely region for chemical concentration and sludge deposition. This area is utilized for location of the blowdown intake. Feedwater flow distribution is accomplished by providing a greater number of flow paths on the portion of the feedwater ring which traverses the hot leg side of the tube bundle. These modifications will be maintained in the replacement assemblies.

2.2.1.7 Corrosion Resistant Support Plate Material

Corrosion in the crevice between the tube and tube support plate has led to denting of the tubing in that area and in some cases affected the steam generator performance in general. Alternative support plate materials have been evaluated, and SA-240 Type 405 ferritic stainless steel has been selected as the optimum material for this application. This material is ASME Code-approved and is resistant to corrosion with the chemistry expected during the operation of the steam generator. In addition, SA-240 has a low wear coefficient when paired with Inconel and has a coefficient of thermal expansion similar to carbon steel. Corrosion of SA-240 results in an oxide which has approximately the same volume as the parent material, whereas corrosion of carbon steel results in oxides which have approximately two times the volume of the parent material. Type 405 also has material properties such as machineability and weldability which are comparable to carbon steel. In addition to the tube support plates, the baffle plate (discussed in Subsection 2.2.1.2) will be constructed of SA-240 Type 405.

2.2.1.8 Quatrefoil Tube Support Plates

The quatrefoil tube support plate design, illustrated by Figure 2.2-3, consists of four flow lobes and four support lands. The lands provide support to the tube during all operating conditions, while allowing flow around the tube. This design has a lower pressure drop than the most current circulation hole designs. This low secondary pressure drop will increase the circulation ratio which, when combined with other improvements, translates into higher sweeping velocities and fewer tubes exposed to a low steam quality at the tubesheet. This design directs the flow along the tubes which will limit steam formation and chemical concentrations at the tube-to-tube support plate intersections. The quatrefoil support plate design results in higher average velocities along the tubes, preventing sludge deposition. The combination of higher velocities in the support plate region and corrosion resistant material will minimize the possibility of support plate corrosion.

2.2.2 Design Refinements To Improve Performance

In the course of evolution of the steam generator design, as derived from operating experience and ongoing research and development programs, certain improvements and refinements have been incorporated in recent designs to improve performance of thermal hydraulic characteristics. These improvements are included in the FPL design and are discussed below. They do not alter FSAR safety requirements.

2.2.2.1 Recessed Tube to Tubesheet Weld

The tubes on the replacement lower assemblies will be recessed slightly into the tubesheet holes and then welded to the tubesheet cladding. Elimination of the protruding tube stub of the original design results in lower entry pressure losses and, therefore, a lower pressure drop in the primary loop. In addition, a possible point of crud buildup and corrosion is likely avoided with this design. This is illustrated in Figure 2.2-4.

2.2.2.2 Tube Lane Blocking Device

Recirculating water exiting at the bottom of the wrapper will tend to preferentially channel to the tube lane and bypass part of the tube array. In order to prevent this tube bundle bypass, a series of plates was installed in the tube lane during prior modifications. These plates are arrayed so that there will be minimal interference with sludge lancing. These blocking devices will be retained in the replacement units.

2.2.2.3 Moisture Separator Improvements

Since the circulation ratio in the steam generator has increased, the duty for the moisture separator equipment will increase. To accommodate this increase, several improvements will be incorporated. New demister vanes will be installed to increase the efficiency of the moisture separators. Perforated plates will be installed on the face of the demister vane housing to distribute the flow evenly through the demisters and provide better moisture separating. The swirl vane barrels previously modified with optimized orifice plates will be realigned. These improvements are shown in Figure 2.2-5.

2.2.3 Design Changes To Improve Maintenance And Reliability

Operational experience, including necessary maintenance and repair, has resulted in certain changes in design which are directed to improving the maintainability and ultimately the reliability of the units. Other changes have been incorporated to prevent occurrences of operational problems which have been experienced. These changes are discussed below and do not alter performance or FSAR safety requirements. (It should be noted that the details provided below may be altered during detailed design.)

2.2.3.1 Access Ports

The lower assemblies will be constructed with additional access ports. Four 6-inch access ports will be located slightly above the tubesheet, approximately 90 degrees apart, with two located on the tube lane. Two 6-inch access ports will be located on the tube lane, between the flow distribution baffle and the first tube support plate. The addition of these access ports will improve and promote inspection of the tubesheet and flow distribution baffle and assist in sludge lancing.

2.2.3.2 Wet Layup Nozzle

A 2-inch nozzle is being added to the upper shell to facilitate the wet layup of the steam generators during periods of inactivity. The wet layup nozzle can be used for addition of chemicals during these periods to prevent any excursions of the water quality in the steam generator. The nozzle can also be used in conjunction with other systems to circulate water through the steam generator during periods of layup.

2.2.3.3 Primary Shell Drain

A 3/8-inch primary shell drain is included in the channel head to improve drainage of the channel head. The improved drainage will lessen downtime and facilitate any maintenance or inspection to be conducted in the channel head.

2.2.3.4 Primary Closure Rings

Closure rings will be welded inside the channel head at the base of each primary nozzle so that closure plates can be installed during primary chamber maintenance. This design allows the plates to be bolted to the rings for quick installation and removal. Closure plates allow maintenance or inspection to be conducted in the channel head with the reactor cavity flooded.

2.3 SHOP TESTS AND INSPECTIONS

The tests and inspections required by the ASME Code, Section III will be conducted during the fabrication of the steam generator lower assembly. In addition to these ASME requirements, further tests and inspections will be conducted at the fabrication facility. The primary side of the steam generator will be hydrotested at the shop in accordance with approved procedures. Each tube will be individually hydrotested prior to use in fabrication. After the tube bundle installation is completed a gas leak test will be performed to demonstrate the integrity of the tube-to-tubesheet welds.

TABLE 2.1-1

STEAM GENERATOR DESIGN DATA (PER STEAM GENERATOR)

	Original	Repaired	
Design Pressure, Reactor Coolant/Steam, psig	2485/1085	N.C.*	
Reactor Coolant Hydrostatic Test Pressure (tube side), psig	3107	N.C.	
Hydrostatic Test Pressure, Shell Side, psig	1356	N.C.	
Design Temperature, Reactor Coolant/Steam, ^{o}F	650/556	N.C.	
Steam Conditions at 100% Load, Outlet Nozzle:			
Steam Flow, lb per hr	3.2×10^6	N.C.	
Steam Temperature, ^{o}F	516.0	N.C.	
Steam Pressure, psig	770	N.C.	
Feedwater Temperature at 100% Load, ^{o}F	436.5	N.C.	
Overall Height, ft-in	63-1.6	N.C.	
Shell OD, upper/lower, in.	166/127	N.C.	
Shell Thickness, upper/lower, in.	3.5/2.63	N.C.	
U-tube OD, in.	0.875	N.C.	
Tube Wall Thickness, (nominal) in.	0.050	N.C.	
Number of Manways/ID, in.	4/16	N.C.	
Number of Handholes/ID, in.	2/6	6/6**	
Number of U-tubes	3260	3214	
Tube length (largest U-bend), in.	397.5	N.C.	
Total Heat Transfer Surface Area, ft^2	44,430	43,467	5
Reactor Coolant Water Volume, ft^3	945	935	
Reactor Coolant Flow, lb/hr	33.83×10^6	N.C.	
Secondary Side Volume, ft^3	4580	4596	5
Secondary Side Mass No Load, lbs	134,000	N.C.	
Secondary Side Mass 100% Power, lbs	76,300	80,300	
Center of Gravity (from the support pads), ft/in.	25/4	N.C.	

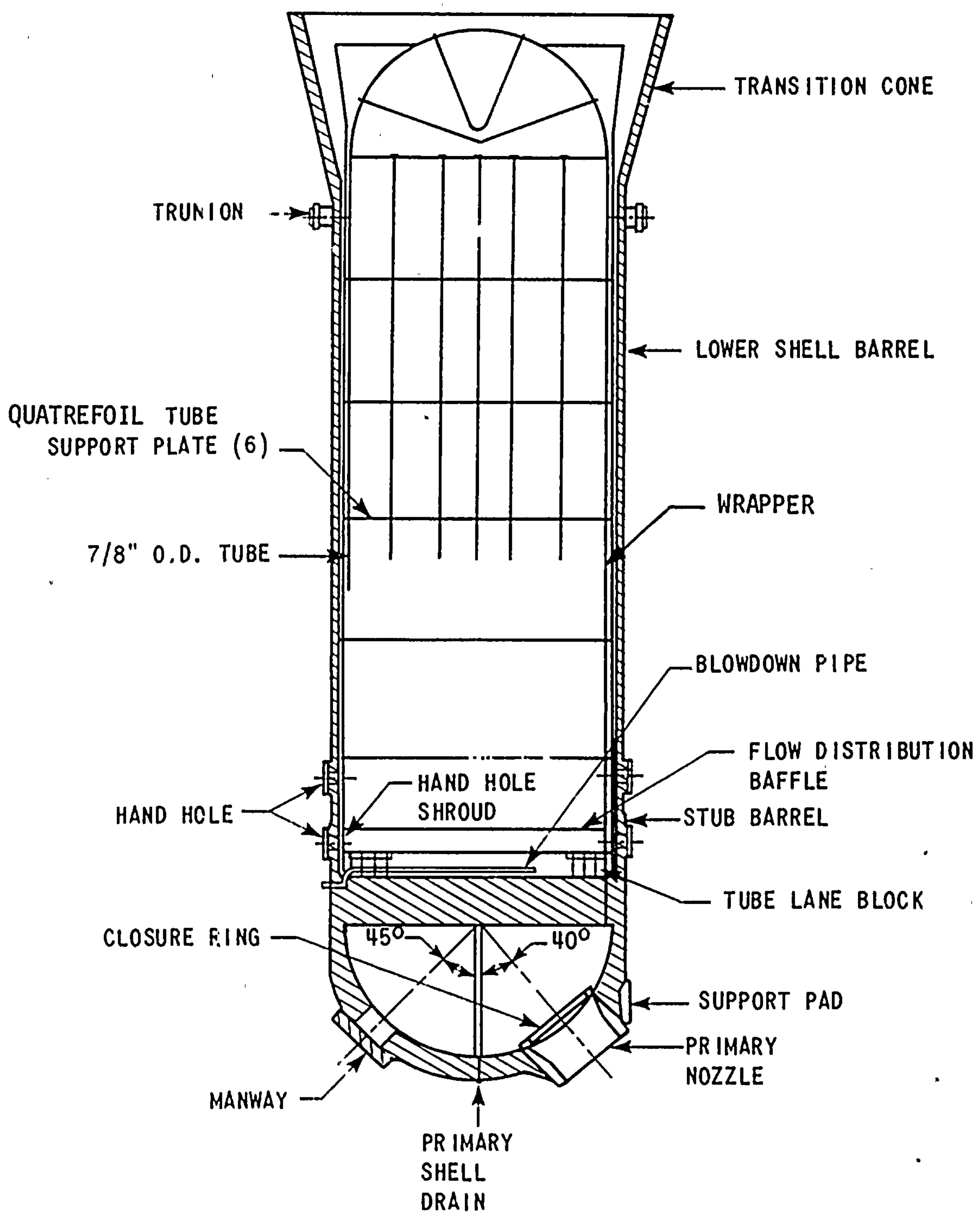
*No change

**Number and size of additional handholes may be altered during final design.

TABLE 2.1-2

STEAM GENERATOR MATERIALS

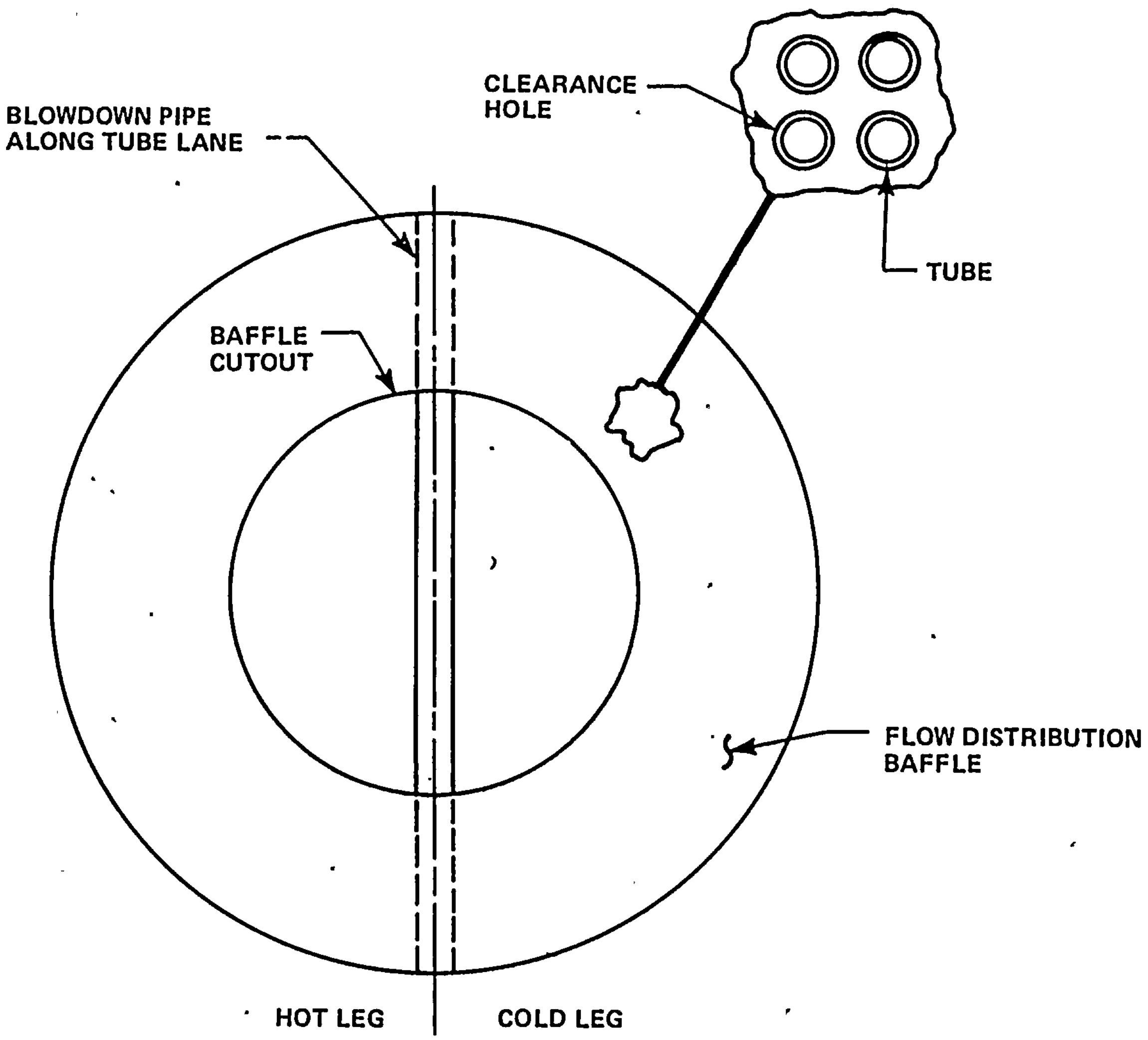
	Original	Repaired
Plate (shell courses)	SA-302 Grade B	SA-533 Grade A Class 2
Tube Sheet Forging	SA-336	SA-508 Class 2a
Channel Head Casting	SA-216 Grade WCC	SA-216 Grade WCC
Support Plates	SA-285 Grade C	SA-240 Type 405
Channel Head Cladding	Stainless Steel, Type 304 or equivalent	Stainless Steel, Type 304 or equivalent
Tube Sheet Cladding	Inconel	Inconel
Tubes	SB-163	SB-163 Thermally Treated



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR LOWER ASSEMBLY

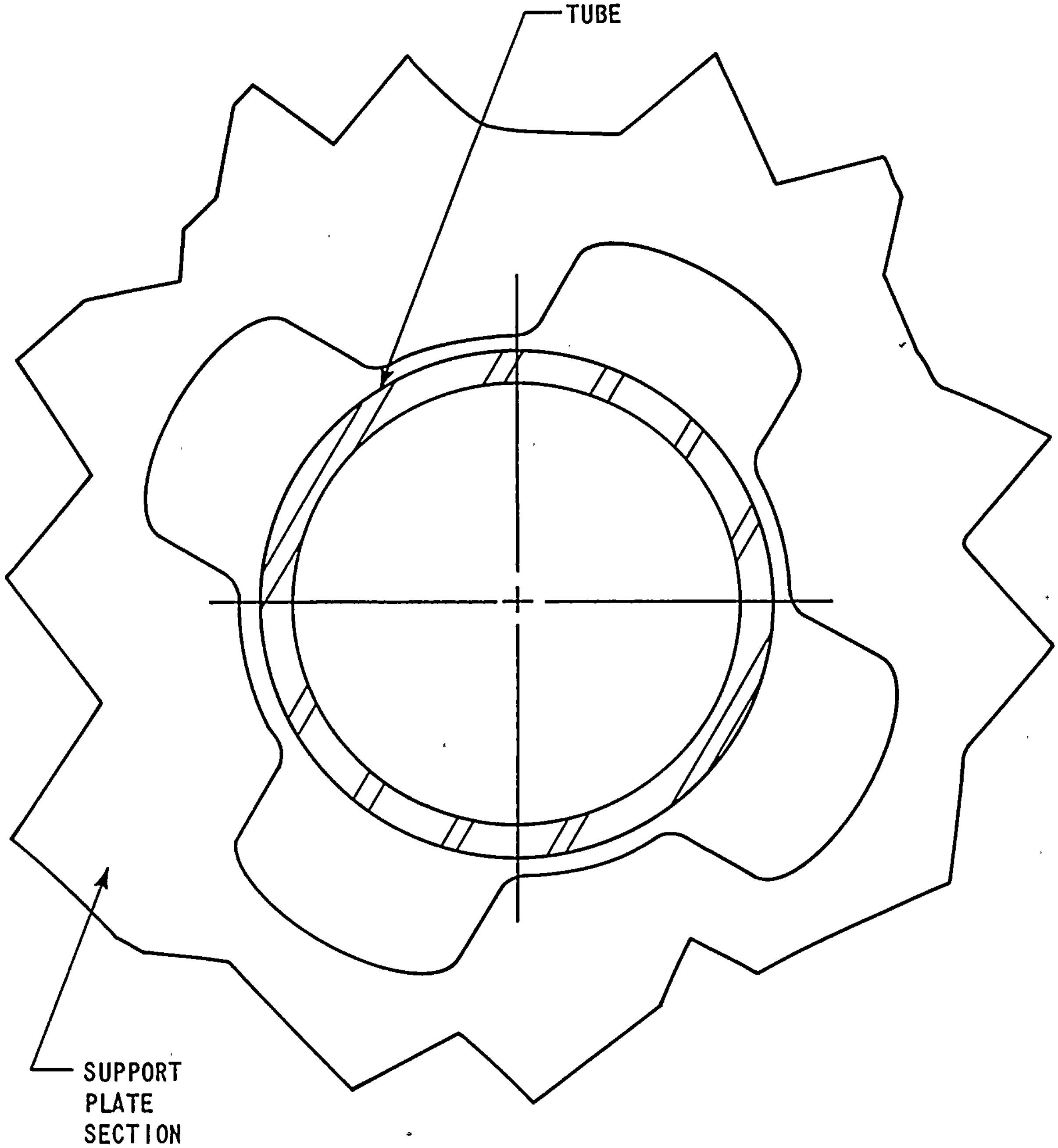
FIGURE 2.2-1



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

FLOW DISTRIBUTION BAFFLE AND BLOWDOWN

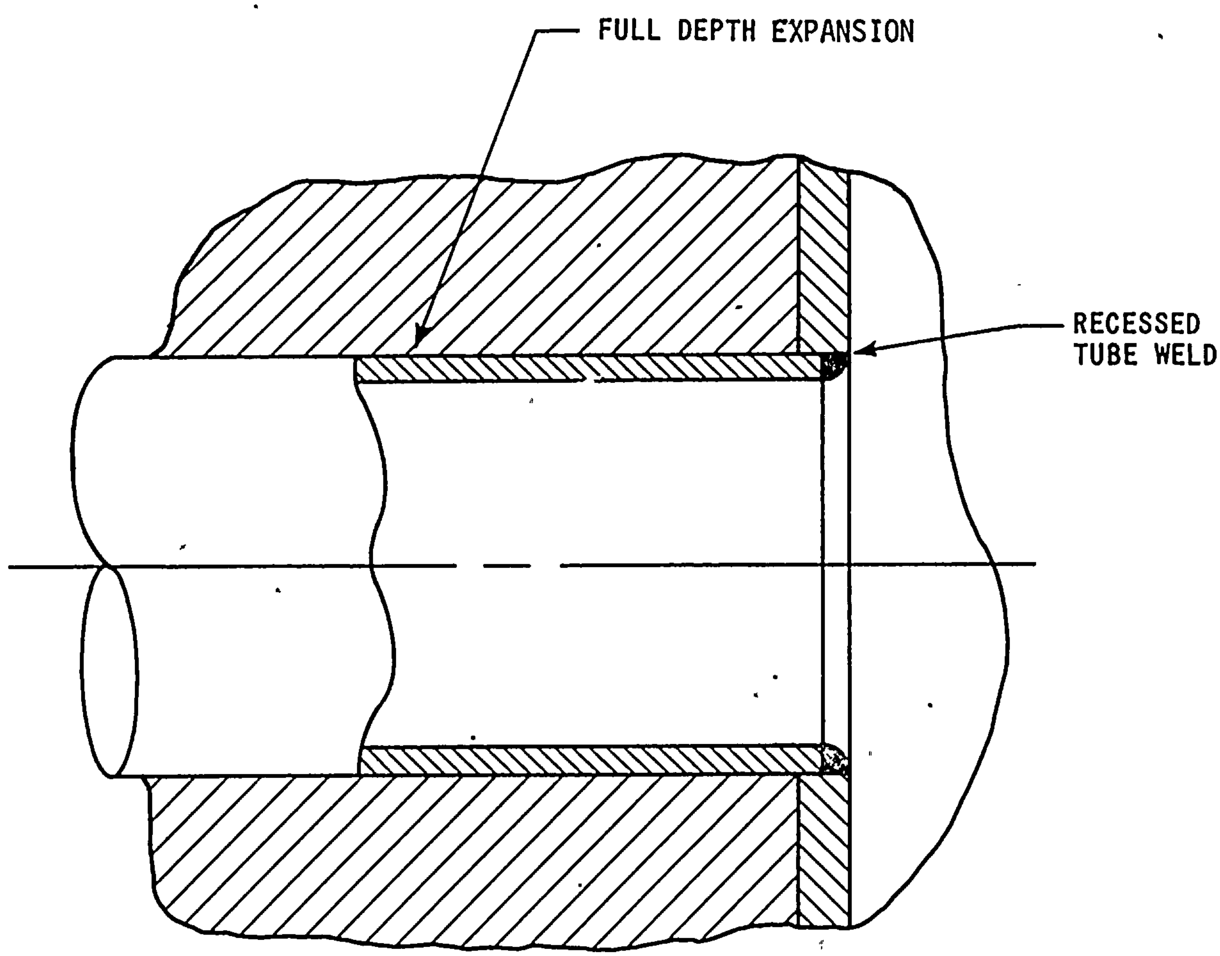
FIGURE 2.2-2



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

QUATREFOIL TUBE SUPPORT PLATES

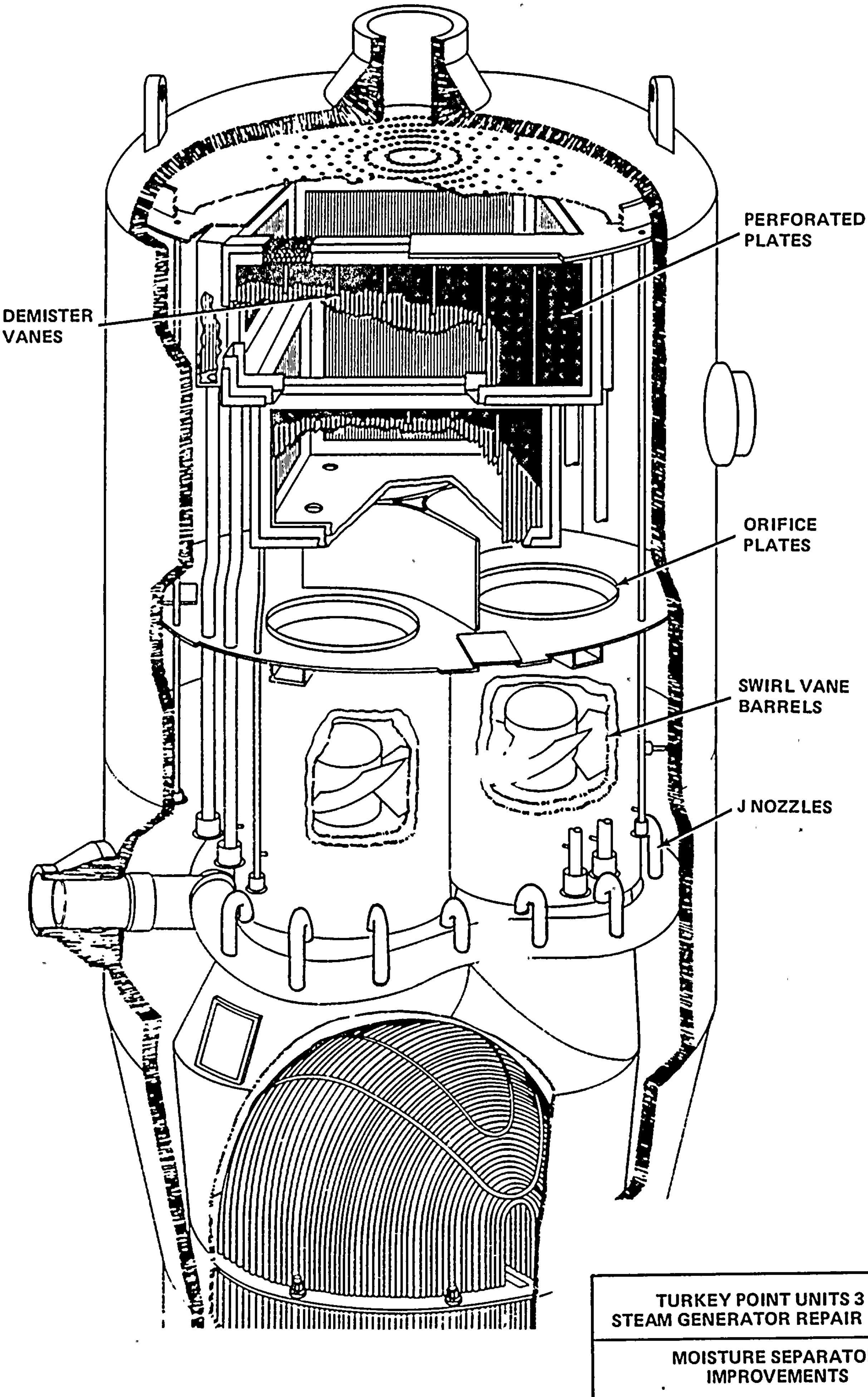
FIGURE 2.2-3



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

TUBE-TO-TUBESHEET JUNCTURE

FIGURE 2.2-4



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

MOISTURE SEPARATOR IMPROVEMENTS

FIGURE 2.2-5

3.0 COMPONENT REPLACEMENT PROGRAM AND PROCEDURES

This section discusses the engineering evaluation of the field activities required to implement the steam generator repair. Figure 3.0-1, Outage Sequence, and Figure 3.0-2, Removal Sequence, illustrate one possible lower assembly removal approach. It should be noted that implementation methods 7
and procedures may vary from that described below as engineering is finalized. The methods below are provided to demonstrate feasibility of implementation. Any changes incurred during detailed design will not alter the envelope of construction incidents postulated in Section 5.2.

The steam generator lower assemblies will be removed and replaced through the equipment hatch. The steam dome assemblies will remain inside the containment and will likely be stored at the operating floor level, elevation +58', until the new lower assemblies are brought in and installed.

Handling of the steam generator lower assemblies inside the containment may require an additional polar crane trolley with a capacity in the range of 250 tons. (Refer to Figure 3.0-2 sequences I through VI.) Handling through the hatch will require special track-mounted tailing devices attached to the steam generator. (Refer to Figure 3.0-2 sequences VII through XI.) Handling outside the containment will be by crawler lift crane or other 7
suitable method.

A rigging platform will be provided inside the containment. This platform will provide support for the lower assemblies while they are being maneuvered from the vertical position to the horizontal position. The temporary construction loads from this platform will be transmitted to the containment base mat. In addition, temporary laydown area will be provided for three steam dome assemblies weighing 100 tons each and for three sets of swirl vane assemblies weighing approximately 5 tons each. Adequate laydown area is available inside the containment, utilizing in some cases temporary beams to span certain operating floor slab areas.

Clearances to accomplish the removal of the lower assembly through the equipment hatch will be provided through the removal of internal concrete in the vicinity of the equipment hatch. The proposed portions to be removed are listed in Section 3.2.5. Impact on existing equipment is minimal and is described in Section 3.2.

The removal of lower assemblies through the existing equipment hatch will have minimal impact on the site layout in terms of new foundations required. Upon exiting from the equipment hatch, the lower assemblies will likely be placed on a lowboy type trailer or rubber wheeled transporter by one or more large cranes or other suitable method, and moved to a suitable location for temporary onsite storage. Special foundations will be required only for the 7
access platform described below, if utilized.

The existing access ramp to the equipment hatch will be modified as neces-
sary to enlarge the access area and to facilitate access by trucks and 7
mobile cranes, and transporting devices during steam generator movement.

No permanent modifications to existing structures are expected. 1

3.1 PATHWAYS AND CONSTRUCTION RESTRICTIONS

3.1.1 Site Preparation

3.1.1.1 Foundations

All heavy hoisting equipment will be located so that foundations will not
interfere with permanent plant installations, either above or below grade.
A steam generator removal platform, if utilized, will require the placement 7
of foundations as shown in Figure 3.1-1. Below grade emplacements may be
removed after completion of the repair work.

3.1.1.2 Roadways, Ramps, and Platforms

The existing access ramps to the equipment hatch of each containment will be
removed as necessary. A service platform (Figure 3.1-1) may be installed to
facilitate access into the containment. It would be designed to support the
weight of the lower assemblies and a manually powered materials cart of
approximately 20-ton capacity. It would also serve as a loading dock for 7
the loading/unloading of a number of trucks or mobile cranes simultaneously.
It would be equipped with jib cranes of 2 to 5 ton capacity for this
purpose.

Having established the method of recipt of lower assemblies and location of onsite disposal area, the following steam generator haul routes apply as shown on Figure 3.1-2:

a. Unit 3 and Unit 4 new lower assemblies were unloaded from barges in the turning basin at a point northeast of Unit 1, and transported south along the east plant road, around the southern end of the Radwaste Building, and then west to a prepared open area southwest of the Unit 4 Containment.

b. During the Unit 4 repair outage, the Unit 4 new lower assem- 7
blies will be transported north on the Unit 4 containment
access road to the equipment hatch.

c. During the Unit 3 repair outage, the Unit 3 new lower assemblies will be transported east then around the southern end of the Radwaste Building, north along the east plant road, then southwest on the containment access road to the equipment hatch area.

The results of all safety considerations as presented in Section 5.0 are bounding. 7

As required, additional ramps to access plant grade will be designed for the maximum wheel loads of anticipated transportation equipment. 1

3.1.1.3 Protection of Buried Facilities

Analyses have been performed for the following safety-related Units 3 and 4 buried facilities to ascertain in situ capability to accommodate transport loads (refer to Figure 3.1-2 for locations). These safety-related buried facilities include all those potentially affected by the haul routes associated with each steam generator site receipt alternative. 1

a. Electrical duct banks beneath the onsite steam generator haul route 1

b. Intake cooling water piping beneath the onsite steam generator haul route 1

c. Electrical duct banks beneath the equipment hatch construction area for Unit 3

d. Intake cooling water piping beneath the equipment hatch construction area for Unit 4

The following handling and transporter equipment was assumed for these analyses:

a. A multi-axle trailer/dolly system would be used to transport the lower steam generator assembly along the onsite haul routes. The total transporter load assumed was 275 tons.

b. One 300-ton capacity crawler crane weighing approximately 245 tons, including crane, a 70-foot crane boom, and counter weight, would most likely be used in the equipment hatch construction area.

The actual external earth pressure due to earth cover and equipment surcharge was determined for comparison with the original Turkey Point design allowables. The results for buried piping, presented in Table 3.1-1, do not include the use of methods such as additional structural fill, timber matting or structural bridging to distribute surcharge loads. Based upon the results of these analyses, additional protection will not be required for the safety-related intake cooling water pipe.

All buried concrete electrical duct banks were analyzed as infinite unreinforced concrete beams on an elastic foundation. These analyses indicate that uniform distribution of the equipment surcharge loads by the use of timber matting or structural fill will be required to maintain concrete tensile stresses within allowable limits. Therefore, the results of the analysis for buried duct banks, presented in Table 3.1-2, are based upon typical timber matting. The duct banks under the haul route are actually 1
reinforced and will have a higher factor of safety than that presented in Table 3.1-2.

Timber matting or the equivalent will be required under crane outriggers.

In all cases, use of timber matting or other equivalent protection will 1
ensure that allowable stresses in buried duct banks and pipes are not exceeded.

3.1.1.4 Steam Generator Receipt on Site

The six steam generators have already been transported to the site by 7
barge.

3.1.1.5 Laydown Facilities Outside Containment

Adequate laydown area for construction materials and equipment is presently available at elevation +5' as shown in Figure 3.1-2. However, for the following reasons this area will be filled to elevation +17'-6" MLW:

a. All construction work will be performed at elevation +17'-6" or inside each Containment. Therefore a laydown area at elevation +17'-6" would eliminate the necessity of developing special access ramps to transport materials between two elevations. 7

b. The use of special access ramps would reduce the amount of laydown area available and reduce the mobility of handling equipment.

c. The fill will provide protection of the storage compound from hurricane storm surge.

Access to adjacent plant areas will be obtained by construction of ramps and roads as mentioned in Subsection 3.1.1.2.

3.1.2 Containment Preparation

3.1.2.1 Temporary Platform Inside Containment

A temporary platform at elevation +30'-6" inside the equipment hatch opening will be provided to handle the steam generators (Figure 3.1-1). The existing deck at this elevation will be removed. 7

3.1.2.2 Other Preparations

Concrete removal and equipment relocation of rigging clearances are discussed in Section 3.2.

3.1.2.3 Laydown Facilities Inside Containment

All fuel assemblies will be removed from the reactor and stored in the spent fuel pool, the reactor internals stored in the reactor, the reactor head stored in place on the reactor, and the control rod drive mechanism (CRDM) missile shield stored on top of the refueling cavity walls. In addition, the water will be drained from the refueling cavity. Laydown of each of the steam dome assemblies may be as follows:

a. One on the CRDM missile shield (Figure 3.1-3)

b. One on elevation +58' over an RCP motor hatch (Figure 3.1-4)

c. One in the reactor head laydown area (Figure 3.1-5)

There is adequate space inside containment for the temporary storage of the additional equipment discussed in Section 3.2.

3.1.2.4 Containment Structural Analyses

Containment structural analyses have been performed in accordance with the design criteria in Appendix 5B of the Turkey Point FSAR for the following:

a. Temporary laydown areas at elevation +58'-0" on the operating floor. These areas will be required to support the steam dome assemblies, pipe sections, and miscellaneous construction equipment. (See Figures 3.1-3 through 3.1-5.)

b. Containment base mat and existing floor support embeds in the containment wall. A temporary transfer bridge platform will be installed inside and outside the containment to facilitate the removal of the lower assemblies. Platform loads inside the containment will be transmitted to the base mat. Platform loads outside the containment will be transmitted to structural fill. (See Figure 3.1-1.)

These analyses indicate that the containment, foundation and internal structures, including the CRDM missile shield, are capable of supporting the construction loads without permanent modifications to the existing structures.

3.1.3 Transportation On-Site

Movement of the new steam generator lower assemblies (186 tons) on site can 7
be accomplished by several methods such as drop-deck or flatdeck trailer. Motive power may be rubber-tired tractor or tracked vehicle, as required.

3.1.4 Rigging Configuration

3.1.4.1 Inside Containment

The existing polar crane bridge is structurally adequate to sustain the loads imposed by a lower assembly in addition to a 250-ton capacity construction hoist weighing approximately 20 tons. Administrative controls will ensure that design bridge truck wheel loading is not exceeded during rotation by requiring that a steam generator lower assembly be moved toward the center of the bridge prior to bridge rotation.

Because of limited lift capacity, the existing polar crane trolley may not be 7
suitable for steam generator lower assembly replacement, and may, therefore, be moved aside to permit the placement of the 250-ton construction hoist on the polar crane bridge. The temporary hoist will be load tested to meet current OSHA Safety Standards prior to its use for construction lifts.

The steam dome assemblies will be parted from the lower assemblies and lifted by existing pad eyes and commercial sling assemblies and relocated to selected storage locations, as discussed in Subsection 3.1.2.3.

The lower assemblies will be lifted from their compartments using conventional hoisting techniques. The hoist lower load block will be linked by pins to a steam generator lift beam equipped with toggle arms or endless grommet type cables. The toggle arms or cables will engage existing lifting trunnions on the assemblies. Each lower assembly will be lifted and transferred in turn to a point approximately 12 feet from the containment inside wall and approximately on the centerline of the equipment hatch. Special tilting assemblies,

such as Hillman roller units and structural members, will be required to permit the assembly to move from the vertical to the horizontal position. Transfer of the lower assemblies through the equipment hatch will require the connection of additional roller assemblies as each lower assembly travels beyond the reach of the polar crane hoist.

3.1.4.2 Outside Containment

The lower assembly will exit the containment approximately 14 feet above grade. Transfer to a trailer/dolly system will be accomplished by a suit- 7
able lifting device, or as described in Appendix A, Question 6.

Shipping saddles and tie downs will be provided for secure attachment while the transport device is in transit to the storage/laydown area.

3.1.5 Rigging and Handling Controls

All lift cranes and transport devices will be located so that postulated boom pendant failure will result in a path of travel that does not adversely impact the ability to achieve and maintain safe-shutdown conditions and provide adequate cooling for stored spent fuel. In order to develop the required rigging and handling controls, it is postulated that a crane boom failure will occur in the plane of normal travel of the boom, i.e., either forward, or as a reaction rearward over the cab of the crane. Administrative controls will limit lift heights so that loads will be raised only to a height sufficient to provide clearance for horizontal movement.

Although administrative controls will be provided, subsection 5.2.1.2 discusses the analyses of postulated crane boom failures. Based on this evaluation, all structures evaluated will maintain structural integrity for the postulated 1
crane boom failure. When traversing the east plant road in the vicinity of the nuclear units, crane booms will be in the lowered position.

Travel speed and travel routes for cranes and transport devices will be controlled to minimize their influence on vital structures in the immediate area. A typical lower assembly haul route is shown in Figure 3.1-2. Controls 1
on the rotation of the polar crane when loaded are discussed in Subsection 3.1.4.1.

3.2 EQUIPMENT AND CONCRETE REMOVAL AND REPLACEMENT

Engineering evaluations, including model sequence studies, have been conducted to determine the impact of repair activities on equipment and structures in containment. This evaluation demonstrates that the repair activity will not result in any safety considerations due to equipment removal or interruption of function. It also demonstrates that there will not be any major impact on structures and equipment (non-steam generator related).

Detailed engineering studies are in progress to precisely define the components, pipes, cables, instruments, etc. within the containment affected by the repair activity. The discussion below provides the results of the study to date. It is provided to illustrate the minimal impact on non-steam generator related equipment within containment.

3.2.1 Mechanical Equipment

The following equipment which interferes with the lower assembly pathway will be temporarily removed and relocated inside the containment:

a. The fan and diffuser assemblies will be unbolted and removed from the control rod drive coolers.

b. The containment emergency air coolers (3V30A and 4V30C) nearest the containment equipment hatch opening will be unbolted, disconnected from their cooling water piping and moved clear of the pathway.

c. The Unit 3 reactor coolant pump motor 3P200A opposite the equipment hatch will be disconnected from its pump and cooling water piping and moved clear of the pathway.

As appropriate, equipment within the containment will be covered to ensure cleanliness during the repair.

Upon completion of the repair, affected equipment will be returned to service using standard procedures followed during routine plant maintenance programs.

Disconnection of power cables to the above equipment is discussed in Section 3.2.3.

3.2.2 Instrumentation

The following instrumentation, sensing lines, and associated supports will be temporarily removed and relocated inside the containment:

Unit #3

a. High and low pressure sensing lines located on the shield walls above elevation +58' for all Steam Generator "A", "B", and "C" level transmitters.

b. Instruments and sensing lines associated with the removal of reactor coolant pump motor 3P200A.

c. Instrument and sensing line support structures for items a and b above. 7

Unit #4

a. Steam generator "C" level transmitters LT-494 and 497 and supports which are mounted to the removable secondary shield wall panels opposite the equipment hatch.

b. The high and low pressure sensing lines for the level transmitters discussed in a. above.

c. The high and low pressure sensing lines located on the shield walls above elevation +58' for the remaining steam generator "A", "B" and "C" level transmitters.

d. The sensing line support structures for the sensing lines discussed in B. and c. above. 7

e. The ECCS accumulator 4T-229C, pressure transmitter PT4-931 and level transmitter LT4-930.

f. The associated sensing lines for the transmitters discussed in e. above.

The open ends of lines will be capped to ensure cleanliness during the repair.

As appropriate, the instrumentation and sensing lines will be returned to service using standard procedures followed during routine plant maintenance programs.

Disconnection of instrumentation cables to the above instruments is discussed 7
in Section 3.2.3.

3.2.3 Cable and Conduit

The steam generator repair does not require the removal and relocation of major pieces of electrical and control equipment such as panels, load centers, transformers or motor control centers. Only power and instrumentation cable and conduit as described below are affected.

Moving the mechanical equipment discussed in Section 3.2.1 will necessitate 7
disconnecting the fan motor power cables. The cable terminations will be disconnected and the cables pulled back and coiled out of the path of the lower assembly. The same cable will then be reconnected when the equipment are returned to their original location.

Table 3.2-1 is provided to illustrate Unit 3 & 4 circuits (power, control and 7
instrumentation) to be temporarily removed.

The conduits associated with these circuits will be removed, tagged, disassembled, and the associated cables pulled back and coiled. When the conduits
are later reinstalled, the cables will then be repulled and reconnected. Pro- 7
cedures will be generated for pulling back, coiling and repulling of cables, and removal and reinstallation of the conduits. Circuit checkout procedures will also be written.

3.2.4 Piping and Systems 7

In order to accomplish the steam generator repair it will be necessary to cut portions of the following systems:

a. Reactor coolant system at the steam generator channel head 7

b. Main steam piping, including small pipe vent lines

c. Main feedwater piping

d. Steam generator blowdown piping and steam generator upper dome 7

e. Cooling water return piping from the containment emergency air cooler

f. Cooling water supply and return piping to reactor coolant pump 7
motor "A" for Unit 3 only

g. Service air piping

h. Primary service water piping

Location of cut areas for reactor coolant system, main steam system, and main feedwater system are shown on Figures 3.2-1, 3.2-2, and 3.2-3, 7
respectively. As appropriate the open ends of cut piping will be covered to ensure cleanliness during the repair.

The governing overall code for the steam generator replacement shall be the ASME Section XI, 1977 Edition with addenda through the summer of 7
1978.

3.2.5 Concrete and Structural Steel

The following structures or portions of structures within the containment will be removed to provide a path for the lower assembly (refer to Figures 3.2-4 through 3.2-6 for illustration):

a. A section of the steam generator "C" shield wall above elevation + 58' for Unit 4. A section of the steam generator "A" shield wall above elevating + 58' for Unit 3.

b. A section of the operating floor concrete at elevation + 58' including a steam generator "A" upper support embed for Unit 3 and steam generator "C" thrust beam for Unit 4

c. The removable secondary shield wall panels opposite the containment equipment hatch from elevation + 30'-6" to elevation + 58', and an additional width of secondary shield wall opposite the equipment hatch from elevation + 30'-6" to elevation + 58'

d. A portion of the floor framing and grating at elevation + 58' above the equipment hatch

e. A portion of the floor framing and removable floor slabs at elevation + 30'-6" at the low point of the equipment hatch

f. The upper portion of the steel stairway near the equipment hatch opening

g. A reinforced grouted pad in the equipment hatch at elevation + 30'-6"

h. A portion of the truss system tie rods to allow for clearance of the temporary polar crane trolley. (The truss system was originally utilized in the construction of the containment and does not perform any structural related function at present.)

i. A portion of the equipment hatch sleeve at elevation +30'-6". (A small section of the steel sleeve will be replaced with thicker steel to assure load transfer to the supporting concrete wall during ingress and egress of heavy equipment. The affected portion of steel to be replaced does not form a part of the 7
containment pressure boundary nor affects the structural integrity of the containment.)

3.2.5.1 Removal of Concrete Structures

Removal of bulk volumes of containment internal structural concrete will utilize equipment and techniques commercially available. The intent is

to remove large blocks of reinforced slab/wall sections by cutting around the perimeter of such blocks and lifting them away with available cranes. Dimensions and weights of the blocks will be as large as practical. Two illustrative methods to indicate feasibility are described below.

a. High-Speed Carbide Hollow Core Drills

Drilling techniques can be used to create a parting line around a 20 to 30 ton section of wall or slab concrete. The final breakout can be accomplished using hydraulic jacks to push out the concrete section. Lifting eyes will be attached to permit control and lifting with the containment polar crane.

b. Concrete/Rebar Burning Equipment

Oxygen-fed burning lances have been utilized to effect the same type of parting line described in paragraph a. above. Fume and slag control capability is available.

As appropriate canvas or herculite dust enclosures equipped with filtered exhaust fans will be utilized to contain concrete dust generated.

3.2.5.2 Replacement of Containment Internal Structures

The containment internal structures will be restored to their original configuration. Replacement of structural materials will utilize standard construction methods in accordance with Section 5.1.6 of the FSAR. Procedures and specifications will be developed for the removal and replacement of structural materials.

3.3 RADIOLOGICAL PROTECTION PROGRAM

The radiological protection program to be implemented for the repair effort will be in accordance with the FPL Health Physics Manual and its implementing procedures. This program is designed to be responsive to the Regulatory Staff Positions in Regulatory Guides 1.8, 8.2, 8.8, and 8.10 and is in operation at Turkey Point.

3.3.1 <u>Supplemental Access Control</u>

Additional facilities will be provided for the repair effort to accommodate the personnel involved. These facilities include:

a. Outside Radiation Area

1. Craft change area
2. Radiological protection training facility
3. Locker area

b. Inside Radiation Area

1. Radiation control point
2. Protective clothing pickup area

3. Protective clothing dressout area
4. Protective clothing undressing area
5. Storage area for protective clothing
6. Toilet
7. Health physics area
8. Laundry

The following is a brief description of the access control pathway currently contemplated for entering and exiting the containment:

> Personnel will enter the locker area, disrobe, and, after proceeding through the radiation control point, pick up their protective clothing in a connecting area and dress in an adjacent area before entering the equipment hatch into the containment. The requirements for protective clothing are specified in the FPL Health Physics Manual and in FPL Protective Clothing Procedure HP-50. Personnel leaving the containment will remove their rubber shoe covers and rubber gloves at an access control point just inside the equipment hatch before stepping onto the step-off pad. These personnel will immediately proceed to the undressing area to undress and be frisked for residual contamination. They will then exit through the radiation control point and return to the locker area for their street clothes. Treatment of contaminated personnel will follow the procedures given in the FPL Health Physics Manual and FPL procedures HP-70 and HP-71.

Additional health physics support will be provided to the FPL site health physics organization in order to implement health physics related activities. Access control at the steam generator compartments, equipment hatch, and at the temporary access control area discussed above will be provided.

Personnel involved in work areas with a potential for high-level contamination will wear two sets of protective clothing. The outer set of protective clothing will be removed when leaving the work area and deposited in a container. The second set will be removed in an area outside containment, as described above.

3.3.2 Laundry

In order to accommodate the laundry expected during the repair effort, additional laundry capability will be provided. Laundering of protective clothing and cleaning and sanitizing of respiratory equipment will be in accordance with the FPL Health Physics Manual.

3.3.3 Control of Airborne Radioactivity and Surface Contamination

Airborne radioactivity inside containment during the steam generator repair
effort will be controlled, monitored and ultimately released via the plant
vent stack. A slightly negative pressure inside of containment will be main-
tained using the containment purge exhaust system. Air will be drawn through
the equipment, personnel and emergency hatches and exhausted by the purge
system via the plant vent, thus precluding airborne radioactive particles 1
or gases from leaving containment openings utilized for construction activi-
ties. Air flow requirements necessary for maintaining a slightly negative
containment pressure are well within the existing purge exhaust system
capacity. The air being exhausted will be monitored as it passes the
existing sampling station located within the main plant vent.

In addition to bulk containment atmosphere control of airborne radioactivity,
appropriate localized control will also be provided. Radioactivity generated
during decontamination and cutting of the reactor coolant system will be 7
contained within specially designed contamination control envelopes, which
will provide local high efficiency filtration. Personnel working inside these
control envelopes will wear respiratory protection equipment, as required,
described and implemented by FPL procedures HP-60 through HP-69, in accord-
ance with plant Technical Specifications and 10 CFR 20.103.

No special provisions are required for other cutting operations inside
containment.

Section 3.3.1 describes the method of controlling the spread of surface
contamination by personnel removing their outer set of protective clothing
when leaving the control envelope.

The radioactive release and dose assessment associated with decontamination
and cutting the reactor coolant system is provided in Subsections 5.2.2.1 7
and 5.2.2.2.

3.3.4 Supplemental Personnel Monitoring Requirements

3.3.4.1 Monitoring of Airborne Radioactivity

Portable air monitors will be used, as required, to monitor the airborne 7
radioactivity inside the control envelopes and in other work areas inside
containment. Airborne radioactivity samplers coupled with laboratory
analyses will also be employed.

3.3.4.2 Monitoring of Workers for Ingested Radioactivity

All workers who are planned to enter a high radiation area will be given an
initial bioassay at the start of their employment. Subsequently, workers 7
will be given bioassays, as necessary, to comply with requirements set forth
in the FPL Health Physics Manual and FPL procedure HP-31.

3.3.4.3 Personnel Monitoring

All personnel entering the radiation controlled area will be provided with
TLDs and self reading pocket dosemeters in accordance with the FPL Health 7
Physics Manual and FPL procedure HP-30.

3.3.4.4 Radiation and Contamination Surveys

Detailed surveys which provide proper control of radiation and contamination will be performed, as required, throughout the repair effort. These surveys will be performed in accordance with FPL procedures HP-20, HP-21, and HP-22.

3.3.4.5 Portable Survey Instruments

Table 3.3-1 provides a typical listing of the types of portable survey instruments available for use during the repair effort.

3.3.5 General ALARA Considerations

Personnel exposures will be maintained as low as is reasonably achievable (ALARA) in accordance with 10 CFR 20.1(c) and the guidance provided by Regulatory Guide 8.8 as defined in:

a. The FPL Health Physics Manual

b. FPL radiation protection program implementation procedures

c. The additional guidelines given below

3.3.5.1 Model of the Turkey Point Containment

The containment model, as described in Section 1.1.1, is a valuable tool in determining occupational radiation exposure and studying methods for reducing doses. Among the more important considerations are:

a. Shielding

The model, in conjunction with actual field survey data, will be used to study radiation fields to determine temporary shielding requirements.

b. Man-Rem Assessment

The model aids in predicting the expected man-rem doses for activities in high radiation areas. Decisions related to radiation exposure, such as employing local decontamination or determining the number of people required for an activity, can be made early in the design phase of the project in order to incorporate the most effective solutions to the reduction of exposures.

c. Work Planning

The model will be used to develop construction work plans to establish the most efficient procedures for performing work in high radiation areas.

d. Craft Training

The model will be used for the orientation and training of supervisory and key craft personnel to supplement construction work plans, to achieve the most efficient utilization

of manpower in high radiation areas, and to reduce overall man-rem doses by minimizing the time spent in these high radiation areas.

3.3.5.2 Temporary Shielding

Shielding will be used, as necessary, to reduce the dose rates from other components such as the regenerative heat exchanger, RHR system valves, and temporary storage areas for contaminated pieces of pipe, rags, and tools. Temporary shielding will be used, as necessary, for the steam generator while it is being cut out of the reactor coolant loop and moved out of the containment. The steam generator shell will also help shield the more contaminated parts of the steam generators.

3.3.5.3 Decontamination

3.3.5.3.1 General Area Decontamination

To lessen the total man-rem accumulated for a repair such as this, a general decontamination of the containment is planned during the initial phases of the steam generator repair. This operation consists of cleaning most of the exposed surfaces of the containment in the areas where work tasks in support of the steam generator repair are scheduled. The removal of a major portion of the slightly radioactive surface dust and dirt from such areas has two beneficial effects: 1) the overall background dose rate is lowered in those areas due to removal of a significant contributing source, hence there is less man-rem dose to the work force during the remainder of the outage and, 2) the spread of surface contamination is significantly reduced, so clean areas remain clean and the incidence of contamination of workers in such areas is greatly lessened. 7

3.3.5.3.2 Primary Surface Contamination

In the channel cut approach some decontamination of the channel head region of the steam generators would be advantageous in maintaining exposures to a minimum. The interior surface of the channel head and divider plate will probably be decontaminated by remote means prior to the final parting cut that will separate the tubesheet/tube bundle and lower shell (midsection) from the channel head. The only operations necessary inside the channel head prior to the parting cut are those in support of cutting the divider plate and the placing of appropriate blocking devices in the coolant pipe. The man-rem expended in the decontamination effort will be balanced against the potential man-rem savings incurred during these cutting operations.

Once this midsection has been removed, then further decontamination operations will be performed on the channel head/coolant pipe interior surfaces, as determined by the man-rem savings possible for the subsequent operations (fitup and welding) compared to the man-rem expended in the decontamination operations.

Several different decontamination procedures are currently available for this primary surface decontamination. These fall into two main groups:

Mechanical: A technique that involves an abrasive grit blast of the surface to remove the surface layer that contains the radioactive contamination. The primary surfaces are radioactive primarily due to this surface layer; industry experience has shown that the metal itself is at most, slightly radioactive. Removal of the layer by abrasive action removes the radioactivity. The waste material from this process could be processed as appropriate and drummed for off-site disposal.

Electo-Polishing: This method involves a process that is best described as reverse electro plating. The cavity to be decontaminated is filled with an electrolyte and an electrode is placed some distance from the surface. As current flows through this setup it causes the surface contaminants on the primary surface to "plate off" of the contaminated surface into electrolyte solution or onto the electrode. After a period of time this surface layer is removed, the electrolyte could be solidified and drummed for off-site disposal along with the electrode. The bowl-shape of a channel head lends itself to such an operation. 7

FPL will continue to evaluate which method or combination of methods will lead to the most effective man-rem utilization in the channel cut approach.

3.3.5.4 Low Background Radiation Waiting Areas

Low background radiation waiting areas will be established where workers may wait between tasks. Special signs will be posted to designate these areas. Signs will also be posted per HP procedures. 7

Health Physics will work with the job supervisors to assure that personnel not required in the work area remain in the waiting area.

3.3.5.5 Training of Craft Personnel

As appropriate, craft personnel will be given the comprehensive course in radiological protection described in the FPL Health Physics Manual and FPL Procedure HP-81. This course will consist of approximately 20 hours of instruction and demonstrations covering in detail the basic theory and practice of radiation protection principles, emergency planning, and the Radiological Protection Program. Successful completion of this course and an associated comprehensive examination will allow personnel to have unescorted access to the radiation controlled area. 7

Craft personnel unable to pass the examination and/or those who take only the orientation course, which consists of one 3-hour class to acquaint individuals with basic safe health physics practices and emergency procedures, will be required to be escorted in the radiation controlled area.

3.3.6 Miscellaneous Waste Disposal

3.3.6.1 Concrete Disposal

Approximately 60 cubic yards of concrete per unit will be removed from the containment internal walls and floors and will be disposed of. The majority of this concrete has an insignificant amount of transferable contamination (transferable contamination is considered insignificant if it is less than 2200 dpm/100 cm^2 per 49 CFR 173.397) without surface decontamination. The concrete which is considered contaminated, (i) may be decontaminated prior to cutting by vacuuming and/or scrubbing with detergent and water to reduce the amount of transferable contamination to as low as is reasonably achievable or, (ii) appropriately packaged for shipment. Following removal from the containment, the concrete will be shipped as "low specific activity" (LSA) material to a licensed land burial site.

3.3.6.2 Miscellaneous Dry Waste Disposal

Metal shavings from the various cutting operations and miscellaneous dry waste, such as paper, rags, etc., will be put in standard shipping containers and shipped as LSA material to a licensed land burial site.

3.3.6.3 Liquid Radwaste Disposal

There are three potential sources of radioactive liquid to be disposed of. These sources are:

a. Water drained from the reactor coolant system

b. Laundry waste water

c. Local decontamination waste fluids

The radioactive releases associated with these sources are discussed in Subsection 5.2.2.4

It is planned that the reactor coolant will be stored on-site for re-use after the repair. However, should it be necessary to release any of the coolant, it will be processed by the chemical and volume control system, as described in Section 9.2 of the Turkey Point FSAR. 7

The laundry waste water may be discharged without processing due to the low activity level as indicated by the estimated laundry waste water specific activities given in Table 5.2-5. If sampling indicates processing is required, the laundry waste water will be treated as part of the normal liquid radwaste processing scheme.

The small amount of liquid waste generated as a result of local decontamination will be treated as part of the normal liquid radwaste processing scheme.

3.3.7 Man-Rem Assessment

3.3.7.1 Man-Rem Assessment for Continuing Operation

A considerable amount of radiation exposure to personnel is associated with inspecting and plugging steam generator tubes. The exposure to 1

personnel over the remaining operating lifetime of the plants can be reduced by the repair effort. This reduction is consistent with the guidance provided by Regulatory Guide 8.8 and ALARA.

For 1978-1979, the amount of radiation exposure at Turkey Point Units 3 and 4 attributable to steam generator tube plugging, eddy current testing, 7 and other inspection related activities was approximately 335 man-rem per year. The eddy current testing and tube plugging techniques employed at Turkey Point are state-of-the-art; therefore, the man-rem attributed to these maintenance/inspection activities is ALARA. Current methods (1976 to date) employ a single position eddy current fixture and explosive plugging.

Eddy current testing techniques have been continually improved in terms of man-rem exposures. Attainment of the present level of design has taken several years with many revisions and improvements to the system. The current tool for conducting eddy current examinations is a single-position fixture. The technique requires only one entry for installation and one entry for removal on each side of the channel head; hence, a total of four entries are required to complete a steam generator inspection. The previously used 1 orthogonal fixture technique required one pre-installation entry, four entries for installation/repositioning, and one entry for removal on each side of the channel head; hence twelve entries were required with the orthogonal fixture technique to complete a steam generator inspection. In addition to the higher number of entries, the orthogonal fixture technique also requires more personnel support activities at the channel head manways than the single fixture technique. These considerations, coupled with the higher mechanical reliability of the single fixture technique, result in an estimated 60 to 75 percent reduction in eddy current testing personnel exposures by using the state-of-the-art single fixture technique at Turkey Point instead of the orthogonal fixture technique.

Explosive plugging of steam generator tubes was developed in the early 1970s and continues to be the most economical approach to plugging in relation to total exposure. Hand welding of plugs in a steam generator would require approximately 15 minutes of steam generator occupancy for each plug, whereas, three explosive plugs can be set and fired with less than 1 minute occupancy time. Hence, if three tubes are to be plugged, this would result in approximately 1/45 the total radiation exposure for an equivalent conventional (manually-welded) plugging operation.

Assuming that the replacement steam generator tubes maintain their integrity during the remaining operating lifetime of the plants, radiation exposure attributed to steam generator work will be reduced. It is not expected to exceed approximately 100 man-rem per year for a tube inspection and plugging 1 operation in accordance with Regulatory Guide 1.83. (This is considered to be a conservative estimate, since it is expected that these inspections will not necessitate tube plugging.) Therefore, approximately 235 man-rem will 7 be saved each year following steam generator repair of both units. 1

3.3.7.2 Man-Rem Assessment for the Repair Effort

In order to assess the man-rem associated with the repair, detailed surveys were made of the radiation levels at the exterior of the steam generators

and surrounding areas with the reactor shut down, and the reactor coolant system drained. The surveys were taken one month after shutdown, which is a conservative estimate for the repair operation since most of the high radiation work will be done more than one month after reactor shutdown. Typical radiation survey results for both Units 3 and 4 are presented in Figures 3.3-1 through 3.3-7. Radiation surveys were taken with a teletector in June 1977 with the steam generators drained. These surveys were then compared with previous area surveys, taken approximately 1 month after shutdown 1
in accordance with the FPL Health Physics Manual and FPL procedures HP-20, HP-21 and HP-22. It was determined that plant survey data is representative of conditions expected during the repair activity. This data was used for the man-rem assessment presented below.

A range is given for the man-rem associated with removal and replacement of the steam generators and with the alternatives associated with the final disposition of the steam generators. The man-rem predictions reflect the best estimates that can be made prior to completion of the detailed engineering phase of the job. The range is given to span the uncertainties associated with these man-rem predictions. It is expected that actual man-rem incurred will fall within the man-rem range estimates provided herein.

Uncertainties in the prediction of man-rem include:

a. Job man-hour uncertainties:

   1. At this time it is difficult to define the precise scope of each task. 2

   2. The man-hour estimate for performing any task is subject to variances resulting from labor productivity, working conditions, effects of other simultaneous activities and accessibility of the work area.

   3. Task man-hours associated with the actual work performed in radiation areas versus those associated with time spent outside the radiation areas cannot be precisely defined.

   4. The assumptions made for splitting the man-hours worked in radiation areas into a range of radiation fields can only be done on a job average basis.

b. Radiation field uncertainties:

   1. Radiation fields were taken from actual Turkey Point surveys, as discussed above. In developing the man-rem predictions it was assumed that the radiation fields would remain constant throughout the repair effort. Actual radiation fields will decrease with time; therefore, the actual total job man-rem are expected to be lower than the calculated values.

2. The effectiveness of temporary shielding, the time required to place such shielding and the type of remote cutting and welding techniques utilized are not precisely defined prior to completion of detailed engineering.

3. The radiation fields used in the man-rem predictions are task average. The exact radiation fields to which an individual is exposed during the work effort cannot be precisely determined, because workers associated with a specific task may not only move around within the task area, where radiation fields could vary significantly with location relative to the sources of radiation, but 2 may also move around outside of the task area to perform task-supportive work.

4. The decontamination factor (DF) for the channel head decon will vary depending on the method utilized. However, the DF 7 used in the analysis should readily be attained by whichever method or methods are used.

Supporting calculations have been performed to indicate the approximate man-rem expected for each task during the repair, and during ultimate disposition, as discussed in Section 3.4.4. The range for the total effort represents the best judgment with respect to the predicted worker dose, 7 considering the uncertainties noted above. Total man-rem are not expected to exceed the upper limit of the ranges for repair and disposition of the existing steam generators. 2

The tasks considered in the man-rem assessment of the removal and installation of the lower assemblies are presented in Table 3.3-2. 7

Based on the information presented above, it is estimated that approximately 1730 to 2480 man-rem per unit could be incurred during the removal and installation of the steam generator lower assemblies and other miscellaneous 7
repair activities. The details of these estimates are given in Table 3.3-2.

Worker exposure to airborne radioactivity was also considered in the overall evaluation of radiation exposure for the repair effort. During the repair effort, the normal contributors to airborne activity inside containment, the reactor coolant system and the refueling cavity, will provide no contribution since both will be drained of radioactive liquids. In addition, as described in Section 3.3.3, airborne radioactivity inside containment will be controlled by operating the containment purge system and radioactivity generated during the cutting and decontamination of the steam generator 7
channel head internal surfaces will be contained within specially designed contamination control envelopes, which will provide high efficiency filtration. Therefore, the airborne radioactivity concentrations to which workers will be exposed for most of the tasks performed during the repair are expected to be insignificant.

Two specific categories of tasks for which protection against airborne activity is required, are discussed below:

a. Airborne radioactivity associated with the cutting and decontamination of the steam generator channel head internal surfaces is considered as an upper bound for all tasks to be performed during the repair effort. Therefore, for the purpose of analysis, it was assumed that the channel head internal surfaces were decontaminated by some method (as described in Section 7
3.3.5.3.2) which yielded a DF of 12. Then, the steam generator midsection was cut and removed from the channel head. Airborne radioactivity concentrations were then estimated assuming that all of the activity remaining on the channel head and divider plate surfaces of one steam generator, after this initial decontamination, became airborne due to further decontamination and cutting. The analysis showed that workers inside the contamination control envelopes would be exposed to inhalation airborne radioactivity concentrations which are approximately 0.6 MPC, based on a 40 hr/wk (2000 hrs/yr) MPC for airborne concentrations listed in 10 CFR 20, Appendix B, Table 1, Column 1. The evaluation considered a protection factor associated with the workers wearing air line, full-face, pressure-demand respirators. For those tasks being performed inside containment, but outside the contamination control envelopes, during the cutting and decontamination of the steam generator primary side surfaces, worker exposure will be negligible because exhaust from the contamination control envelopes will be routed, through filters, directly to the containment purge system exhaust.

b. It is also expected that during the removal of contaminated nonmetallic insulation there may be localized airborne activity. However, FPL health physics procedures require workers removing such insulation to wear respirators as a precautionary measure;

therefore, the airborne radioactivity to which these workers will be exposed is expected to be insignificant. Furthermore, experience at the plant indicates that containment airborne radioactivity does not show a measurable increase, except locally, during insulation removal.

As discussed in Section 7.5, repair of the steam generators can be accomplished in one of three alternative ways, defined therein as Schemes I, II and III. In Schemes II and III, the steam generators may be removed and replaced in one piece. Thus, the man-rem doses associated with Schemes II and III would be similar to those given for Scheme I, except that the man-rem doses associated with the tasks related to the cutting of the steam generator at the transition cone do not apply. It is estimated that these tasks could result in a radiation exposure of approximately 140 man-rem for three steam generators; therefore, the range in man-rem dose associated with the removal/replacement effort for Schemes II and III is approximately 140 man-rem less than the total for Scheme I. 7

There are other tasks which vary between schemes, such as:

a. Scheme I - removal and replacement of concrete shield wall.

b. Scheme II - removal and replacement of a section of the containment wall.

c. Scheme III -- removal and replacement of a section of the containment dome.

However, these tasks are of considerably less significance from a man-rem dose standpoint than those associated with the steam generator transition cone cut. Therefore, the man-rem dose given above for Schemes II and III provides a reasonable estimate of total exposure.

3.3.7.3 Conclusions

In addition to meeting the quarterly, annual and cumulative occupational doses specified in 10 CFR 20.101 for any individual associated with the repair, the man-rem assessments presented in Subsection 3.3.7.1 and 3.3.7.2 were evaluated in order to determine whether or not the radiation received by personnel during the repair is ALARA. Although the man-rem associated with the repair effort are greater than the man-rem associated with the 1978-1979 tube plugging, eddy current testing and other inspection-related activities at Turkey Point, it has been determined that approximately 235 man-rem will be saved per year by implementing the repair. Thus after 9 years the man-rem savings will exceed the man-rem expended to perform the repair. Therefore, implementation of the repair is in accordance with Regulatory Guide 8.8 (ALARA) guidelines. Man-rem doses associated with the optimum disposal alternative are not appreciable (see 7

Section 3.4.4). Thus, disposal considerations do not alter this conclusion. The primary basis of Regulatory Guide 8.8 is the conservative assumption that a mon-threshold linear relationship exists between dose and biological effects, independent of the dose rate. On this basis, the projected long 1 term man-rem savings gained from the repair effort will result in a linear reduction in any biological risk that is assumed to be proportional to the dose.

3.4 DISPOSITION OF STEAM GENERATOR LOWER ASSEMBLIES

The lower assemblies to be removed from Units 3 and 4 represent the single largest source of solid radioactive waste to be disposed of during the repair effort. The disposal effort is independent of the repair and is evaluated on that basis.

The primary side surfaces of the steam generators are contaminated by a tenacious film of deposited radioactive corrosion products made up primarily of iron, cobalt and manganese isotopes. Based on actual Turkey Point data provided in Section 5.2.2, it is estimated that at the time the lower assemblies are removed, each will contain approximately 250 curies of deposited gamma activity. 3

3.4.1 Objectives of Handling/Disposal Operations

The objectives of handling/disposal operations are as follows:

a. To dispose of the lower assemblies safely and economically

b. To provide means to handle/dispose of the steam generator lower assemblies so that radiation exposures to plant and contract personnel are as low as is reasonably achievable

c. To minimize the release of radioactivity to the environment so as to keep radiation exposure to the public as low as is reasonably achievable and within the limitations of 10 CFR 20

d. To package and ship the lower assemblies to a licensed land burial site in accordance with applicable state and federal regulations, including 10 CFR 71, 46 CFR 148, and 49 CFR 170-178, or to store the lower assemblies onsite for decommissioning with the plant. 1 7

3.4.2 Onsite Storage

A temporary onsite storage area will be provided for the storage of the lower assemblies. The lower assemblies will be stored in this area until they can be shipped offsite to a licensed land burial site or decommissioned with the plant. Prior to removal from the containment, the openings in the lower assemblies will be sealed to prevent the release of radioactivity during transfer and subsequent onsite storage.

As discussed in Section 3.4.6, the only radiological consideration associated with storage is the direct radiation from the steam generators. Shielding will be provided to ensure acceptable radiation levels external to the storage facility. Section 3.4.7 demonstrates that there are no accident considerations associated with onsite storage.

Based on the above considerations, the required storage facility design criteria are:

a. Appropriate shielding for direct dose

b. Provisions for periodic surveillance of steam generator seal integrity 1

c. Total enclosure of the sealed steam generators is not required.

Should lower assembly cutting and packaging operations be conducted, temporary filtered ventilation systems and enclosure envelopes will be provided, as required, to control airborne radioactivity which may be generated. Appropriate provisions to monitor the release of activity resulting from these operations will also be provided.

The need for a temporary storage facility is based on immediate shipment of the lower assemblies not being possible.

3.4.3 Offsite Disposal

Disposal of the steam generator lower assemblies at a licensed land burial site was investigated. Rail, truck and barge were considered viable alternatives for shipment of the lower assemblies offsite.

3.4.3.1 Preparation for Shipment by Truck and/or Rail

In preparation for shipment of the lower assemblies to a licensed land burial site, they will be cut into sections suitably sized for shipment by rail and/or truck. Based on the expected curie content of the lower assemblies, the cut-up sections will then be packaged in strong, tight packages and shipped as "low specific activity" (LSA) material in accordance with applicable state and federal regulations.

Cutting operations on the lower assemblies will be performed in enclosure envelopes, as required, to minimize the spread of airborne activity. The

enclosure envelopes will be provided with a HEPA filtration system to reduce the potential release of radioactivity to the environment and will be designed to allow the use of remote cutting techniques to reduce personnel exposure to radiation during cutting. Temporary shielding will also be provided, as required, to further reduce personnel radiation exposure. Radiation detection and measurement during cutting operations will be in accordance with the FPL Health Physics Manual.

Although it is expected that radiation exposure to personnel will be relatively low during the cutting operations because of remote cutting techniques and temporary shielding, additional man-rem reduction techniques have also been investigated. These techniques, which could be implemented in the temporary storage facility, include: a) decontaminating the primary side surfaces or b) filling the primary side with a solidification agent.

Both decontamination and solidification agent injection are viable methods for reducing radiation exposure to personnel during cutting of the lower assemblies. Decontamination may be mechanical or chemical, although chemical decontamination appears to be more favorable in reducing radiation

exposure to personnel since much of the expected radiation exposure is associated with the cutting up and packaging of the lower assemblies. If chemical decontamination is used, the liquid waste generated will be processed by one or more of the following: a) a mobile radwaste evaporator package, b) a mobile filter/demineralizer system, c) a mobile solidification unit. These mobile units are commercially available and have been used at several nuclear power plants. The solidification agent injection method involves filling the primary side of the steam generators with a suitable solidification agent prior to cutting. The solidification agent is used to fix any loose contamination in the primary side of the steam generators and to provide shielding to further reduce the radiation dose rates during the cutting operations.

The expected radiation exposure in man-rem associated with shipment preparation of the lower assemblies is presented in Section 3.4.4.

3.4.3.2 Preparation for Shipment by Barge

The steam generator lower assemblies will be sealed prior to removal from
the containment so that the radioactivity will be contained within a strong,
tight package, as required by 49 CFR 173. When the lower assemblies are
to be shipped to a licensed land burial site, each one will be placed into
a licensed type A cask, transported to the barge facility, and shipped as
LSA material, in accordance with applicable state and federal regulations.
The cask associated with the shipment of the generators will meet all DOT
transportation regulations and will reduce radiation levels to less than
10 mR/hr at 6' from the accessable surface of the package. Temporary
shielding will also be utilized, as required, to further reduce personnel
radiation exposure. Evey effort will be made to maintain exposures to 7
levels that are as low as reasonably achievable consistent with job
performance.

3.4.3.3 Shipment

Rail, truck, and barge transportation were investigated for shipment of the lower assemblies to a licensed land burial site.

Should rail transport of the lower assemblies be employed, each lower
assembly will be cut up and packaged in approximately seven packages.
The tubesheet of each lower assembly will make up one package, the tubes 7
will make up approximately four packages, and the shell and other miscellaneous pieces will require another two packages. The heaviest load
will be the tubesheet. In addition, combination rail/truck shipments may
be utilized; that is, rail would be used for the tubesheet and truck would 7
be used for the other pieces. If rail is utilized, rubber-tired tractors
or tracked vehicles are required to transport the packages from the jobsite to a rail spur near Florida City, Florida (approximately 8 miles)
and from the rail spur near the burial site at Barnwell, South Carolina
(approximately 3 miles).

Should truck transport of the lower assemblies be employed, methods similar to those described above for rail transport will be used, except that the number of packages would have to be increased in order to meet the weight limitations for over-the-road shipments.

Should barge transport of the lower assemblies be employed, upon removal of the steam generator from the reactor containment building, it will be transferred to a temporary laydown area. The steam generators will be taken one at a time from this temporary laydown area, placed in a type A cask, and secured to a special transport vehicle. The cask and steam generator will then proceed to the barge slip by a route that has been utilized previously for heavy equipment transport. The special transport vehicle, cask, and steam generator will be driven directly onto the ocean going barge. The vehicle and cask assembly will be tightly secured to the barge. The ocean going barge and steam generator will then be taken from Turkey Point through Biscayne Bay to the ocean where it will be transported to the Savannah River. The barge will be brought up the Savannah River to a prepared landing. The vehicle and cask assembly will be driven off the barge with subsequent over-the-road transportation on an approved and prepared route to a low level waste disposal facility. The steam generator will be removed from the shielded cask and placed in a trench for ultimate interment in accordance with applicable state and federal regulations.

Regardless of the transportation method used, applicable requirements of 10 CFR 71, 46 CFR 148, and 49 CFR 170-178 will be met. 7

3.4.4 Man-Rem Assessments

The evaluations presented in this section are based on the steam generator lower assembly configuration which results if the repairs were conducted by means of the pipe cut approach. For the channel cut repair approach, the resulting lower assemblies would be smaller and contain fewer curies of activity, thus requiring fewer cuts and less handling during disposition. Hence, man-rem and radioactive release effects resulting from the disposition of the lower assemblies removed by the channel cut repair approach are bounded by the discussions below.

If the steam generator lower assemblies are shipped by rail and/or truck, they must be cut into suitably sized sections prior to shipment. Based on the radiation survey results given in Figure 3.3-7, it is estimated that the following man-rem doses per unit will be received during the preparation and shipment of a unit's lower assemblies for each of the given alternatives:

a.	Cutting up, packaging and shipping of the lower assemblies, assuming no decontamination and no solidification agent injection	750-1525 (man-rem)	
b.	Cutting up, packaging and shipping of the lower assemblies, assuming solidification agent injection prior to cutting and a dose reduction factor of 2 is achieved	400-825 (man-rem)	
c.	Cutting up, packaging and shipping of the lower assemblies, assuming decontamination of primary side surfaces prior to cutting and a DF of 10 is achieved	125-575 (man-rem)	
d.	Cutting up, packaging and shipping of the lower assemblies, assuming 35 years storage prior to cutting	35-45 (man-rem)	
e.	Placing in cask, securing, and shipping of the lower assemblies, via barge, assuming no decontamination and no solidification agent injection	26.5-28 (man-rem)	7
f.	On-site surveillance of 35 years and decommissioning of the steam generator lower assemblies	25.5-26.5 (man-rem)	

The man-rem dose to workers by task for each of the above alternatives is contained in Table 3.4-3. A discussion of the uncertainties in predicting worker man-rem ranges for the alternatives is contained in Subsection 3.3.7.2. The man-rem dose to the public for each alternative is contained in Table 3.4-4.

7

It should be recognized that although the alternative which includes decontamination of the lower assemblies appears attractive from a man-rem savings standpoint, there is a significant cost penalty associated with it. Thus, the optimum cost/man-rem disposal alternative may not incorporate decontamination.

The man-rem dose associated with disposal of the steam generators is considered acceptable since a considerable man-rem savings is achieved by the repair effort (see Section 3.3.7.3).

3.4.5 Radioactive Releases and Doses Assessment Associated With Offsite Disposal 1

The evaluations presented in this section are based on the steam generator lower assembly configuration which results if the repair were conducted by means of the pipe cut approach. For the channel cut repair approach, 7 the resulting lower assemblies would be smaller and contain less activity, thus requiring fewer cuts and less handling during disposition. Hence, the radioactive releases resulting from the disposition of the lower assemblies removed by the channel cut repair approach are bounded by the discussions below.

Radioactive airborne and liquid releases have been evaluated for the disposal effort using conservative, bounding parameters and assumptions. The total calculated release per unit for the disposal effort was found to be a small fraction of the period July 1978 through June 1979 radioactive 7 releases per unit at Turkey Point and, therefore, is considered acceptable.

3.4.5.1 Airborne Releases

Radioactive airborne effluent releases to the environment resulting from the disposal effort have been estimated using the following assumptions and parameters:

a. Airborne releases are assumed to occur during the cutting operations, i.e., the cutting of the tubesheets, channel heads and tube bundles.

b. The primary side surfaces of the lower assemblies are expected to be contaminated primarily by deposited corrosion products. Typical corrosion product activities expected on the primary side surfaces after nine years of commercial operation are 7
given in Table 5.2-1. (Nine years is based on the projected operation of Unit 3 prior to the repair effort. For Unit 4, the deposited corrosion product activity is bounded since Unit 4 will have less operating time, a projection of seven years.)

c. It is conservatively assumed that all the activity present in the vicinity of the cut will become airborne and be available for release to the environment.

d. Ninety days of radioisotope decay were assumed prior to cutting operations, based on the earliest reasonable time as dictated 1
by the repair effort. No credit was taken for radioisotope decay during cutting operations or for usage of water sprays.

e. The lower assemblies are assumed to be cut up in a facility which will be provided with HEPA filters in the exhaust line. The HEPA filters are conservatively assumed to be 99 percent efficient for capturing particulates.

Radioactive airborne effluent release to the environment based on the above assumptions is approximately 3.49×10^{-2} Ci for Unit 3. For Unit 4, this release is bounded by the release for Unit 3. Details of the airborne 7
effluent release by isotopes are given in Table 3.4-1. There are no radioactive airborne releases associated with onsite steam generator lower 1
assembly storage, see Section 3.4.6.

3.4.5.2 Environmental Consequences of Airborne Releases

The critical organ (lung) and whole body doses for a teenager (critical age group) at the worst site boundary location resulting from the estimated airborne effluent releases during the disposal effort were evaluated in the same 7
manner as the repair effort described in Subsection 5.2.2.2. The critical organ and the whole body doses at the site boundary are estimated to be 6.18×10^{-3} mrem and 8.50×10^{-6} mrem, respectively during the disposal effort for Unit 3. The corresponding doses for Unit 4 would be bounded.

3.4.5.3 Comparison with Observed Gaseous Releases and Estimated Doses During Normal Operation

The estimated releases of radioactive airborne effluents per unit during the disposal effort are found to be much smaller than the observed gaseous effluent releases per unit for the Turkey Point Plant during the period 7
July 1978 through June 1979. Observed airborne effluent releases during 1978-1979 are compared with estimated releases during the disposal effort in Table 3.4-2.

The critical organ dose (thyroid) and whole body doses to a child (critical age group) at the worst site boundary location due to the release of gaseous effluents for 1978-1979 were calculated to be 0.02 and 0.09 mrem/unit, 7
respectively. The estimated critical organ dose for the disposal effort is less than 33 percent of the calculated critical organ dose for 1978-1979.

The estimated whole body dose for the repair effort is less than 0.01 percent 7
of the calculated whole body dose during 1978-1979.

3.4.5.4 Liquid Effluent Releases

The only liquids associated with the disposal of the lower assemblies are
those associated with decontamination. If the lower assembly primary side
surfaces are decontaminated prior to cutting and the liquid wastes resulting
from the decontamination operation are processed through a radwaste evaporator
and subsequently discharged, the total radioactive liquid effluent release
would be approximately 0.06 Ci/unit. (Note that other methods of decontam- 7
ination waste processing may be used which may slightly alter these results.
These methods include solidification of the chemical solvents, filter/demin-
eralization of the rinses, etc.)

3.4.6 <u>Radioactive Releases and Dose Assessment Associated With Onsite Storage</u>

The evaluations presented in this section are based on the steam generator
lower assembly configuration which results if the repair were conducted by
means of the pipe cut approach. For the channel cut repair approach, the
resulting lower assemblies would be smaller and contain fewer curies of 7
activity. Hence, the direct radiation doses resulting from the storage of
the lower assemblies removed by the channel cut repair approach are bounded
by the discussions below.

The annual dose equivalent to any member of the public as a result of onsite storage of the used steam generator lower assemblies was evaluated. The evaluation considered a member of the public at the north site boundary for a period of 1 year.

As indicated in Section 3.4.2, prior to removal from the containment, the
openings in the steam generator lower assemblies will be sealed to prevent
the release of radioactivity during transfer and subsequent onsite storage.
Since the lower assemblies will be completely sealed, there will be no air- 1
borne or liquid radioactive releases as a result of lower assembly onsite
storage.

As discussed in Section 3.4.7, the radioactivity within the steam generators is immobile. Thus, if seal integrity were lost, releases to the environment are not likely. Nonetheless, a surveillance program will be implemented, comprised of a quarterly visual inspection of the external surfaces of the lower assemblies, area radiation surveys and random swipes of the welds sealing the covered openings in the lower assemblies. This surveillance program will provide further assurance that there are no unanticipated releases of radioactivity to the environment.

The only contribution, therefore, to the annual dose equivalent to any member of the public is from direct radiation emanating from the storage facility. The storage facility will be shielded, as required, in order to limit the dose rate at the outside limits of the storage facility to 2.5 mr/hr. The resulting dose equivalent to an individual at the north site boundary for a full year would be approximately 5.2×10^{-3} mrem, which is considered an insignificant contribution to the offsite dose. Furthermore, it is highly unlikely that

an individual would be continuously exposed for a period of 1 year at the north site boundary; therefore, the actual annual dose equivalent to any individual at this location will be lower than that given above.

3.4.7 Accident Considerations Associated with Onsite Storage

The only potential concern associated with steam generator lower assembly storage is the release of radioactivity to the environment. The majority of this radioactivity is on the primary side surfaces of the lower assembly in the form of a protective corrosive film of metal oxides which is very adherent and very refractory. (Radioactivity would be present in negligible concentrations on the secondary side of the steam generator.) The following indicates how tenacious this corrosive film actually is:

> "Experience has shown that mechanical methods or simple solution techniques have negligible effects on the removal of contamination. The radioisotopes are absorbed on or diffused into the tenacious oxide film so strongly that it is impossible to remove the contamination without removing all the protective corrosion film." (Reference 1.)

As discussed in Section 3.4.6, an additional measure of radioactivity confinement will be attained by welding cover plates over all lower assembly openings.

Radioactivity could conceivably be released to the environment only if both of the conditions below occurred: 1

a. Radioactivity is dislodged from the primary side surfaces.

b. The lower assembly primary side boundary is breached.

There are three mechanisms which could potentially dislodge the corrosion film:

a. Thermal shock

b. Chemical/corrosive attack

c. Mechanical shock

Temperature variations in the lower assembly can only occur during weather variations, but these are too slow to produce a thermal shock effect. Since the lower assemblies will be drained and sealed against moisture, chemical and corrosive attack will not occur. The possibility of mechanical shock during storage is not great since the steam generators are shielded on four sides. Even if a mechanical shock is assumed, the tenacious nature of the corrosive film is such that it would not dislodge more than an insignificant amount of radioactivity.

Since it is highly unlikely that more than an insignificant amount of radioactivity would be dislodged from a primary side surface, the second condition for radioactivity release to the environment, breaching the lower assembly primary side boundary, need not be considered.

Based on the above, it is concluded that there are no radiological accident considerations associated with onsite storage.

3.4.8 Conclusions

The steam generator lower assemblies will ultimately be disposed of in a licensed land burial site or decommissioned with the plant. Radiological considerations associated with each disposal alternative are acceptable. 1
Thus, the final decision on the alternatives to be employed for storage, handling, packaging and shipping of the lower assemblies will be based on economics and man-rem exposure.

3.4.9 References

1. "Decontamination of a PWR Primary System, SENA Plant, " Volume 33, Proceedings of the American Power Conference, 1971.

2. "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," WASH 1238, December 1972. 2

3.5 PLANT SECURITY

Specific plans for the physical protection of the Turkey Point Nuclear Units during the steam generator repair will be as stated in Chapter 9, Special Security Measures During Refueling/Major Maintenance Operations, and Chapter 10, Special Security Measures During Construction Operations, of the Turkey 7
Point Physical Security Plan. In accordance with these chapters, special procedures and orders shall be developed prior to the outage and they will comply with the facility operating license and the existing Security Plan.

3.6 QUALITY ASSURANCE

The Quality Assurance Programs for FPL, Bechtel Power Corporation, and Westinghouse Electric Corporation are described in this section.

3.6.1 FPL Quality Assurance Program

The FPL Quality Assurance Program is described in "FPL Topical Quality Assurance Report," (FPLTQAR 1-76A), Revision 2, September 8, 1977. The 3
Topical Quality Assurance Report is an integral part of the corporate Quality Assurance Manual (FPL-NQA-100).

The following amplifications of FPLTQAR 1-76A are applicable for the Turkey Point Project.

FPL currently contemplates assuming responsibility for conducting the following on-site construction activities:

a. Field Procurement Functions 1

b. Field Quality Control Functions

c. Field Test Control Program

d. Field Special Processes

e. Field Quality Assurance Functions

3.6.2 Bechtel Power Corporation Quality Assurance Program

Home Office Engineering and miscellaneous services for the quality assurance program used by Bechtel for the Turkey Point Plant Project are in accordance with the applicable portions of "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report BQ-TOP-1, Revision 2A, July 1977. Re- 7
sponsibility for the Turkey Point Plant Project has been assigned to the Gaithersburg Power Division of the Bechtel Power Corporation.

The following amplifications of BQ-TOP-1 are applicable for the Turkey Point Project.

Bechtel is not responsible for conducting the quality assurance activities speci- 1
fied in Section 3.6.1. 7

If the scope of work should change and Bechtel is assigned any or all of the functions of a. through e. of Section 3.6.1, these activities shall be performed in conformance with the applicable sections of Topical Report BQ-TOP-1, Revision 2A.

3.6.3 Westinghouse Electric Corporation Nuclear Energy Systems Division Quality Assurance Program

The quality assurance program used by Westinghouse during the design and fabrication of the steam generator lower assemblies is in accordance with "Westinghouse Nuclear Energy Systems Divisions Quality Assurance Plan," WCAP-8370. For activities which occurred during the period from January 1, 1975 to September 30, 1977, the program is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the program is presented 3
in WCAP-8370, Revision 8A.

3.7 REGULATORY GUIDE APPLICABILITY TO REPAIR PROGRAM

Section 2.1.4 discusses Regulatory Guide compliance during manufacture of the lower assemblies. Regulatory Guide applicability to repair program activities other than manufacture and other than those addressed in the FPL and Bechtel QA programs are discussed below.

1.31 Control of Stainless Steel Welding (Rev. 1, June 1973).

Control of stainless steel welding complies with the Interim Position on Regulatory Guide 1.31 (Branch Technical Position MTEB 5-1, dated 11/24/75) except as discussed below:

1. Reference: Paragraph B.1.b of Interim Position. Austenitic stainless steel welding filler materials used in the fabrication and installation of ASME Section III, Class 1, 2 and 3 components are controlled to deposit from FN 5 to 20 delta ferrite, except for 309 and 309L welding filler materials which are controlled to deposit from 5 to 15 percent delta ferrite and are used only for welding carbon or low alloy steel to austenitic stainless steel. Use of 309L welding filler material is further limited to the overlay deposit on the carbon or low alloy steel component nozzles or connecting pipe when postweld heat treatment is required. 7

   These limits for delta ferrite in austenitic stainless steel welding materials comply with Interim Regulatory Guide 1.31 since the upper limit of 20 percent delta ferrite does not apply for welds that are not heat treated after welding (Paragraph 3b), except for solution heat treatment. Solution heat treatment, although not required after welding, is permitted in order to avoid sensitization.

   Determination of delta ferrite is in accordance with ASME Section III, Division 1, 1977 edition, Paragraph NB-2433, except that an undiluted weld deposit is required for each heat of bare wire used with the Gas Metal-Arc (GMA) process.

2. Reference: Paragraph B.2 of the Interim Position. This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1977 edition.

3. Reference: Paragraph B.3.a of the Interim Position. Magnetic measurement of production welds for delta ferrite is unnecessary when austenitic stainless steel welding materials are controlled to deposit FN 5 to 20 delta ferrite based 7
on chemistry, except for 309 and 309L welding materials which are controlled to deposit 5 to 15 percent delta ferrite based on chemistry.

4. Reference: Paragraph B.3.b of the Interim Position. This paragraph is complied with for welding material certification.

5. Reference: Paragraphs B.4.a, .b, and .c of the Interim Position. Measurement of production welds for delta ferrite is not performed.

1.44 Control of the Use of Sensitized Stainless Steel (May 1973)

Subject to the following statements, the use of unstabilized austenitic stainless steel for components which are part of the reactor coolant pressure boundary, which are relied upon to permit adequate core cooling during any mode of normal operation or postulated accident conditions, complies with Regulatory Guide 1.44.

1. Reference: Paragraph C.1 of the Regulatory Guide. Contamination of austenitic stainless steels (Type 300 series) by compounds which could cause stress corrosion cracking is avoided during all stages for repair welding. Except for trichlorotrifluoroethan (TCTFE) which meets the requirements of Military Specification MIL-C-8130 2B, cleaning is limited to solutions which contain not more than 200 ppm of chlorides. Rinsing or flushing is with water containing less than 200 ppm of chlorides. Special rinsing techniques are used to assure complete removal of TCTFE when crevices or undrainable areas are present. Foreign substances in contact with austenitic stainless steel (dye, lubricants, penetrant materials, marking materials, masking tape, etc.) are controlled so as not to contain more than 200 ppm of chlorides, or are removed immediately following the operation in which they are used. Crevices 3
and undrainable areas are protected prior to use of materials containing more than 200 ppm of chlorides. All substances in contact with austenitic stainless steel are removed prior to any elevated temperature treatment.

In the field, austenitic stainless steel components are stored clean and dry to prevent contamination. System hydrostatic tests are performed with water which contains less than 200 ppm of chlorides. The influent water quality during final flushing or pre-operational testing of the completed system is at least equivalent to the quality of demineralized water as defined in ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants.

2. Reference: Paragraph C2 of the Regulatory Guide. All grades of austenitic stainless steels (Type 300 series) are required to be furnished in the solution heat-treated condition before fabrication or assembly into components or systems. The solution heat treatment varies according to the applicable ASME or ASTM material specification.

3. Reference: Paragraph C.3 of the Regulatory Guide. All austenitic stainless steels are furnished in the solution heat-treated condition in accordance with material specification. For material solution heat treated by the material manufacturer, testing to determine susceptibility to intergranular corrosion is performed only when required by the material specification. During repair welding, austenitic stainless steels are not permitted to be exposed to temperatures in the range of 800° F to 1500° F except for welding. Welding practices are controlled to avoid severe sensitization, as described in 6 below. Unless otherwise required by the material specification, the maximum length of time for cooling from the solution heat treat temperature to below 800° F is specified in the equipment specification. Corrosion testing in accordance with ASTM A 262-70, Practice A or E, or ASTM A 393 may be required if the maximum length of time for cooling below 800° F is exceeded or the solution heat treat 3
condition is in doubt.

4. Reference: Paragraph C.4 of the Regulatory Guide. Use of low carbon (0.03% maximum) unstabilized austenitic stainless steel is not required since the reactor coolant meets acceptable water chemistry requirements over 250° F.

5. Reference: Paragraph C.5 of the Regulatory Guide. Heat treating austenitic stainless steel in the temperature range from 800° F to 1500° F is not permitted. Since sensitization is avoided, testing to determine susceptability to intergranular attack is not performed.

6. Reference: Paragraph C.6 of the Regulatory Guide.
Welding practices are controlled to avoid severe sensitization in the heat-affected zone of unstabilized austenitic stainless steel as described below. Unless otherwise stated, the position applies to both Bechtel and Bechtel suppliers and subcontractors.

   a. Weld Heat Input

   Bechtel controls weld heat input during field installation by using shielded metal-arc welding (SMAW) and gas tungsten-arc welding (GTAW) processes only, and by limiting the size of electrodes for each process to 5/32-inch and 1/8-inch diameter maximum, respectively.

   b. Interpass Temperature

   The interpass temperature is controlled so as not to exceed 350° F.

1.71 Welder Qualification for Areas of Limited Accessibility (December 1973).

The response to this Regulatory Guide is as follows:

1. Reference: Paragraph C.1 of the Regulatory Guide. Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX. Welding conducted in areas of limited access is subjected to the required nondestructive testing and no waiver or relaxation of examination methods or acceptance criteria because of the limited access is permitted.

2. Reference: Paragraph C.2 of the Regulatory Guide. Requalification is required when any of the essential variables of ASME Section IX are changed, or when any authorized inspector questions the ability of the welder to perform satisfactorily the requirements of ASME Sections III or IX.

3. Reference: Paragraph C.3 of the Regulatory Guide. Production welding is monitored and welding qualifications are certified in accordance with 1. and 2. above.

1.123 Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (October 1976).

The FPL position regarding ANSI N45.2.13 is as stated in "FPL Topical Quality Assurance Report," FPLTQAR 1-76A, Revision 2, September 8, 1977, Appendix C. 3

The Bechtel position is as stated in "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report, BQ-TOP-1, Revision 2A, July 1977, Introduction. 7

TABLE 3.1-1

RESULTS OF ANALYSES FOR SAFETY-RELATED BURIED PIPING

Item Description	Approximate Location	Governing Load Condition	Allowable External Pressure (1) (PSI)	Maximum External Pressure (2) (PSI)	Factor of Safety
30" Diameter Intake Cooling Water Pipe	Beneath haul route to Unit 3	Trailer Transporter Loads	50.0	17.4	2.87
30" Diameter Intake Cooling Water Pipe	Beneath equipment hatch construction area for Unit 4	Crawler Crane Loads	50.0	16.0	3.13
36" Diameter Intake Cooling Water Pipe	Beneath haul route to Unit 3	Trailer Transporter Loads	53.1	14.7	3.61

NOTES

(1) Pipe allowable determined per AWWA C101-67

(2) Without the use of timber matting, structural fill, or structural bridging.

TABLE 3.1-2

RESULTS OF ANALYSES FOR SAFETY-RELATED BURIED ELECTRICAL DUCT BANKS

Item Description	Approximate Location	Governing Load Conditions	Allowable Extreme Fiber Stress in Tension (1) (PSI)	Maximum Extreme Fiber Stress in Tension (2) (PSI)	Factor of Safety
2000 psi Concrete Duct Bank	Beneath haul route to Unit 3 and beneath equipment hatch construction area for Unit 3	Trailer Transporter or Crawler Crane Loads	71.55	22.95	3.1

NOTES

(1) Allowable extreme fiber stress in tension equals 1.6 $\sqrt{f'_c}$ per ACI 318-63

(2) Using timber matting

TABLE 3.2-1

CONDUIT TO BE REMOVED

CONDUIT NUMBER	CLASS IE (YES/NO)	CABLES CONNECTED TO
4D002	YES	LT-496 STEAM GENERATOR "C" LEVEL
4D004	YES	LT-495 STEAM GENERATOR "C" LEVEL
4D004	YES	FT-435 REACTOR COOLANT LOOP "C" FLOW
4D005	YES	FT-435 REACTOR COOLANT LOOP "C" FLOW
4D006	YES	LT-495 STEAM GENERATOR "C" LEVEL
4D008	NO	LT-930 ECCS ACCUMULATOR "C" LEVEL
4D008	NO	PT-931 ECCS ACCUMULATOR "C" PRESSURE
4D010	YES	LT-497 STEAM GENERATOR "C" LEVEL
4D013	YES	FT-434 REACTOR COOLANT LOOP "C" FLOW
4D062	NO	LT-930 ECCS ACCUMULATOR "C" LEVEL
4D338	YES	LT-494 STEAM GENERATOR "C" LEVEL
4D342	NO	LT-930 ECCS ACCUMULATOR "C" LEVEL
4D343	NO	PT-931 ECCS ACCUMULATOR "C" PRESSURE
4D441	NO	4H200 REACTOR CAVITY MANIPULATOR CRANE POWER
4D645	YES	LT-495 STEAM GENERATOR "C" LEVEL
4D645	YES	FT-435 REACTOR COOLANT LOOP "C" FLOW
4C418	NO	TE-1499 CONTAINMENT ATMOSPHERE TEMPERATURE
4C449	NO	TE-3456 CHARCOAL BED 3B, 3C TEMPERATURE
4C449	NO	TE-3457 CHARCOAL BED 3B, 3C TEMPERATURE
4C449	NO	TE-3458 CHARCOAL BED 3B, 3C TEMPERATURE
4C449	NO	TE-3459 CHARCOAL BED 3B, 3C TEMPERATURE
4C449	NO	TE-3460 CHARCOAL BED 3B, 3C TEMPERATURE
4C449	NO	TE-3461 CHARCOAL BED 3B, 3C TEMPERATURE
4C449	NO	TE-3462 CHARCOAL BED 3B, 3C TEMPERATURE

TABLE 3.2-1 (Cont'd)

CONDUIT TO BE REMOVED

CONDUIT NUMBER	CLASS IE (YES/NO)	CABLES CONNECTED TO
3D092 *	NO	PI-131B REACTOR COOLANT PUMP "A" SHAFT SEAL Δ P
3D093	NO	PT-131 REACTOR COOLANT PUMP "A" SHAFT SEAL Δ P
3D008	NO	FT-156A REACTOR COOLANT PUMP "A" SEAL INJECTION FLOW
3D271	NO	FT-156A REACTOR COOLANT PUMP "A" SEAL INJECTION FLOW
3D007	NO	FT-156B REACTOR COOLANT PUMP "A" SEAL INJECTION FLOW
3D272	NO	FT-156B REACTOR COOLANT PUMP "A" SEAL INJECTION FLOW
3D067	NO	PI-156B REACTOR COOLANT PUMP "A" SHAFT SEAL Δ P
3D006	NO	PT-156 REACTOR COOLANT PUMP "A" SHAFT SEAL Δ P
3D189	NO	LC-406B REACTOR COOLANT PUMP "A" STANDPIPE LEVEL
3D190	NO	LC-406A REACTOR COOLANT PUMP "A" STANDPIPE LEVEL
3C033	YES	FT-414 REACTOR COOLANT LOOP "A" FLOW
3D012	YES	FT-415 REACTOR COOLANT LOOP "A" FLOW
3D017	YES	FT-416 REACTOR COOLANT LOOP "A" FLOW
3D079	NO	LC-417 REACTOR COOLANT PUMP "A" UPPER BEARING OIL RESERVOIR LEVEL
3D072	NO	PC-417 REACTOR COOLANT PUMP "A" OIL LIFT PUMP PRESSURE
3D016	YES	LT-474 STEAM GENERATOR "A" LEVEL
3D015	YES	LT-475 STEAM GENERATOR "A" LEVEL
3D018	YES	LT-476 STEAM GENERATOR "A" LEVEL
3D014	NO	LT-477 STEAM GENERATOR "A" LEVEL

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TABLE 3.2-1 (Cont'd)

CONDUIT TO BE REMOVED

CONDUIT NUMBER	CLASS IE (YES/NO)	CABLES CONNECTED TO
3D037	NO	LT-920 ECCS ACCUMULATOR "A" LEVEL
3D036	NO	PT-921 ECCS ACCUMULATOR "A" PRESSURE
3C626	NO	CV-853A ECCS ACCUMULATOR "A" NITROGEN FILL VLAVE
3C423	NO	TE-1496 REACTOR COOLANT PUMP "A" AREA CONTAINMENT TEMPERATURE

*Note: The first digit of the conduit number indicates the associated unit; "3" indicates Unit 3 and "4" indicates Unit 4.

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TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 4 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description	
1	4J02-4P200C/1	No	Reactor Coolant Pump C Motor Power	
2	4J02-4P200C/2	No	Reactor Coolant Pump C Motor Power	
3	4P10-4H200/1	No	Manipulator Crane 4H200	
4	4P200C-T4P53/1	No	Reactor Coolant Pump C Motor Heater Power	
5	4P232C-T4P52/1	No	Reactor Coolant Pump C Lift Oil Pump 4C	
6	FIC154-T4C23/1	No	Reactor Coolant Pump C B/P Flow Seal Injection	7
7	FIC492-T4C23/1	No	RTP Bypass Flow Indicator Controller	
8	LC408A-T4C23/1	No	Reactor Coolant Pump C Standpipe Level Controller	
9	LC408B-T4C23/1	No	Reactor Coolant Pump C Standpipe Level Controller	
10	LC4437-T4C23/1	No	Reactor Coolant Pump C Upper Oil Reservoir Level	
11	LS1571-T4C21/1	No	NPSH for South Containment Sump #4 Recirculation	
12	M0865C-T4C23/1	Yes	Accumulator C Discharge Stop Valve Control	
13	M0865C-T4P52/1	Yes	Accumulator C Discharge Stop Valve Motor	
14	PC437-T4C23/1	No	Reactor Coolant Pump C Lift Oil Pump 4C	
15	T4C23-TB4129/1	No	Accumulator C SI Test Solenoid Valve 4V850E	
16	T4C23-TB4129/2	No	Accumulator C SI Test Solenoid Valve 4V850F	
17	T4C23-TB4129/3	No	Accumulator C Makeup Stop Solenoid Valve 4V851C	7
18	T4C23-TB4129/4	No	Accumulator C Makeup Stop Solenoid Valve 4V852C	

TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 4 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description	
19	T4C23-TB4129/5	No	Accumulator C Makeup Stop Solenoid Valve 4V853C	
20	T4121/LT484/1	Yes	Steam Generator B Channel I Narrow Range Level	
21	T4121-TB4369/1	Yes	Steam Generator C Channel I Narrow Range Level	
22	T4121-TB4369/2	Yes	Reactor Coolant Flow - Loop C	
23	T4123-LT476/1	Yes	Steam Generator A Channel III Narrow Range Level	7
24	T4124-LT485/1	Yes	Steam Generator B Channel II Narrow Range Level	
25	T4114-TE1499/1	No	Temperature Element on Containment Perimeter Wall	
26	TB4115-TE3456/1	No	Temperature Element on Emergency Filter Instrumentation	
27	TB4115-TE3457/1	No	Temperature Element on Emergency Filter Instrumentation	
28	TB4115-TE3458/1	No	Temperature Element on Emergency Filter Instrumentation	
29	TB4115-TE3459/1	No	Temperature Element on Emergency Filter Instrumentation	
30	TB4115-TE3460/1	No	Temperature Element on Emergency Filter Instrumentation	
31	TB4115-TE3461/1	No	Temperature Element on Emergency Filter Instrumentation	
32	TB4115-TE3462/1	No	Temperature Element on Emergency Filter Instrumentation	
33	TB4115-TE3463/1	No	Temperature Element on Emergency Filter Instrumentation	

TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 4 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description
34	TB4129-T4C13/1	No	Reactor Coolant Pump C Seal Water Return By-Pass Valve 4V307
35	TB4129-T4C13/2	No	Reactor Coolant Pump C Seal Water Discharge Solenoid Valve 4V303C
36	TB4200-FIC154/1	No	Reactor Coolant Pump C Seal Injection B/P Flow
37	TB4200-FT154A/1	No	Reactor Coolant Pump C Seal Injection High Range
38	TB4200-FT154B/1	No	Reactor Coolant Pump C Seal Injection Low Range
39	TB4201-LT930/1	No	Accumulator Tank C Level
40	TB4367-LT474/1	Yes	Steam Generator A Channel I Narrow Range
41	TB4367-LT477/1	No	Steam Generator A Channel I Wide Range Level
42	TB4369-FT434/1	Yes	Reactor Coolant Flow Loop C
43	TB4369-LT494/1	Yes	Steam Generator C Channel I Narrow Range Level
44	TB4371-FT154A/1	No	Reactor Coolant Pump C Seal Injection High Range
45	TB4371-FT154B/1	No	Reactor Coolant Pump C Seal Injection Low Range
46	TB4371-FT435/1	Yes	Reactor Coolant Flow Loop C
47	TB4371-HCV137/1	Yes	Reactor Coolant Leaving Excess Let Down HX Hand Control Valve
48	TB4371-LT495/1	Yes	Steam Generator C Channel II Narrow Range Level
49	TB4371-LT497/1	No	Steam Generator C Channel I Wide Range Level

7

TABLE 3.2-1

CIRCUITS IDENTIFIED FOR REMOVAL
INCIDENT TO UNIT 4 STEAM GENERATOR
REMOVAL

Item No.	Circuit No.	Class IE (Yes/No)	Equipment Description	
50	TB4371-LT930/1	No	Accumulator Tank C Level	
51	TB4379-LT496	Yes	Steam Generator C Channel III Narrow Range Level	
52	TB4371-PI125B/1	No	Reactor Coolant Pump C Shaft Differential Pressure	7
53	TB4371-PT138/1	No	Excess Letdown Heat Exchanger Pressure Indicator	
54	TB4371-PI154/B1	No	Reactor Coolant Pump C #1 Seal Injection Flow	
55	TB4371-P1931/1	No	Accumulator Tank C Pressure Indicator	
56	TB4372-LT475/1	Yes	Steam Generator A Channel II Narrow Range Level	
57	TB4379-LT486/1	Yes	Steam Generator B Channel III Narrow Range Level	
58	TB4379-LT496/1	Yes	Steam Generator C Channel III Narrow Range Level	
59	4V2A-T4P11/1	No	Reactor CRDM Cooler 4A	
60	4V2A-T4P51/1	No	Reactor CRDM Cooler 4A Motor Heater	
61	4V2B-T4P12/1	No	Reactor CRDM Cooler 4B	
62	4V2B-T4P51/1	No	Reactor CRDM Motor 4B Heater	
63	4V30C-T4P43/1	Yes	Emergency Containment Cooling Fan 4C Power Cable	

TABLE 3.3-1

TYPICAL PORTABLE SURVEY INSTRUMENT SPECIFICATIONS

Instrument	Detector	Power Source	Type of Radiation Measured	Range	Controls	Application
Eberline Model RD-1 Rad. Owl	3 inch diameter air-filled ionization chamber, with beta shield for window	Two "D" size cells, 1.0-1.7 volts each	Beta-gamma	5,50,500 mR int; 5,50,500 mR/hr; 5,50,500 R/hr; full scale ±10%	One zero check pushbutton. One zero set knob. One range switch with OFF and BATT. Check positions.	Multi-range determination of personnel exposure level
Victoreen Model 740-F and Model 740-D Cutie Pie	Air-filled ionization chamber with beta shield for window	Four 22-1/2-volt batteries and one 1.34-volt mercury	Beta-gamma	25,250,2500 & 25,000 mR/hr ±10%	One zero adjust switch, One selector switch with seven positions for zero set, battery check and scale multiplications.	Low and medium range determination of personnel exposure level
Victoreen Model 592-B	Ionization chamber with no beta window	Six 22-1/2-volt batteries and three 1.3-volt mercury cells	Gamma	10,100 & 1000 mR/hr ±10%	One zero adjust switch. One selector switch with five positions for zero set and scale multiplication.	Low range determination of personnel exposure levels for gamma
Eberline Model PIC-6A	Gas-filled ionization chamber with beta window on bottom (60 cm. Hg pressure-propane)	Two NEDA type 1604 batteries 6.5-9V	Beta-gamma	1-1000 mR/hr 1-1000 R/hr ±20%	One control switch turns instrument OFF, provides battery check and selects the range.	Multi-range determination of personnel exposure levels
Nuclear Chicago Model 2592	Air-filled ionization chamber with beta window	Three "D" size flashlight cells	Beta-gamma	0-10,0-100, 0-1000 R/hr 0-10 R ± 10%	One zero adjust switch, one switch for dose, zero set and dose rate. One switch for OFF, battery check and scale multiplication.	High-range determination of personnel exposure levels
Atometer	Utilizes one G-M tube	One "C" size 1.5-volt cell	Beta-gamma	0-150 mR/hr 0-15 R/hr ±15%	One rotary switch for the OFF position, BATT check and scale selector. One pushbutton to be depressed to obtain dose rate.	Used for measuring low to low intermediate range levels of beta - gamma
Eberline Model PNR-4	Nine-inch-diameter cadmium loaded, polyethylene sphere with a BF_3 tube in the center	Five standard "D" size cells	Neutron	5,000 mRem/hr full scale ±10%	One selector switch for OFF, ON, and BATT. One H. V. adjust.	Used for measuring low to medium range levels of neutron dose rates

TABLE 3.3-1 (Cont'd)

TYPICAL PORTABLE SURVEY INSTRUMENT SPECIFICATIONS

Ludlum Model 12	4mm x 4mm enriched lithium iodide thermal neutron detector and a 10-inch spherical polyethylene moderator	Two standard "O" flashlight cells	Neutron	500, 5,000. 50,000 cpm full scale ±10%	One selector switch for OFF, Battery Test, and scale multiplication.	Used for measuring low range levels of neutron flux
Eberline Model E-400	HP-177B G-M detector "side window" with beta shield	Five standard "D" size cells	Beta-gamma	0-2 mR/hr. 0-200 mR/hr, ± 10%	One selector switch for OFF, ON, and BATT. One H. V. adjust.	Used for detecting low-level radiation, beta - gamma
Eberline Model 6112 "Teletector"	Two G-M tubes inside a telescoping probe that extends over thirteen feet	Four "C" size cells	Beta-gamma	2,50 mR/hr & 2,50,1000 R/hr ± 10%	One selector switch for OFF (AUS), battery and scale changes. One connection for Aural indication.	Used for monitoring radiation at all levels. Probe can be used underwater
Eberline Model Rm-16	HP-210 G-M detector "Pancake"	One 6.3-volt gelled electrolite battery with trickle charge connection for AC	Beta-gamma	0-20,000 mR/hr ± 10%	One reset for Hi-Lo Alarm. One switch for power ON. One pushbutton for BATT check. Has connection for scaler.	Used for detecting medium to high range beta - gamma contamination levels
Nuclear Chicago Model 2650	Model 2660 side window G-M probe	Four "D" size cells.	Beta-gamma	0.1, 0.3, 1, 3, 10, 30, & 100 mR/hr 150, 1500, 15,000 and 150,000 cpm full scale ±10%	One selector switch for OFF, BAT SET, & scale multiplication. One BAT ADJ control for optimum voltage. One switch to adjust time-constant on 3 lowest mR/hr ranges.	Used for measuring low medium range levels of contamination and radiation
Eberline Model E-520	Utilizes two G-M tubes: an external B-shield (HP-177) detector for the four low ranges and an internal detector for the highest range	Two "D" size cells	Beta-gamma	0-0.2, 0-2, 0-20 mR/hr ±8% 0-200 mR/hr ±15%, 0-2,000 mR/hr ±10%	One selector switch provides OFF position, BATT check & five scale multiplications. One reset pushbutton. One fast-slow response knob.	Used for detecting low level radiation - beta and gamma
Eberline Model Rm-14	HP-177B G-M detector "side window" with beta shield.	One Ni-Cd battery with trickle charge connection for 105-125V AC	Beta-gamma	0-500, 0-5000, 0-50,000 cpm ± 5%	One selector switch for OFF, BATT and scale multiplication. One volume control for Aural indication. One Reset switch. One Alarm set switch (in back).	Used for determining contamination levels of beta - gamma
	HP-210 G-M detector "Pancake"	Same	Beta-gamma	Same	Same	Same

TABLE 3.3-1 (Cont'd)

TYPICAL PORTABLE SURVEY INSTRUMENT SPECIFICATIONS

Eberline Model Rm-15	HP-210 G-M detector "Pancake"	One Ni-Cd battery with trickle charge connection for 105-125V AC	Beta-gamma	0-500, 0-5000, 0-50,000, 0-50,000 CPM, ± 15%	One selector switch for OFF, BATT, and scale multiplication. One volume control for Aural indication. One Reset switch. One Alarm set switch (in back). One Response switch.	Used for determining contamination levels of beta - gamma
	ZnS (Ag) Scintillation detector	Same	Alpha	Same	Same (Voltage must be adjusted)	Used for determining contamination levels for alpha
Eberline Model PRM-4	HP-177B G-M detector "side window" with beta shield	Five standard "D" size cells	Beta-gamma	1-500 K cpm ± 10%	One selector switch for OFF, ON, and BATT. One H. V. adjust.	Used for detecting contamination, beta - gamma.

TABLE 3.3-2

ESTIMATED MAN-REM DOSE TO WORKERS FOR THE

REPAIR OF THREE (3) STEAM GENERATORS - CHANNEL CUT METHOD

		Range of Radiation Field (Rem/Hr.)	Estimated Manhours In Radiation-Fields (Hour)	Task-Estimated Dose (Man-Rem)
1.	Concrete and structural steel removal and replacement.	0.001-0.030	13,660	88
2.	Construction of pedestal cranes, preparation of polar crane, miscellaneous cribbing platforms, S. G. transfer bridge.	0.001-0.020	10,280	32
3.	Removal, modification & reinstallation of S. G. upper assemblies and major piping.	0.001-1.5	24,600	256
4.	Construction of temporary facilities & support services.	0.001-0.150	19,120	215
5.	General decontamination & disposal of contaminated materials/cleanup.	0.001-0.010	42,310	201
6.	Removal & reinstallation of miscellaneous piping equipment & insulation.	0.001-0.040	8,850	125
7.	Non-manuals (e.g., QC, Engineers, HPs).	0.001-0.200	68,540	436
8.	Decontamination of the channel head.	0.005-20	1,840	214
9.	Cut channel head & remove old S. G. lower assembly.	0.001-3	3,240	166
10.	Weld shield cover on lower assembly:			
	a. at channel head	0.003-0.075	760	40
	b. at transition end		530	53
11.	Cut and remove old divider plate, weld new divider plate.	0.001-6	2,640	29
12.	Install new S. G., weld channel head.	0.005-5	11,000	204
13.	Placement of steam generator in storage.	0.001-0.075	225	25
	TOTAL		182,800	2,084
	Estimated Range			1730 - 2480

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TABLE 3.4-1

ESTIMATES OF AIRBORNE RELEASES
TO ENVIRONMENT DURING STEAM GENERATOR
LOWER ASSEMBLY DISPOSAL EFFORT (1)

Isotope	Release/Unit (Ci)
Co-58	9.54 E-3
Co-60	1.56 E-2
Mn-54	1.48 E-3
Fe-59	1.04 E-4
Cr-51	1.2 E-4
Zr-95	7.73 E-4
Nb-95m	1.64 E-5
Nb-95	1.42 E-3
Ru-103	2.81 E-4
I-131	1.73 E-5
Ba-137m	5.63 E-5
Cs-137	5.99 E-5
Ce-141	9.86 E-5
Ce-144	2.80 E-3
Pr-144	2.68 E-3
TOTAL	3.49 E-2

7

NOTE (1) Disposal effort releases are given for Unit 3 which also bound the releases expected for Unit 4.

TABLE 3.4-2

COMPARISON OF GASEOUS EFFLUENT RELEASES

Isotope	Average 1978-1979 Release/Unit (Ci)	Estimated Release/Unit During SG Disposal Effort (Ci) (1)
Noble gases	11,300	Negligible
Iodines	0.055	1.73×10^{-5}
Particulates	0.032	3.49×10^{-2}
Tritium	0.5	Negligible

7

NOTE (1) Disposal effort releases are given for Unit 3 which also bound the releases for Unit 4.

TABLE 3.4-3

ESTIMATED MAN-REM DOSE TO WORKERS FOR STEAM GENERATOR DISPOSITION ALTERNATIVES (1)

Task	Alternative a (2)	Alternative b (2)	Alternative c (2)	Alternative d (2)	Alternative e (2)	Alternative f (2)
1. Placement of steam generators in storage facility or laydown area.	25	25	25	25	16	25
2. Preparation for cutup at the storage facility	Negligible	Negligible	Negligible	Negligible	NA	NA
3. Placement of temporary shielding at the lower assemblies	220	NA	NA	NA	NA	NA
4. Solidification agent injection	NA	20	NA	NA	NA	NA
5. Allowance for cutup problems due to solidification agent injection	NA	60	NA	NA	NA	NA
6. Decontamination operations, including processing of decontamination waste	NA	NA	150	NA	NA	NA
7. Surveillance for 35 years	NA	NA	NA	1.1	NA	1.1
8. Cutup and handling	1160	615	100	12.2	NA	NA
9. Packaging and shipping	105.7	52	12	1.1	3.37	NA
10. Decommissioning	NA	NA	NA	NA	NA	0.2
11. Miscellaneous support activities by manual crafts at the steam generators (e.g., clean up, erect and remove scaffolding, temporary lighting, etc.)	0.5	0.15	0.04	0.04	0.51	0.02
12. Supervisory and nonmanual time at the steam generators (e.g., health physics, security, field engineering, etc.)	14	6	1.2	0.13	Negligible	0.04
Total for the Alternative	1525.2	778.15	288.24	39.57	27.66	26.36
Estimated range for the Alternative	750 - 1525	400 - 825	125 - 575	35 - 45	26.5-28	25.5 - 26.5

(1) Based on 3 steam generators.
(2) Corresponds to alternatives listed in Subsection 3.4.4.

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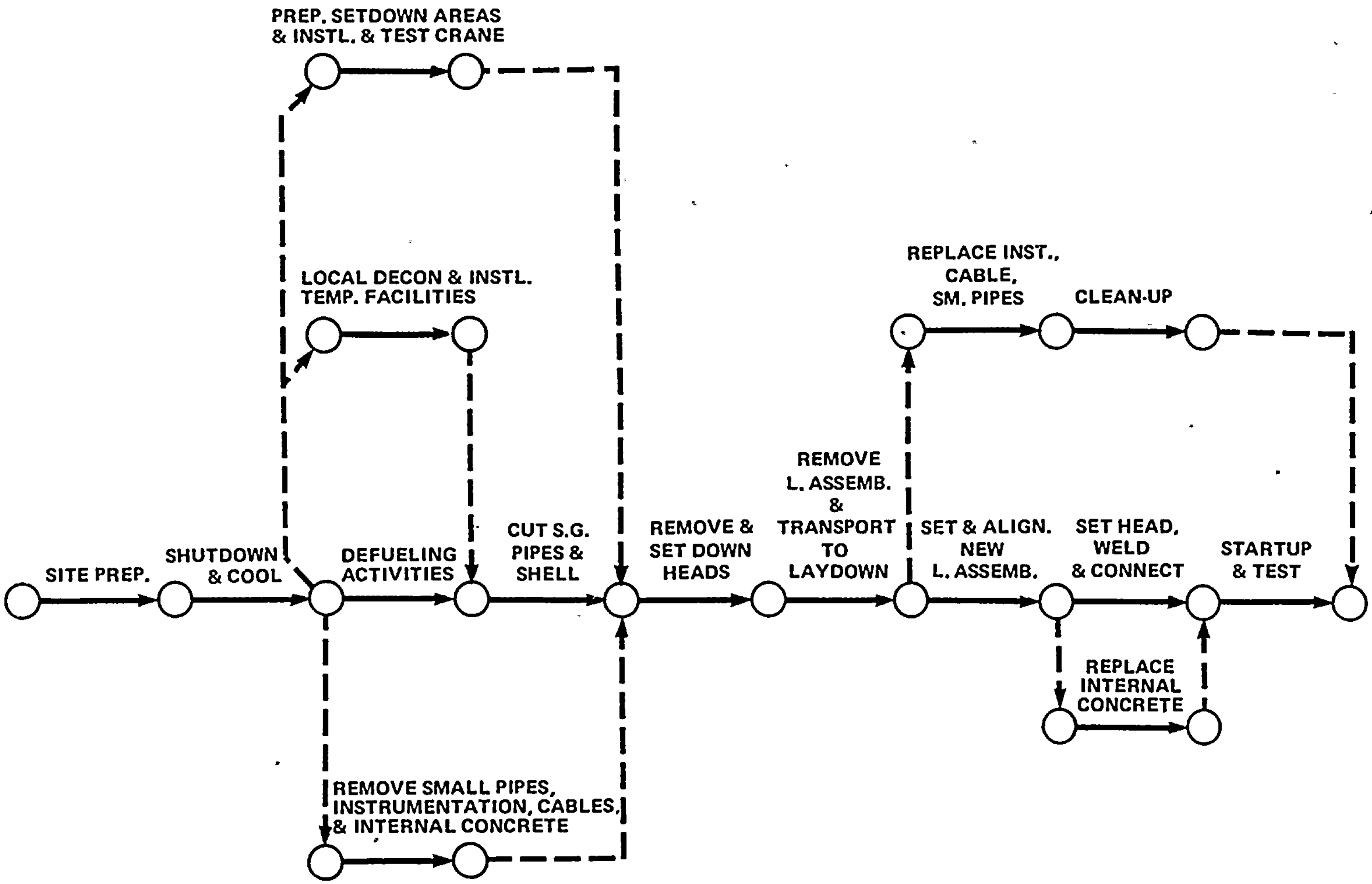
TABLE 3.4-4

ESTIMATED MAN-REM DOSE TO THE PUBLIC FOR STEAM GENERATOR DISPOSITION ALTERNATIVES

	Cumulative Whole Body Dose at the Site Boundary (1) (man-rem)	Critical Organ (Lung) Dose at the Site Boundary (1) (man-rem)	Cumulative Direct Dose During Transport (2) (man-rem)
a. Cutting up, packaging and shipping of six steam generator lower assemblies, assuming no decontamination and no solidification agent injection.	2.8×10^{-6}	5.5×10^{-5}	2.1
b. Cutting up, packaging and shipping of six steam generator lower assemblies, assuming solidification agent injection prior to cutting.	1.5×10^{-6}	5.5×10^{-5}	1.2
c. Cutting up, packaging and shipping of six steam generator lower assemblies, assuming decontamination of the steam generators.	2.8×10^{-7}	5.5×10^{-6}	1.2
d. Cutting up, packaging and shipping of six steam generator lower assemblies, assuming 35 years of storage prior to cutting.	3.7×10^{-5}	4.1×10^{-7}	0.9
e. Placing in cask, securing and shipping by barge of six steam generator lower assemblies, assuming no decontamination and no solidification agent injection.	6.0×10^{-6}	NONE	0.03
f. Store steam generators on site for 35 years and decommission with plant.	3.7×10^{-5}	NONE	NA

NOTES: (1) The site boundary (4164 feet) was chosen for the cumulative direct and particulate doses associated with storage, cutup and packaging of the steam generator lower assemblies for conservatism. It is obvious that no individual will be at the site boundary for the duration. The closest permanent resident to the site is approximately five miles away.

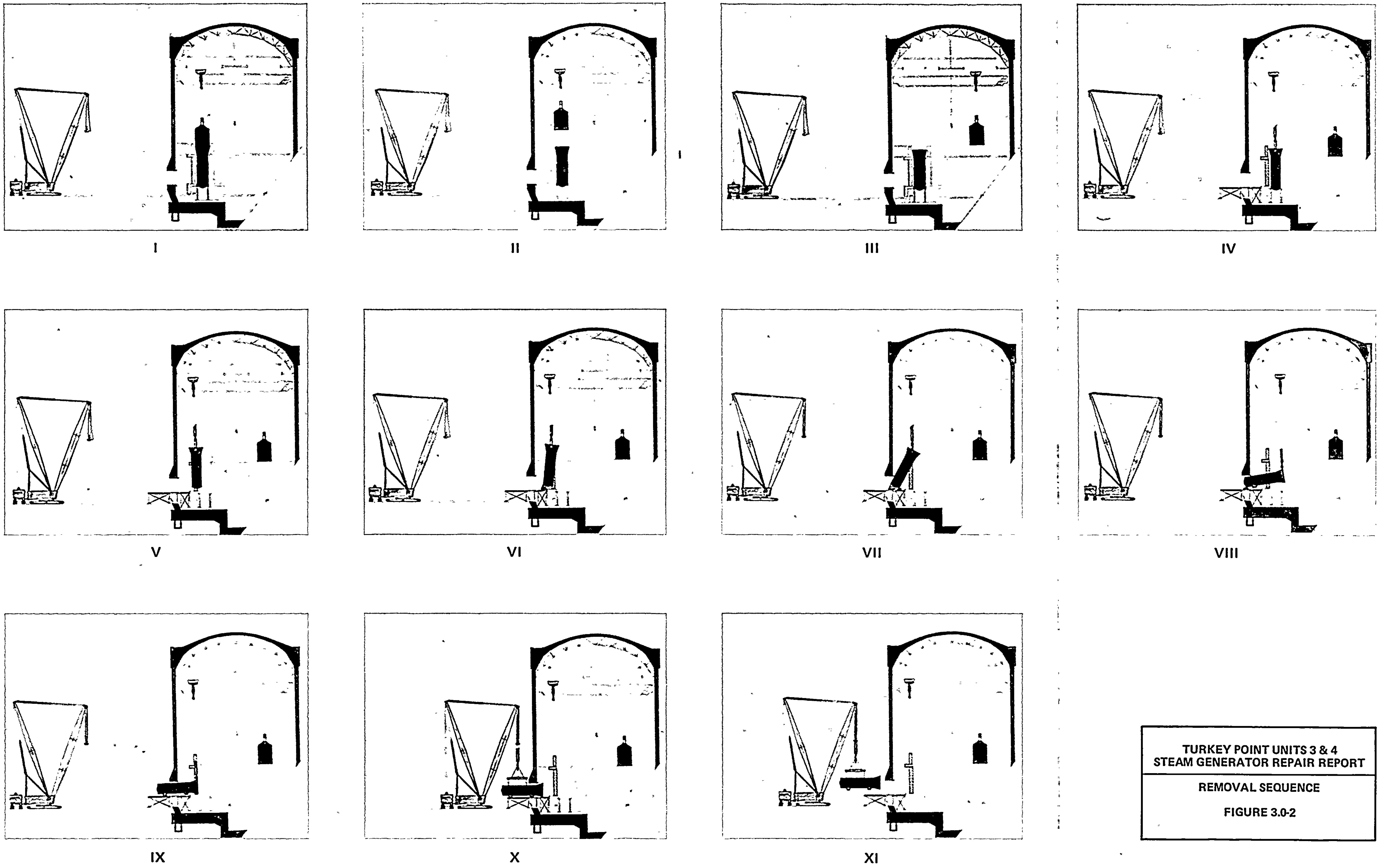
(2) Public doses during transportation of the steam generator pieces are based on the conservative guidelines provided in Reference 2.



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

OUTAGE SEQUENCE

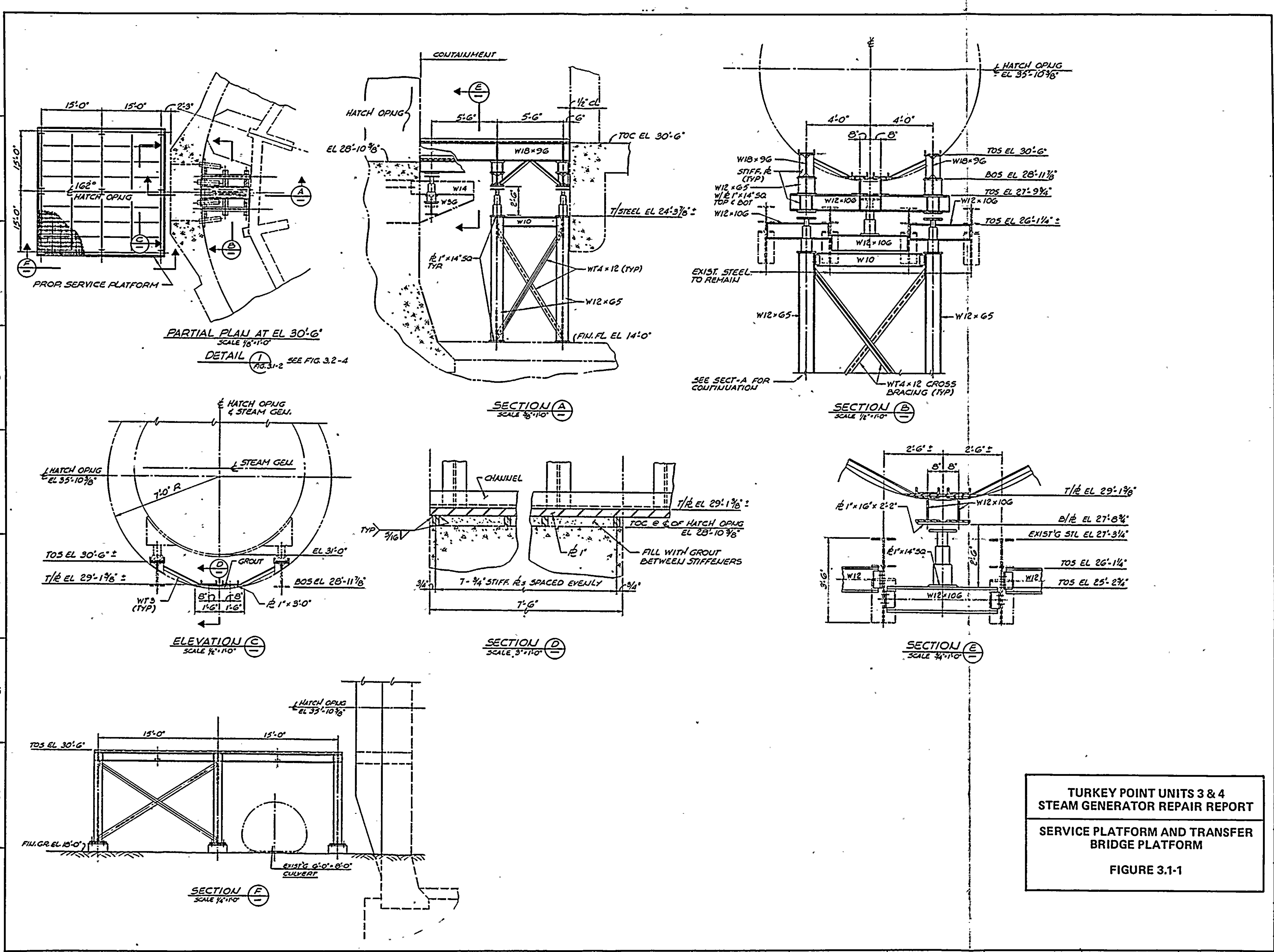
FIGURE 3.0-1



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

REMOVAL SEQUENCE

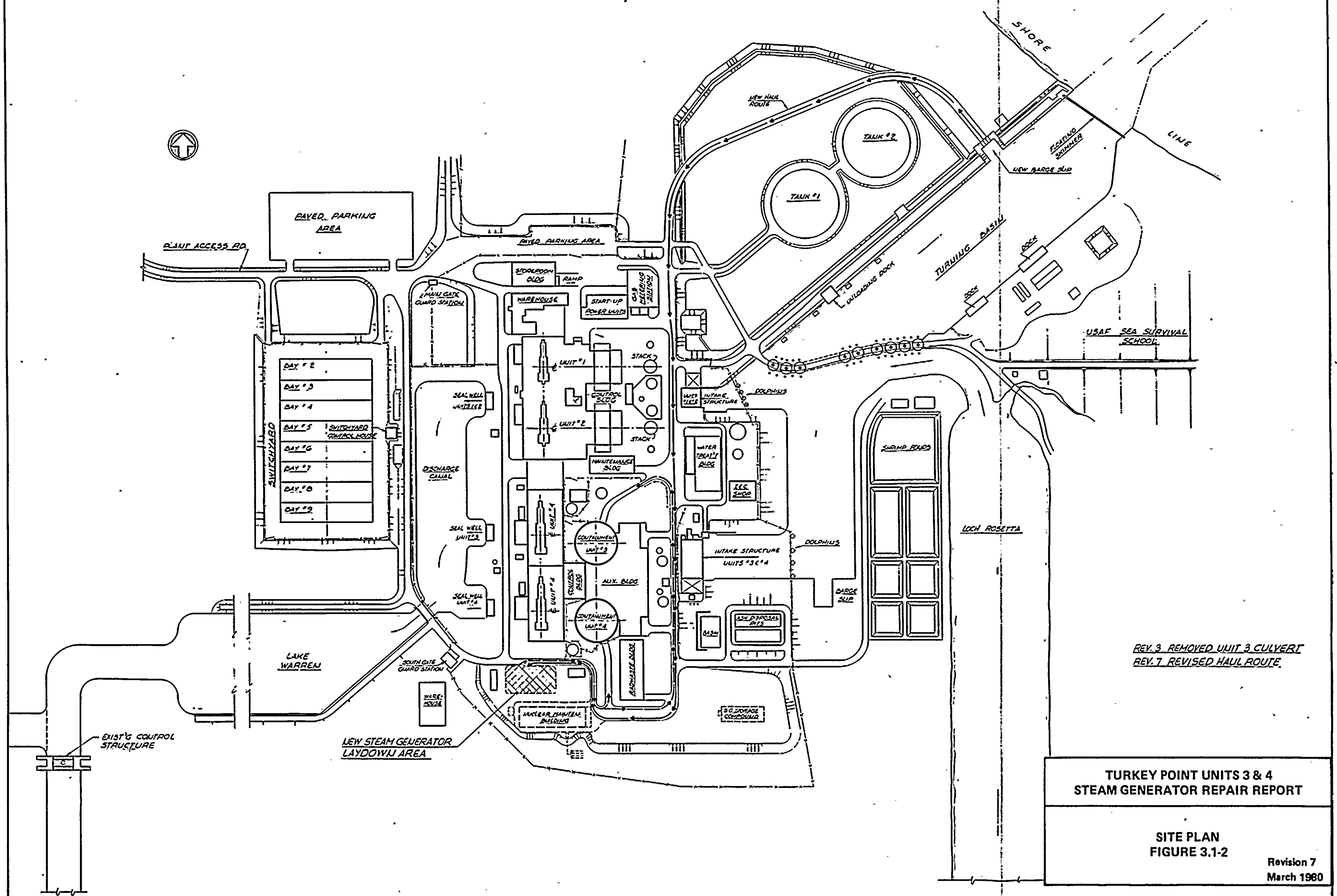
FIGURE 3.0-2



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

SERVICE PLATFORM AND TRANSFER BRIDGE PLATFORM

FIGURE 3.1-1

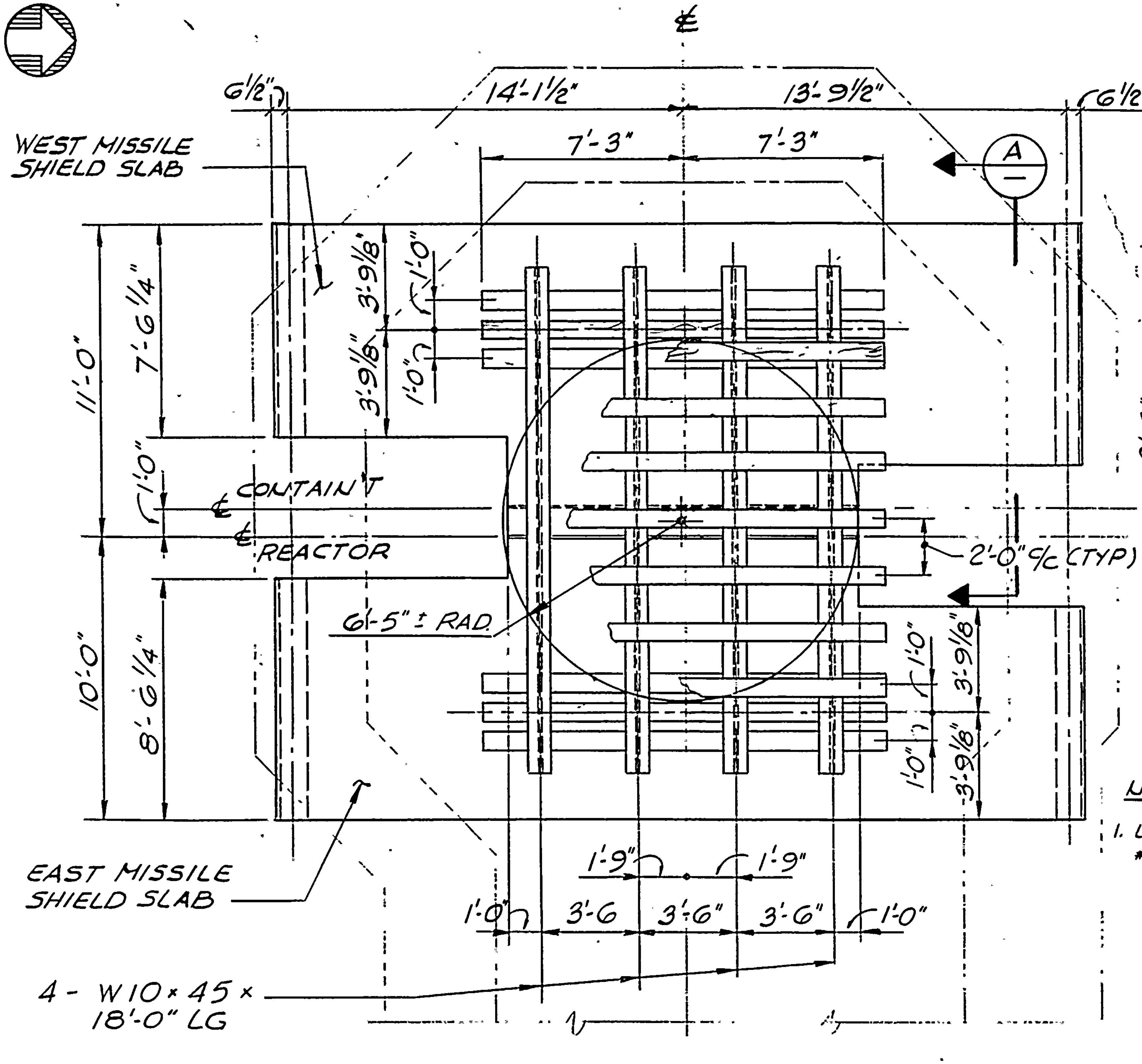


REV. 3 REMOVED UNIT 3 CULVERT
REV. 7 REVISED HAUL ROUTE

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

SITE PLAN
FIGURE 3.1-2

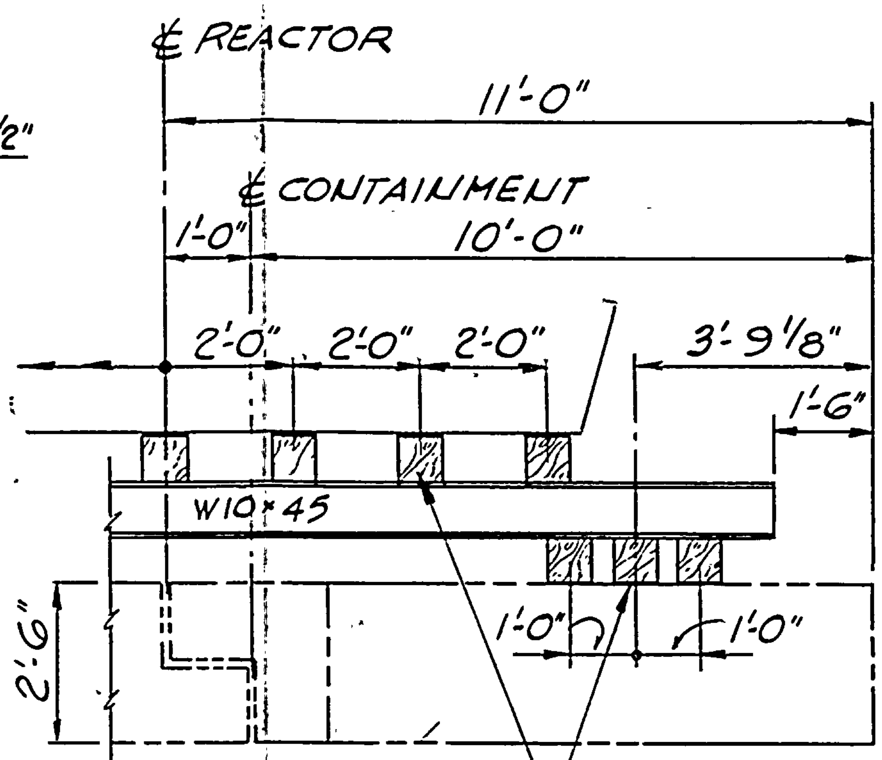
Revision 7
March 1980



STEAM GEN. LAYDOWN PLATFORM

PLAN

1/4" = 1'-0"



SECTION A

SCALE 3/8" = 1'-0"

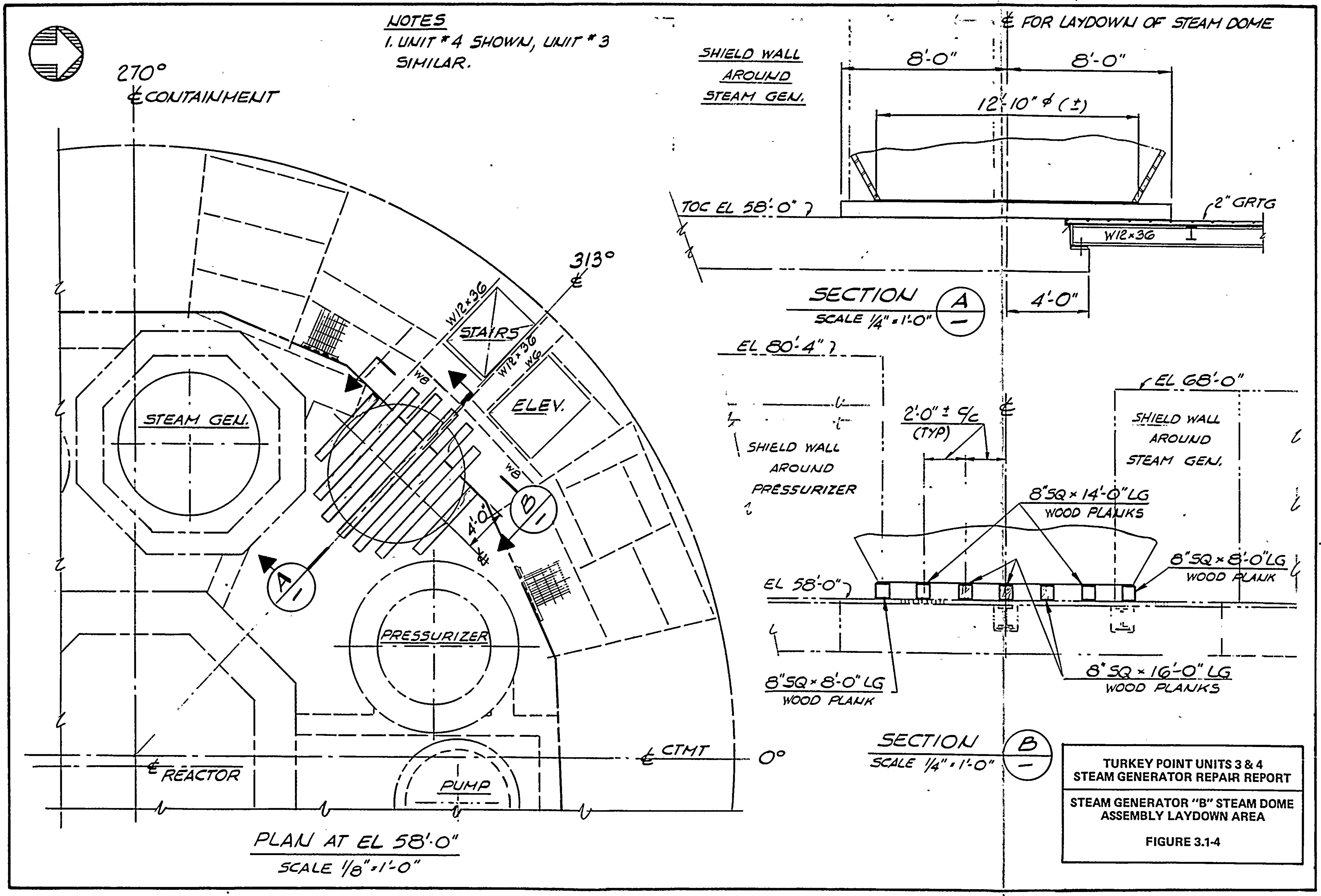
NOTES

1. UNIT #4 SHOWN; UNIT #3 SIMILAR.

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR "A" STEAM DOME ASSEMBLY LAYDOWN AREA

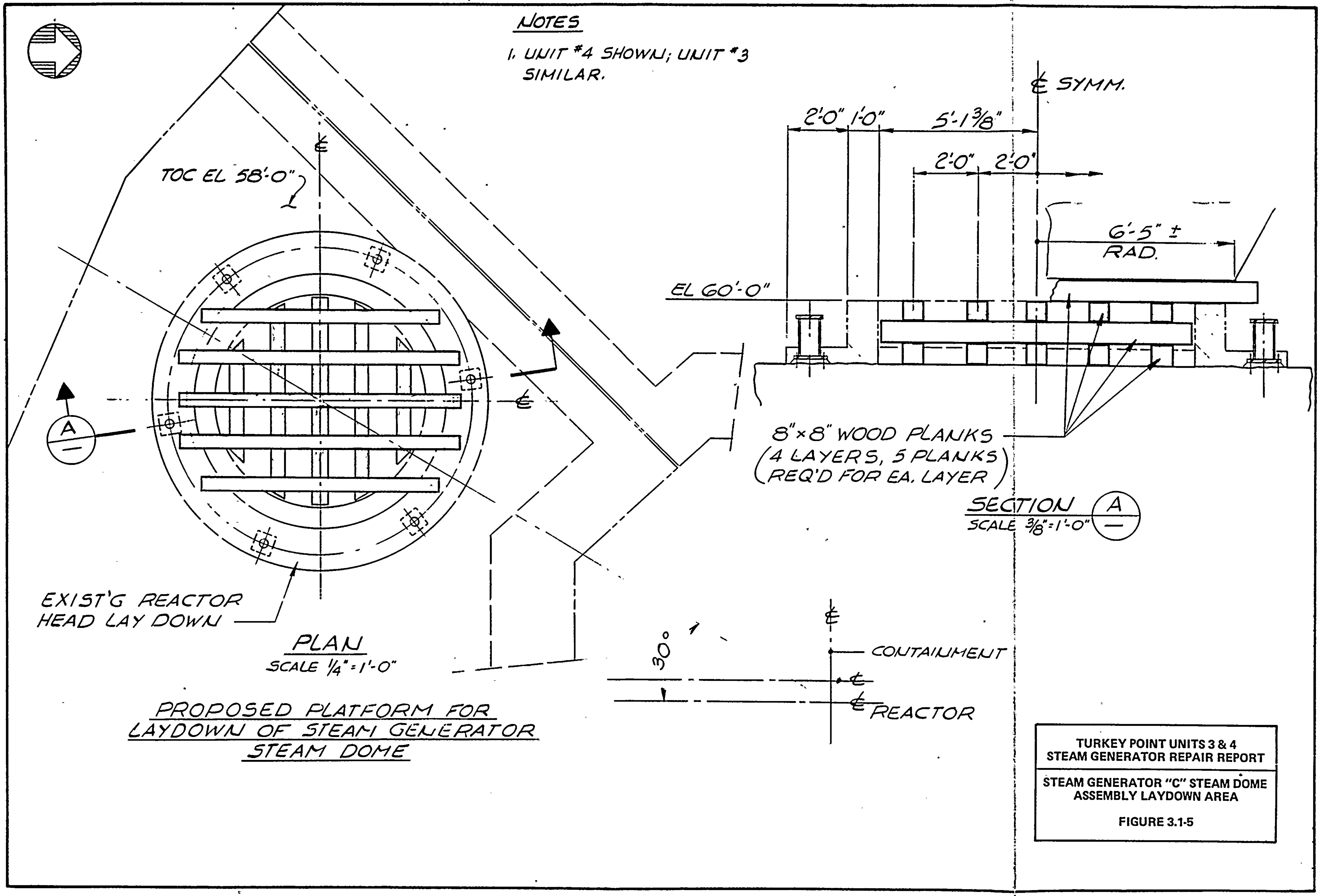
FIGURE 3.1-3



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR "B" STEAM DOME ASSEMBLY LAYDOWN AREA

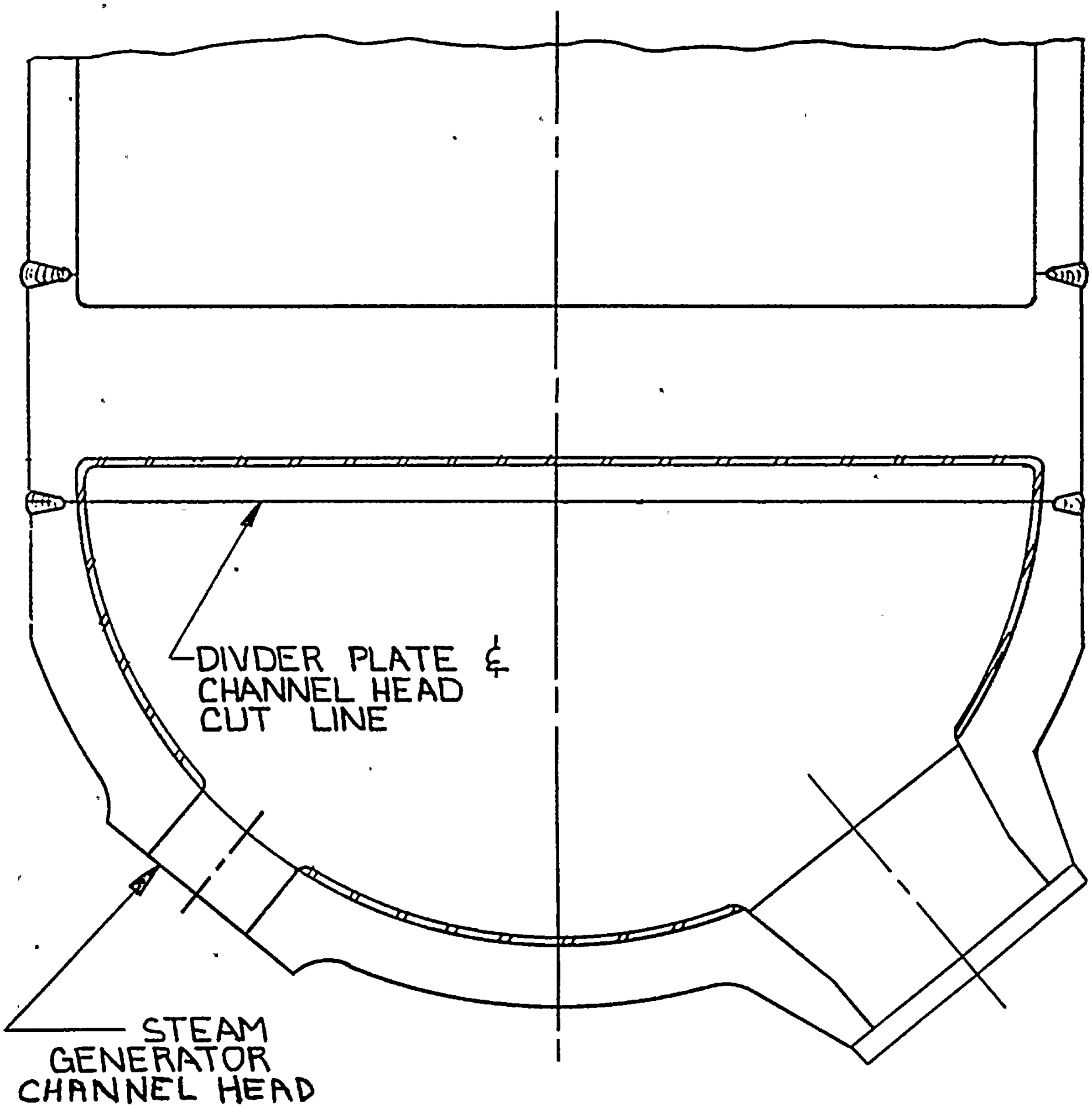
FIGURE 3.1-4



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR "C" STEAM DOME ASSEMBLY LAYDOWN AREA

FIGURE 3.1-5



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

CHANNEL HEAD CUT LOCATION
FIGURE 3.2-1

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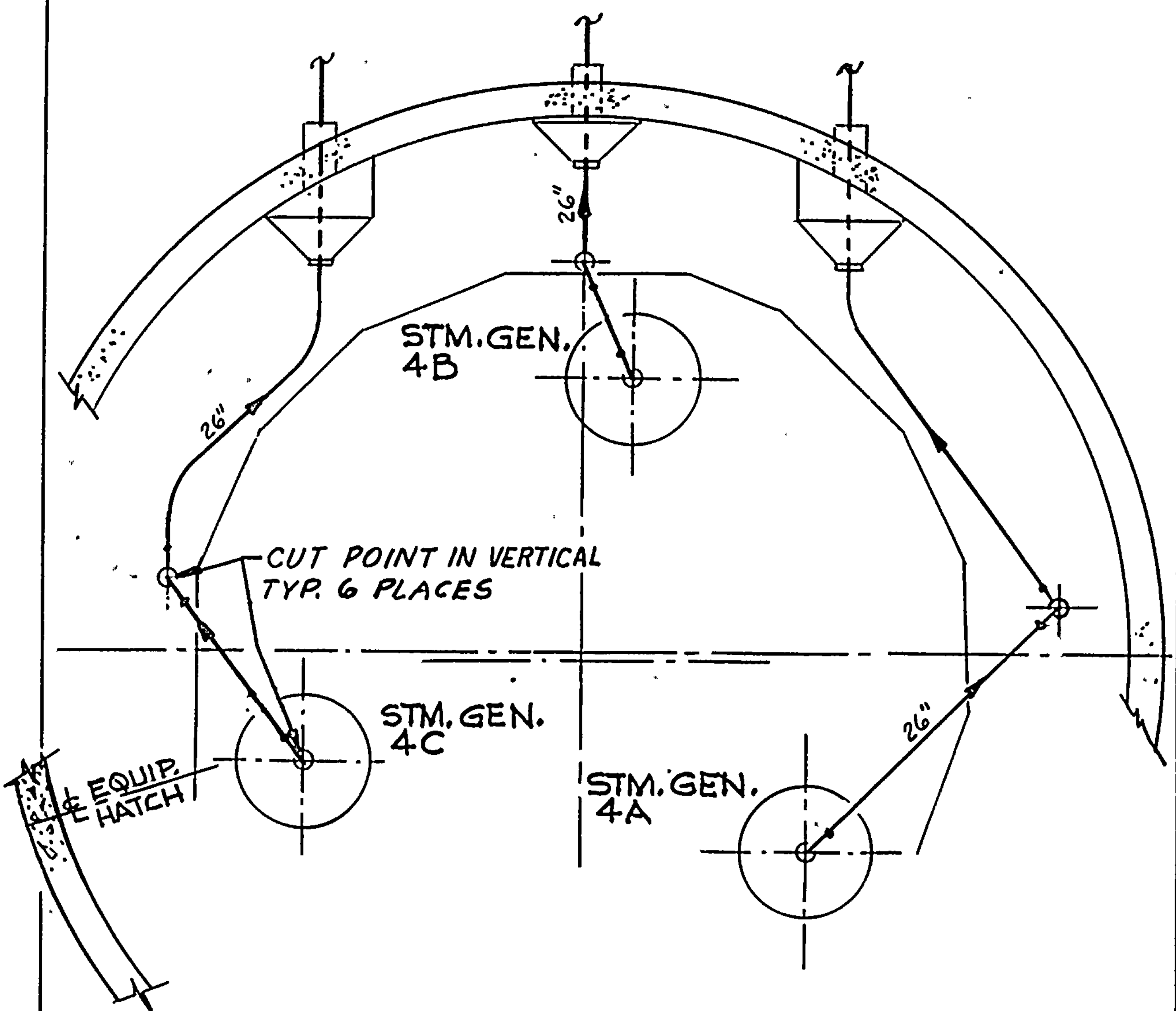

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

MAIN STEAM PIPING CUT POINTS

FIGURE 3.2-2

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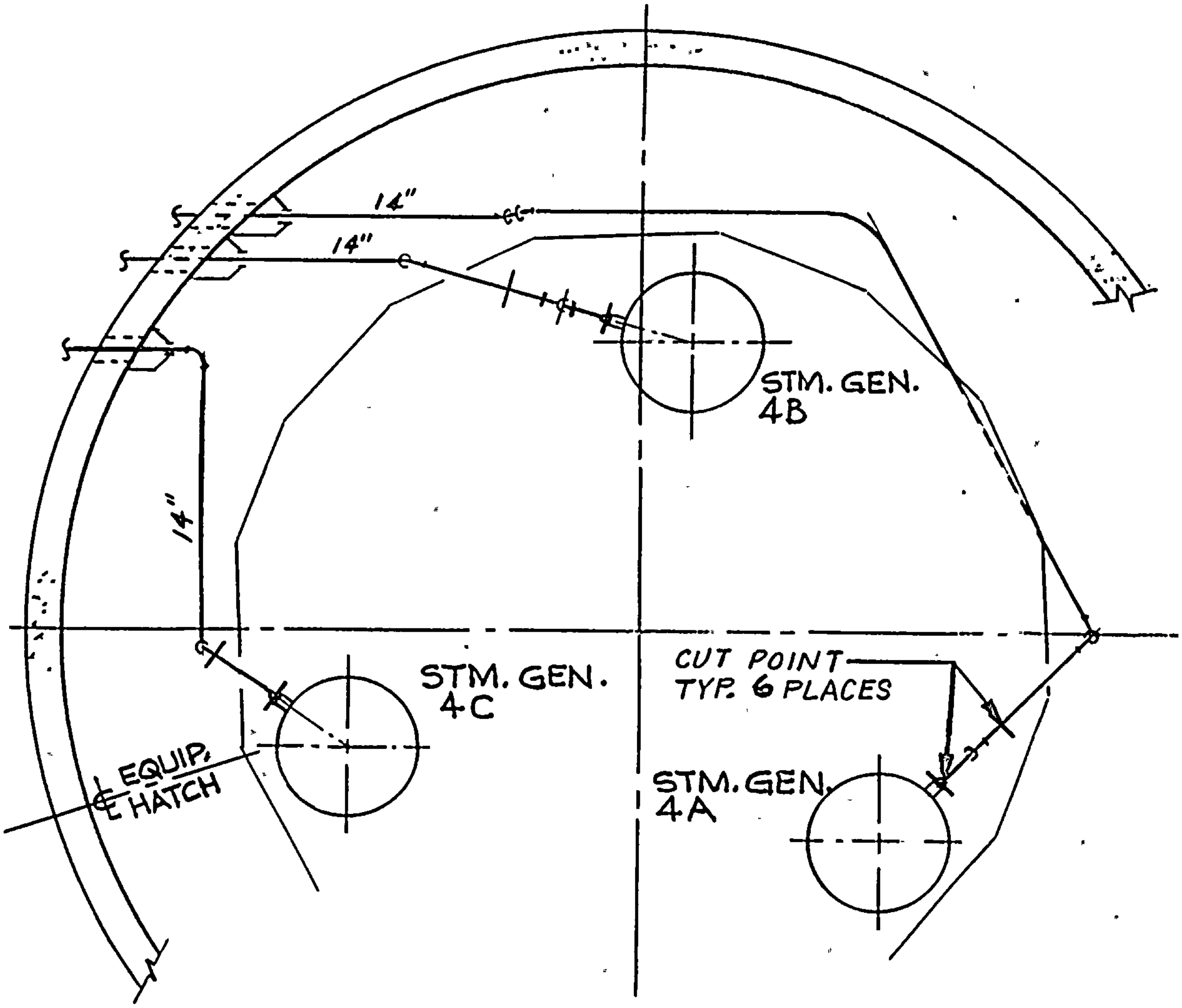

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

MAIN FEEDWATER PIPING CUT POINTS

FIGURE 3.2-3

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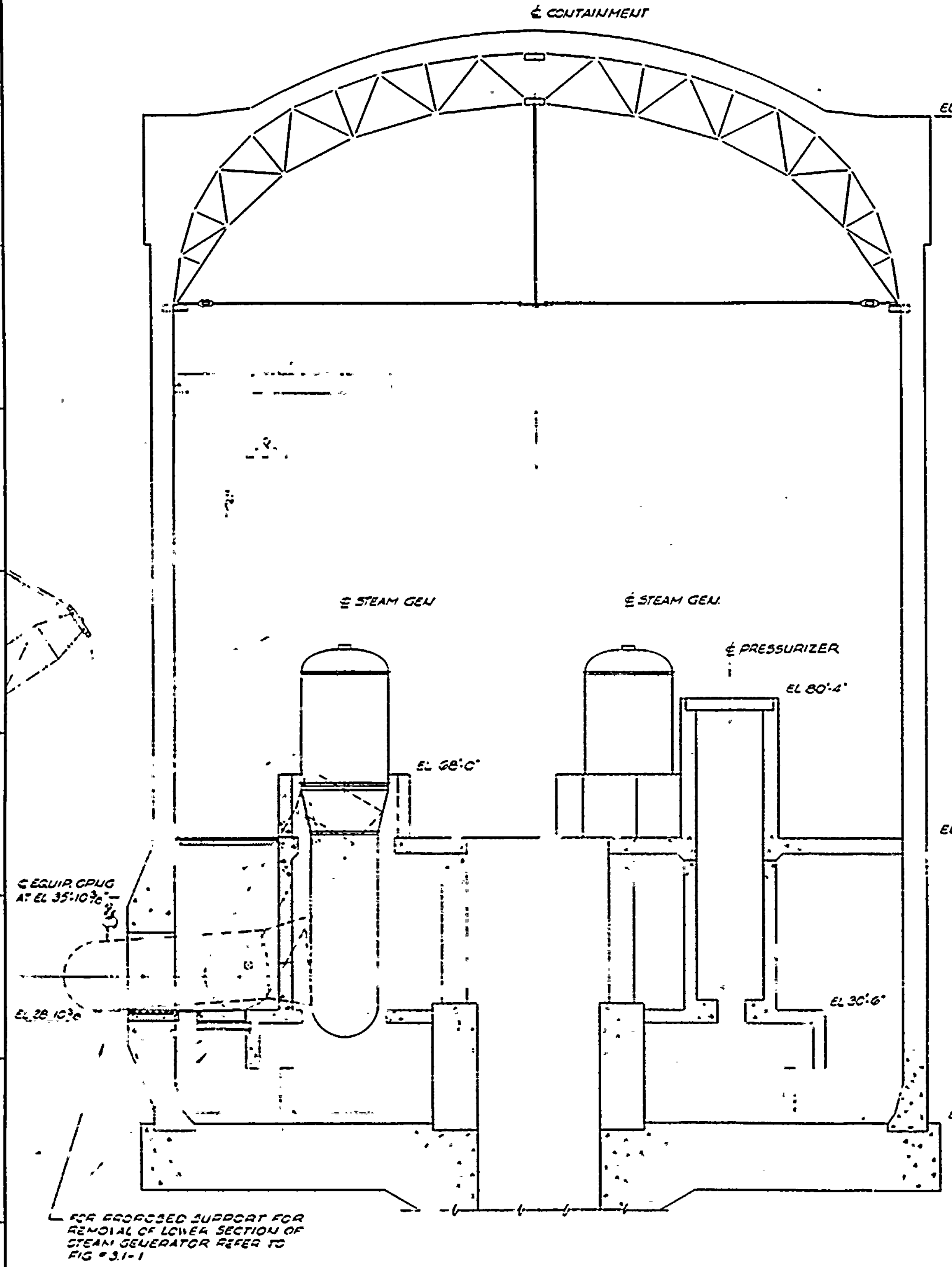
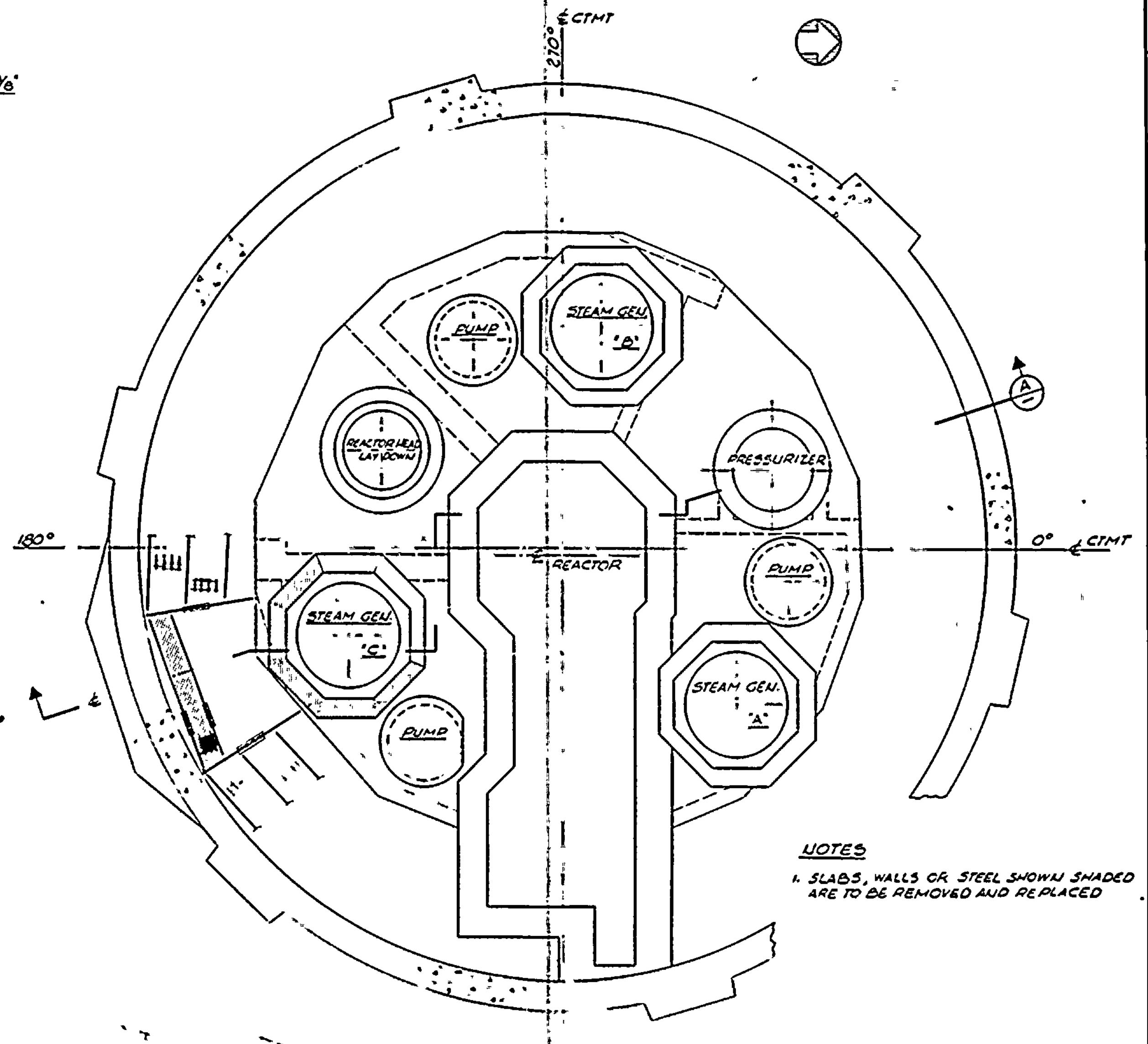
March 1980

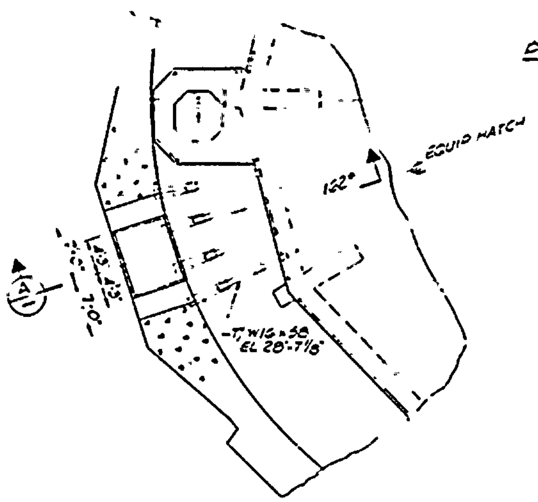


SECTION (A)
SCALE ⅛"=1'-0"



PLAN AT EL 58'-0"
SCALE ⅛"=1'-0"

NOTES
1. SLABS, WALLS OR STEEL SHOWN SHADED ARE TO BE REMOVED AND REPLACED

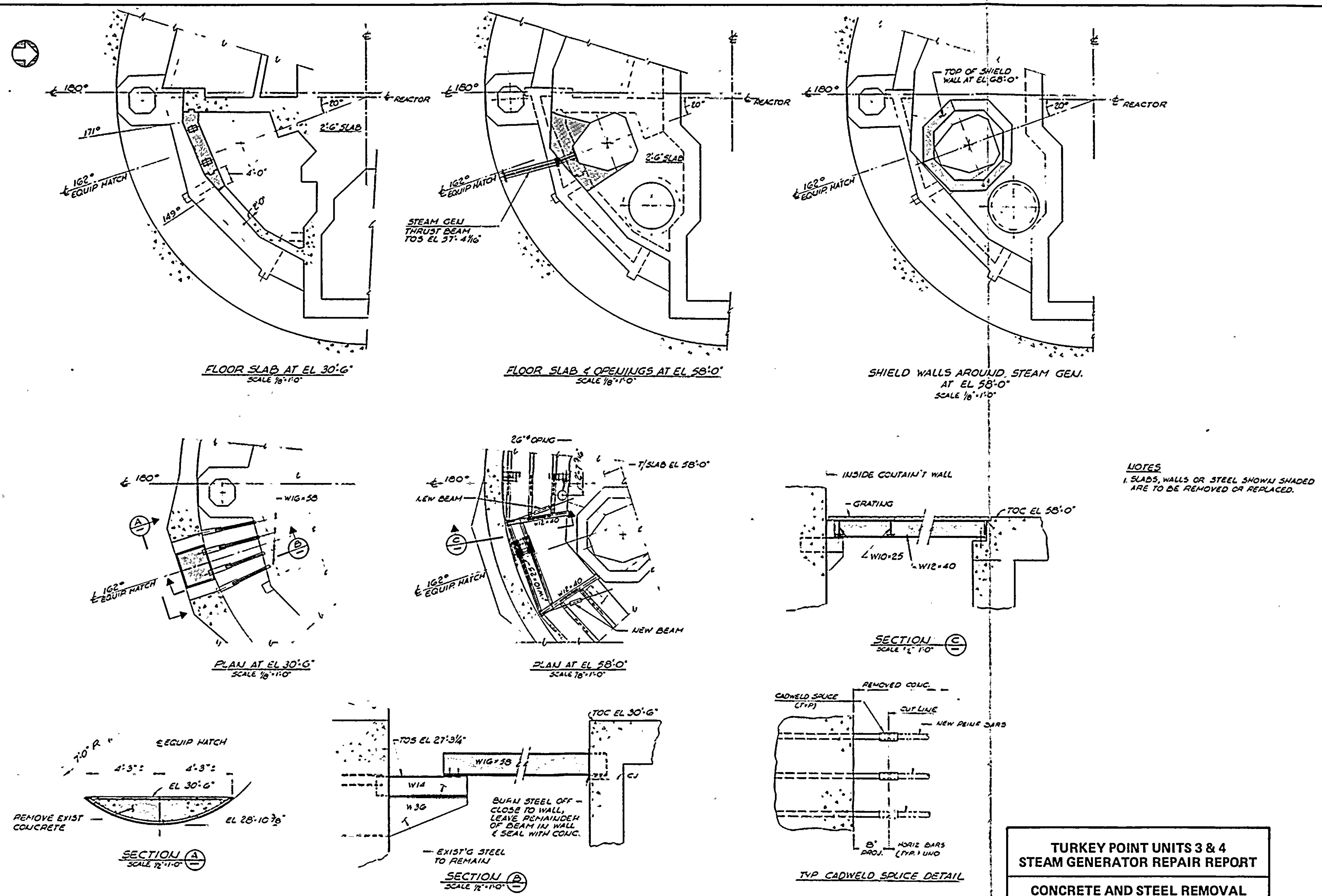


PARTIAL PLAN AT EL. 30'-6"
SCALE ⅛"=1'-0"

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

CONCRETE AND STEEL REMOVAL
UNIT 4

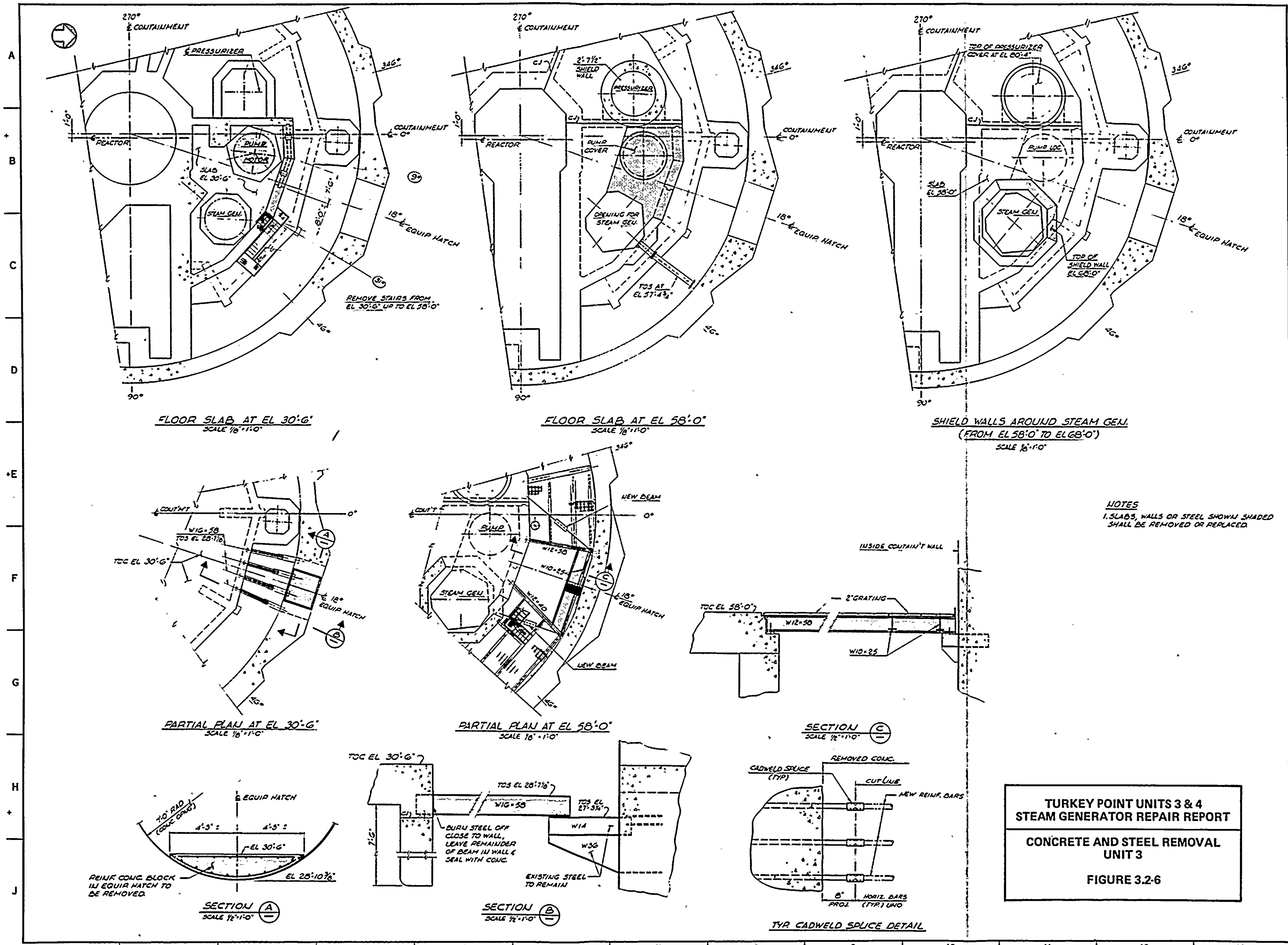
FIGURE 3.2-4



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

CONCRETE AND STEEL REMOVAL
UNIT 4

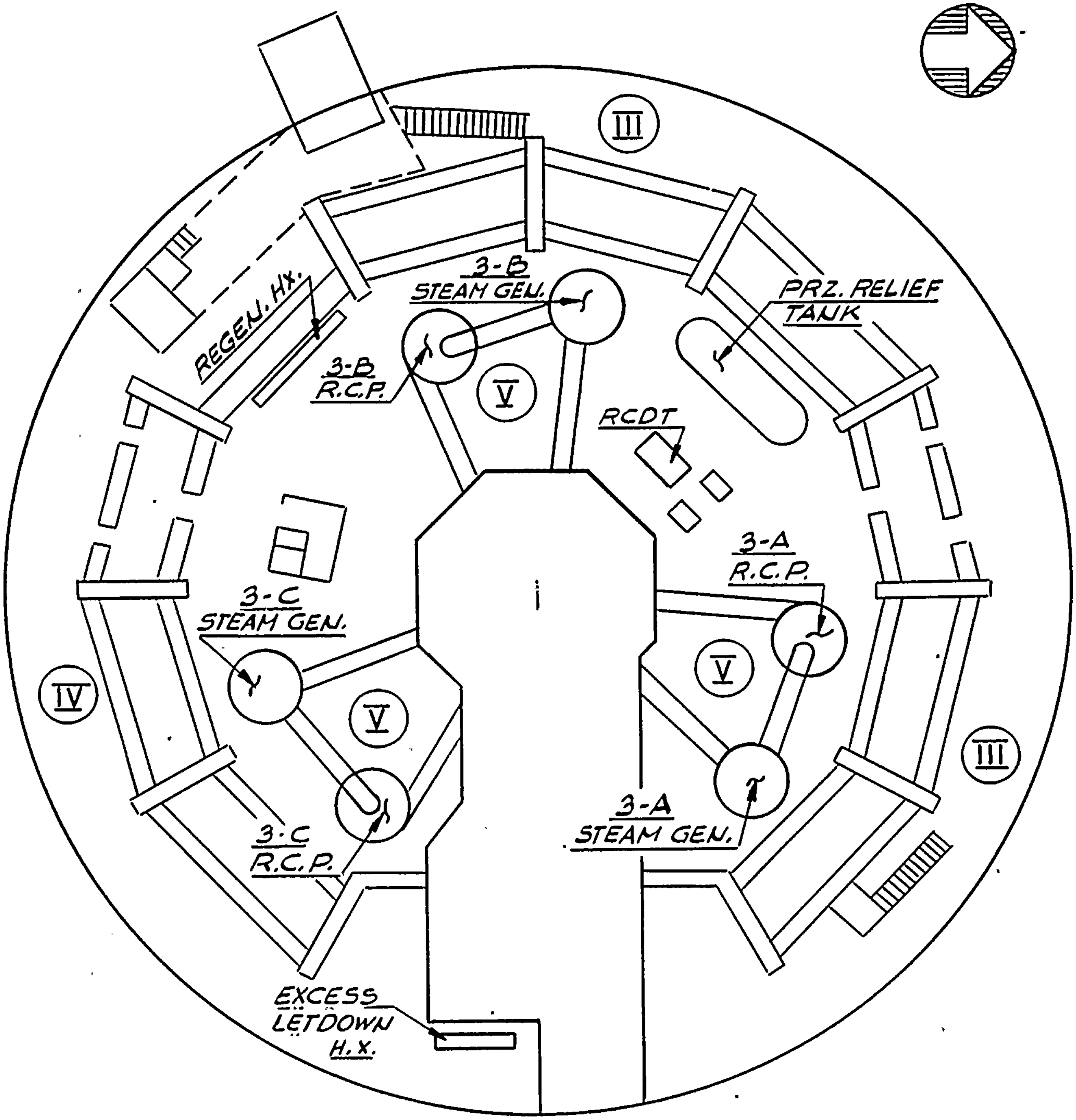
FIGURE 3.2-5



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

CONCRETE AND STEEL REMOVAL
UNIT 3

FIGURE 3.2-6



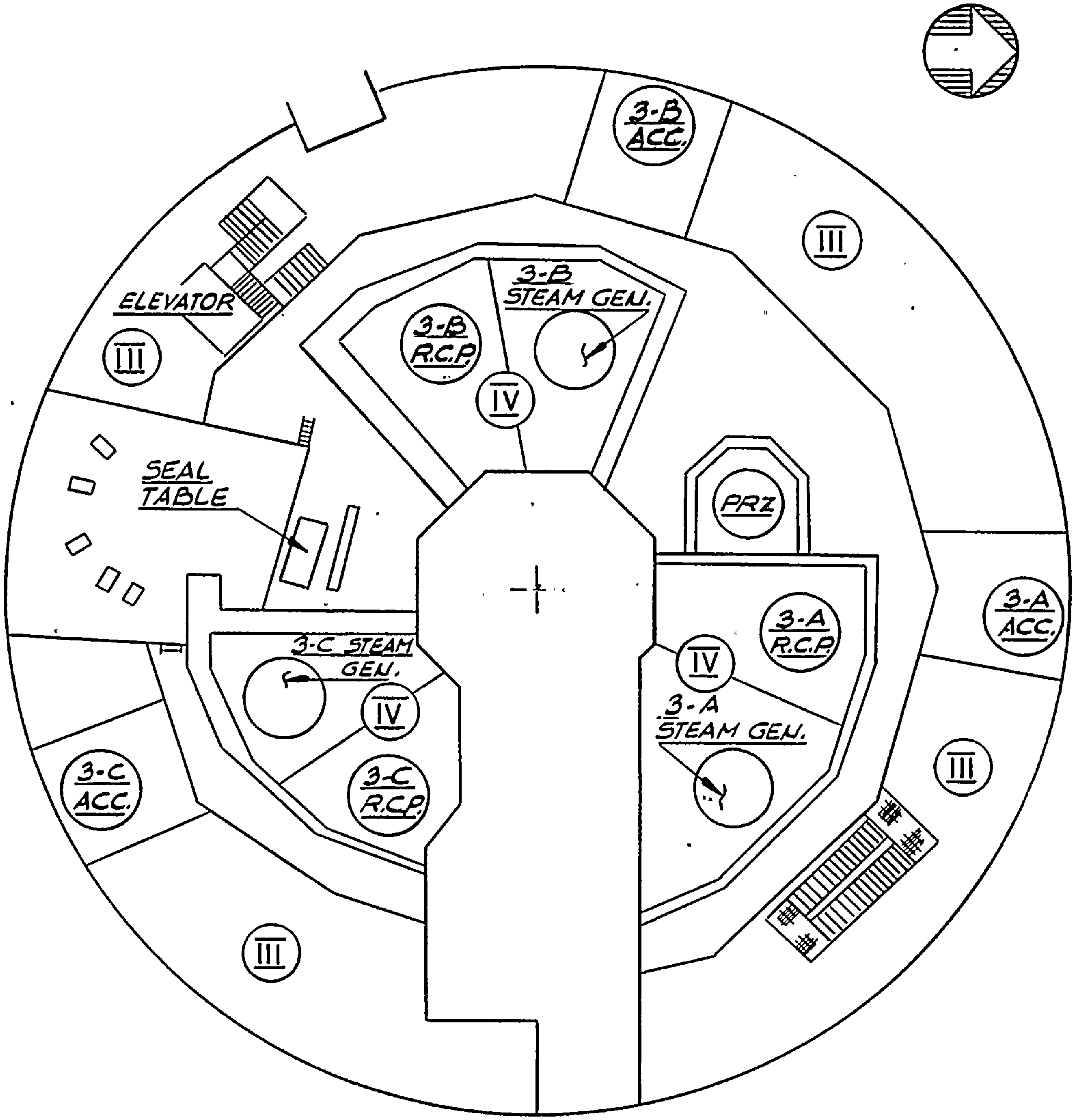
NOTES

ZONE	DOSE RATE RANGE (mrem/hr)
I	≤ 0.5
II	≤ 2.5
III	≤ 15
IV	≤ 100
V	> 100

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

RADIATION ZONE DESIGNATIONS, UNIT 3, CTMT EL. 14'-0"

FIGURE 3.3-1



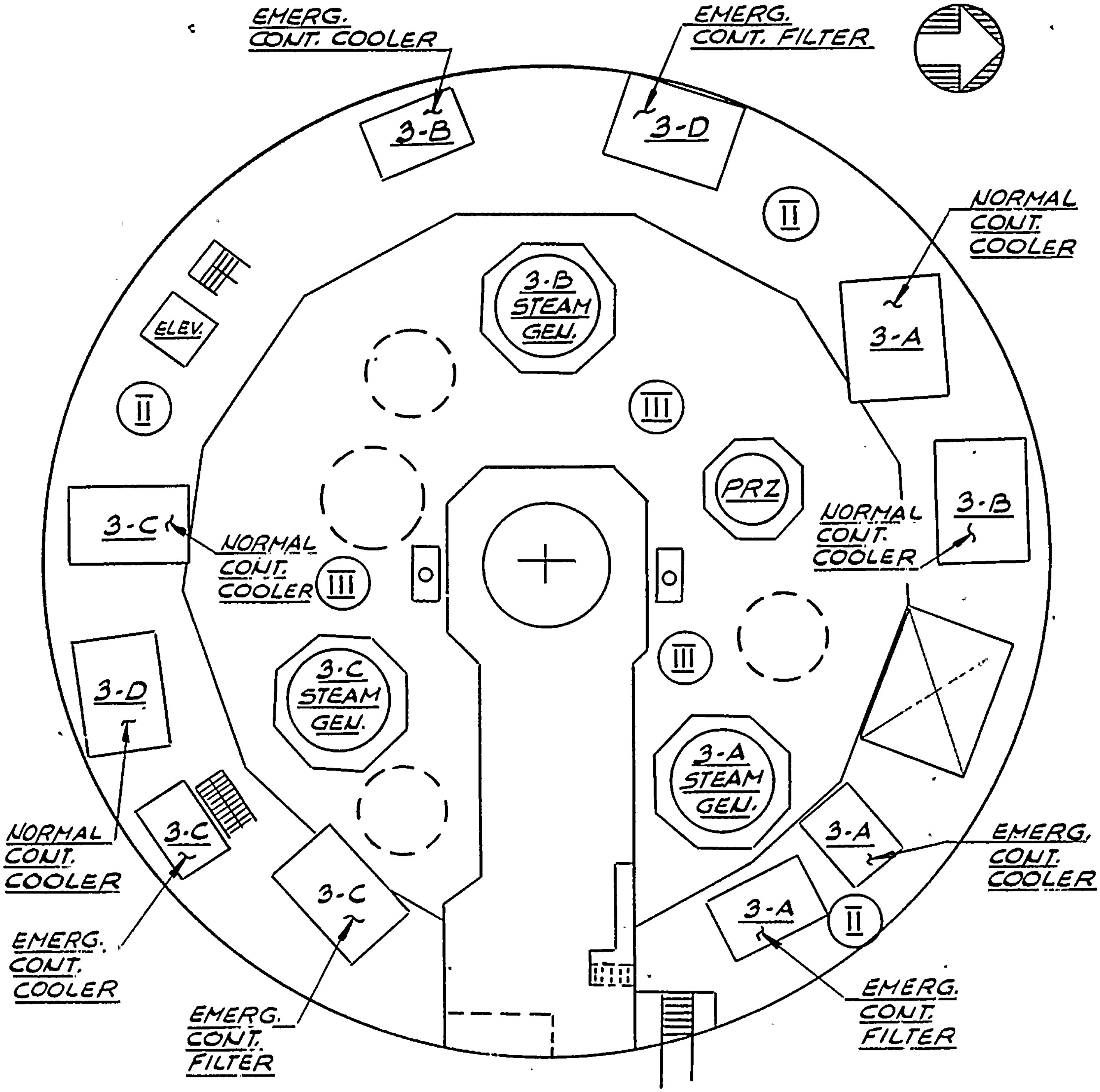
NOTES

ZONE	DOSE RATE RANGE (mrem/hr)
I	≤ 0.5
II	≤ 2.5
III	≤ 15
IV	≤ 100
V	> 100

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

RADIATION ZONE DESIGNATIONS,
UNIT 3, CTMT EL. 30'-6"

FIGURE 3.3-2



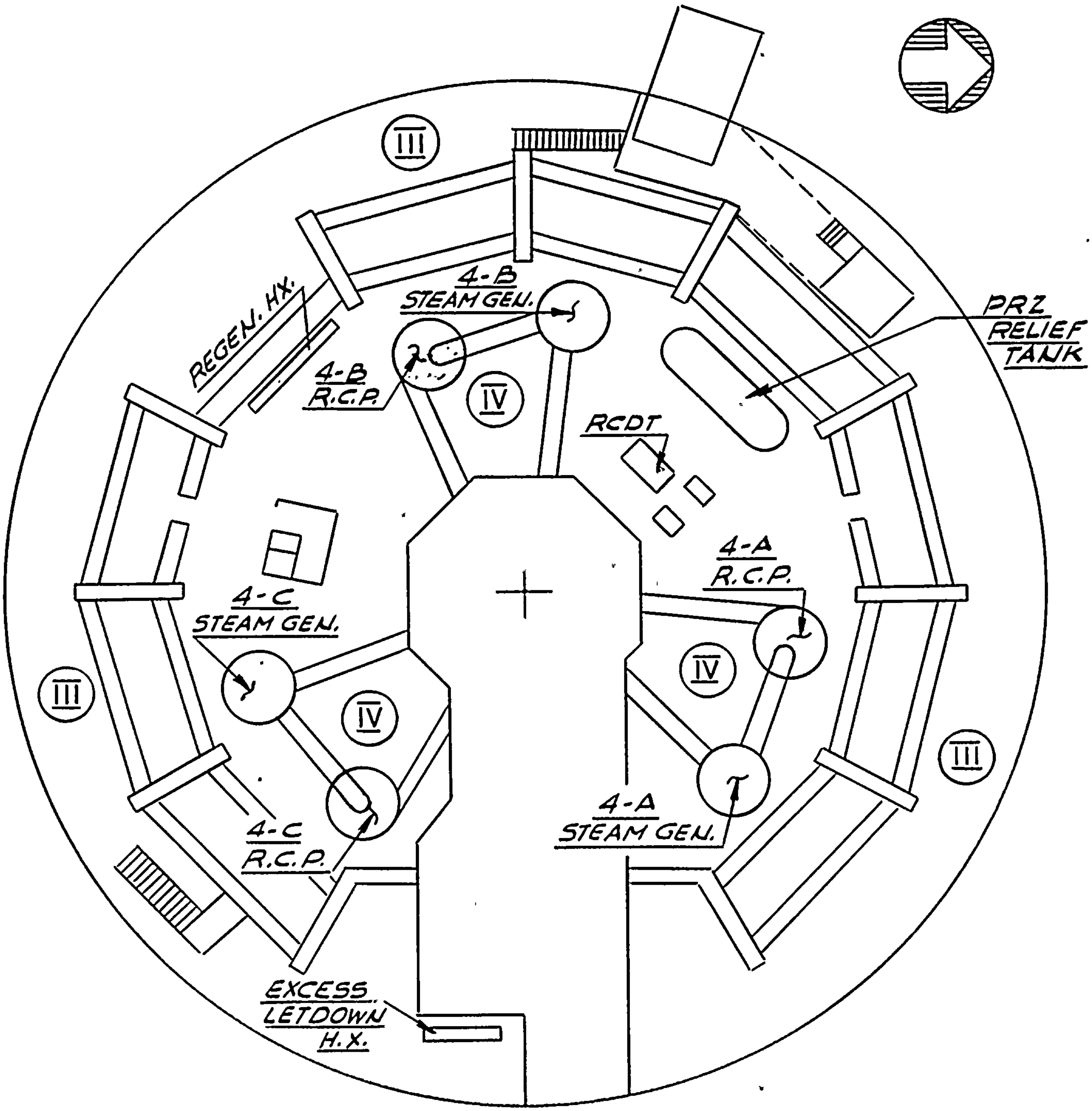
NOTES

ZONE	DOSE RATE RANGE (mrem/hr)
I	≤ 0.5
II	≤ 2.5
III	≤ 15
IV	≤ 100
V	> 100

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

RADIATION ZONE DESIGNATIONS, UNIT 3, CTMT EL. 58'-0"

FIGURE 3.3-3



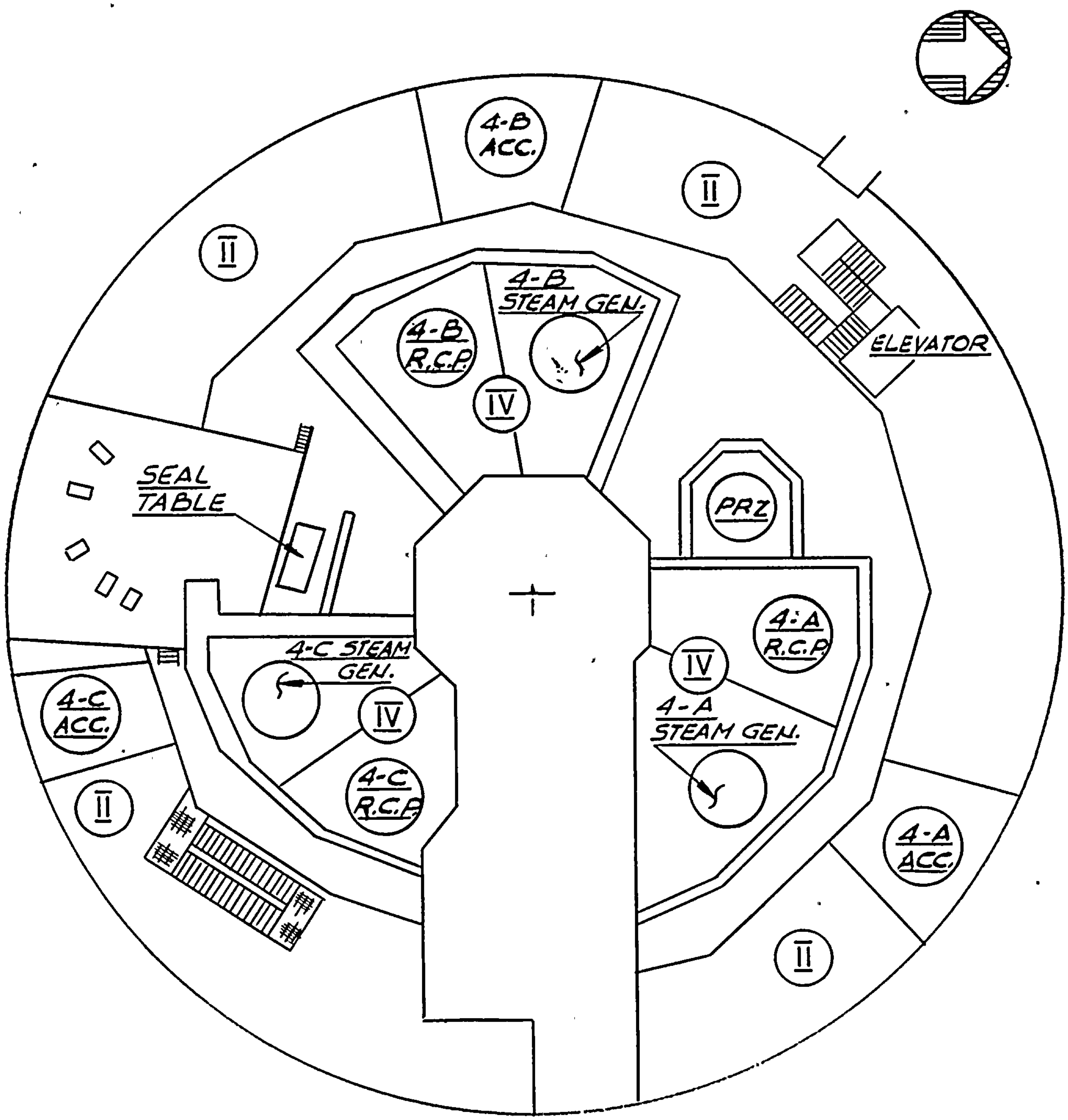
NOTES

ZONE	DOSE RATE RANGE (mrem/hr)
I	≤ 0.5
II	≤ 2.5
III	≤ 15
IV	≤ 100
V	> 100

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

RADIATION ZONE DESIGNATIONS,
UNIT 4, CTMT EL. 14'-0"

FIGURE 3.3-4



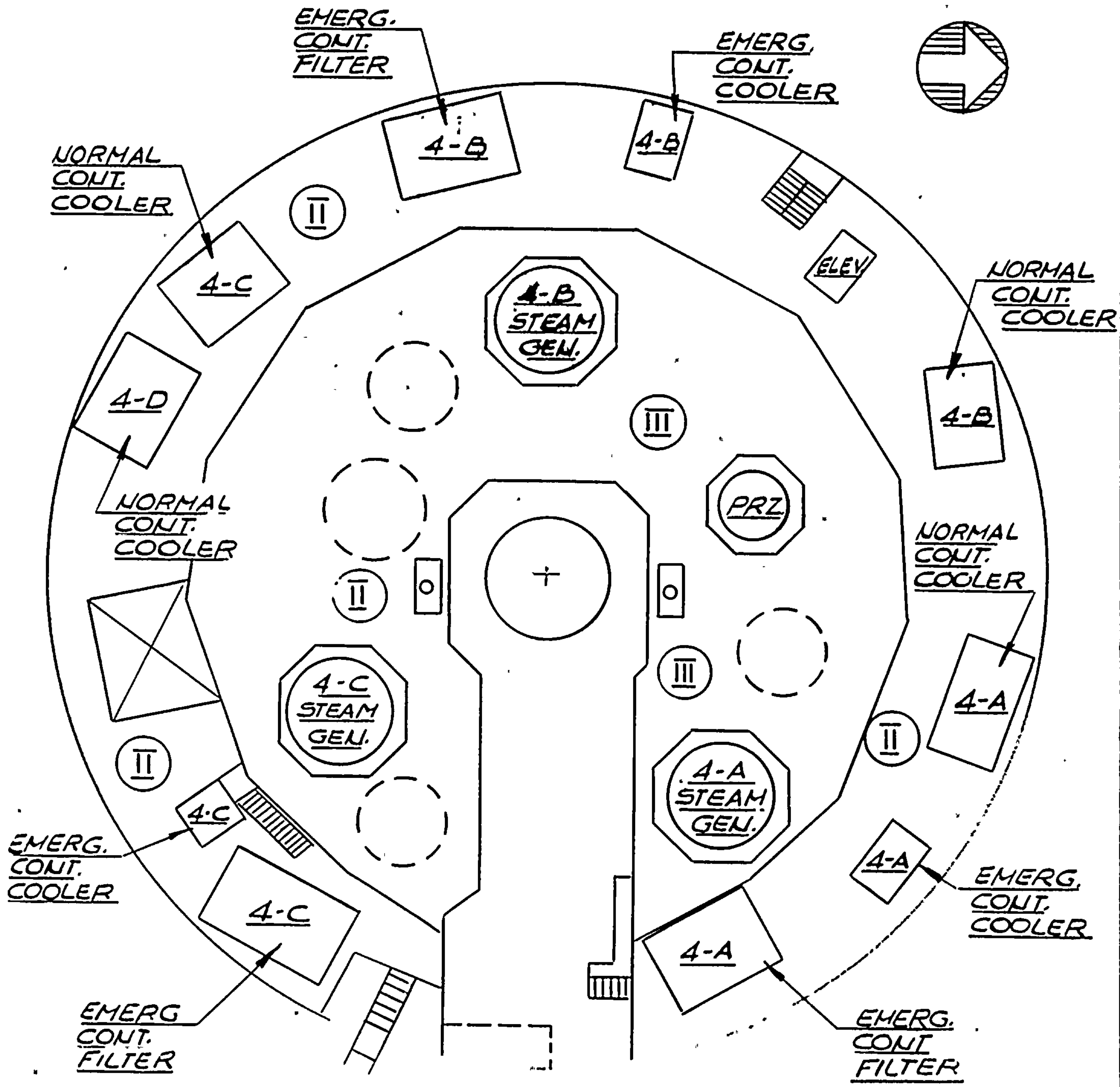
NOTES

ZONE	DOSE RATE RANGE (mrem/hr)
I	≤ 0.5
II	≤ 2.5
III	≤ 15
IV	≤ 100
V	> 100

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

RADIATION ZONE DESIGNATIONS,
UNIT 4, CTMT EL. 30'-6"

FIGURE 3.3-5



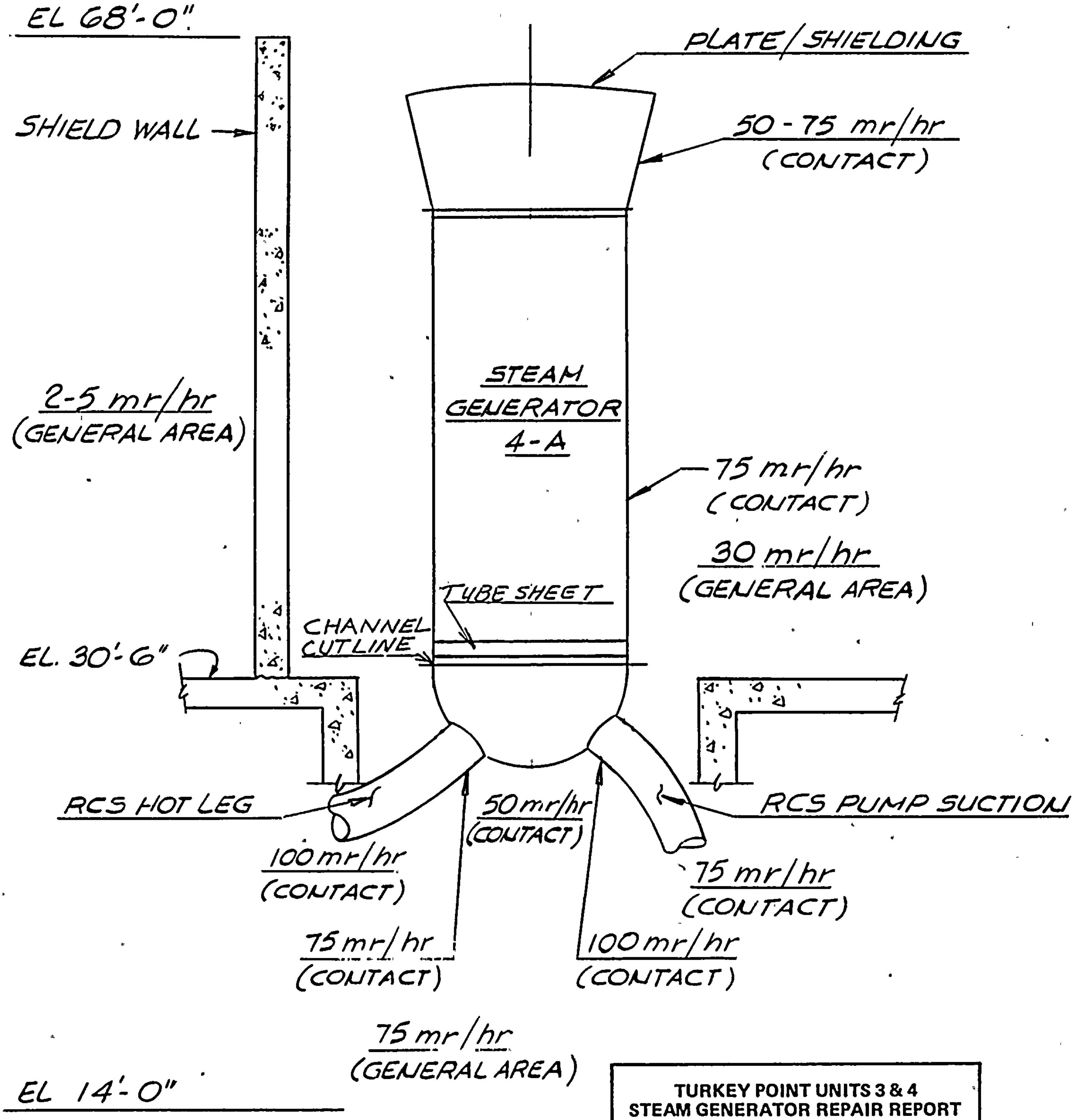
NOTES

ZONE	DOSE RATE RANGE (mrem/hr)
I	≤ 0.5
II	≤ 2.5
III	≤ 15
IV	≤ 100
V	> 100

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

RADIATION ZONE DESIGNATIONS,
UNIT 4, CTMT EL. 58'-0"

FIGURE 3.3-6



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

DOSE RATES AROUND STEAM
GENERATOR 4A

FIGURE 3.3-7 Revision 7
March 1980

4.0 RETURN-TO-SERVICE TESTING

4.1 GENERAL 7

Following steam generator repair, a preoperational testing program will be conducted as required to provide the necessary assurance that the facility can be operated in accordance with design requirements and in a manner that will not endanger the health and safety of the public.

4.1.1 Integrity of the Reactor Coolant System After Steam Generator Repair

Test requirements for the integrity of the reactor coolant system (RCS) will be performed in accordance with Technical Specification 4.3 and 10 CFR 50.55a. Integrity of the RCS will be verified in accordance with requirements of ASME Section XI, Article IWB-2000 for weldings and Article IWB-5000 for hydrostatic test. The edition and addenda to be used for all testing will be the 1977 Edition with addenda through summer 1978.

Test requirements for the integrity of Class 2 components will also be performed in accordance with 10 CFR 50.55a. Integrity of Class 2 components will be verified in accordance with the requirements of the ASME Section XI, Article IWB-2000 for weldings and article IWB-5000 for hydrostatic test. The edition and addenda to be used for all testing will be the 1977 Edition with addenda through summer 1978. 7

4.1.2 Integrity of the Reactor Containment Building

The removal of the steam generator lower assemblies from the Containment Building will be through the equipment hatch. No modifications impacting the integrity of the equipment hatch as a pressure boundary structure are contemplated at this time.

Local leak rate test of the containment will be performed before startup by Technical Specification 4.4.2. The procedures used for the local leak rate test are:

a. O.P. 13404.1 - Containment Boundary Isolation Valves - Local Leak Rate Test

b. O.P. 13404.2 - Containment Penetration Canisters - Local Leak Rate Test

c. O.P. 13513 - Personnel and Emergency Air Locks - Local Leak Rate Test

d. O.P. 13531.1 - Equipment Access Hatch - Local Leak Rate Test

5.0 SAFETY EVALUATION

5.1 FSAR EVALUATIONS

5.1.1 Introduction

The purpose of this section is to evaluate the impact, if any, of the repaired steam generators on the accident analysis transients for Turkey Point Units 3 and 4. Under the guidelines specified in 10 CFR 50.59 such an evaluation is required to verify that no unreviewed safety concerns or changes to the Technical Specifications occur. This section provides a qualitative discussion of the effect on the accident analysis of steam generator parameter changes resulting from steam generator repair. Conclusions are made in this section concerning the validity of the original FSAR to the repaired units. Consistent with the requirements of 10 CFR 50.59, licensing regulations and guidelines of the original licensing of the Turkey Point Units are assumed to apply, and only changes in the safety analyses due to equipment changes are considered.

The relevant plant operating parameters and steam generator design parameters have been compared in Table 5.1-1 and Table 5.1-2, respectively, for the original and repaired steam generators. While incorporating design improvements that will improve the flow distribution and tube bundle accessibility and reduce secondary side corrosion, the repaired steam generators continue to match the design performance of the original steam generators. It may be noted from Tables 5.1-1 and 5.1-2 that there is very little change in plant operating parameters in repairing the steam generators. It is, therefore, to be anticipated that the impact on the accident analyses will be insignificant.
The results of the accident evaluation show that the repair of steam generators resulting in physically and functionally similar units has not resulted in any adverse changes in the plant operating conditions used in the FSAR, and, therefore, the analyses presented in the FSAR are still valid. This section establishes that no unreviewed safety concerns exist due to operation with the repaired Turkey Point steam generators.

5.1.2 Non-LOCA Accidents

The Turkey Point FSAR includes analyses of fifteen non-LOCA accidents in Sections 14.1 and 14.2. These are:

a. Uncontrolled rod cluster control assembly (RCCA) withdrawal from a subcritical condition

b. Uncontrolled RCCA withdrawal at power

c. Malpositioning of part length rods

d. RCCA drop

e. RCCA ejection

f. Loss of reactor coolant flow

g. Excessive load increase incident

h. Chemical and volume control system malfunction

i. Startup of an inactive reactor coolant loop

j. Reduction in feedwater enthalpy incident

k. Loss of external electrical load

l. Loss of normal feedwater

m. Loss of all ac power to the station auxiliaries (blackout)

n. Turbine generator design analysis

o. Rupture of a main steam pipe (steam break)

The impact of the secondary system on the results of transients a. through e. is of no consequence since constant heat extraction is still maintained with the repaired steam generators. The main reason for this is that the nuclear and thermal time constants of the fuel are much smaller than the fluid mixing and transport time, the latter being mechanisms responsible for secondary to primary interaction. For the rod withdrawal and rod ejection accidents reactor trip occurs at a time near the magnitude of the coolant loop transport time. The limiting consideration for the rod drop accident and malpositioning of part length rods is the neutron flux redistribution resulting from the control rod movement and is clearly not coupled to steam generator performance. It can be validly concluded, therefore, that the first five accident transients named above are unaffected by the repair of the Turkey Point steam generators.

Loss of reactor flow transients can be discussed collectively as was the case for the five reactivity insertion accidents above. Included in this general category are the following:

a. Total loss of reactor coolant flow

b. Partial loss of reactor coolant flow

c. Locked rotor

If the reactor is at power, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature, which could result in departure from nucleate boiling (DNB). However, the reactor is tripped on low frequency, low voltage or low coolant flow trips such that the consequences are within the bounds of the FSAR analyses. The low flow protection system, consisting of the low voltage, low frequency and low flow trips, rapidly detects and protects against loss of coolant flow events. Changes in coolant temperature due to secondary parameter changes would not be detected in the core during the time frame of interest to this transient. Therefore it can be concluded that the repair of the steam generator will not affect the loss of coolant flow transients.

The chemical and volume control system malfunction is a boron dilution in the reactor coolant system caused by adding unborated water to the reactor coolant via the makeup control system. Factors to be considered in the analysis of this transient are that the maximum dilution rate depends on the charging pump characteristics and primary volume, also that the malfunction must be recognized and terminated by operator action. The small reduction in primary side steam generator volume will result in a negligible change of the operator action times originally reported in the FSAR. Repair of the steam generators with physically and functionally similar units will neither affect 5 the initiating circumstances nor the corrective actions for the chemical and volume control system malfunction.

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The accident is analyzed in the FSAR to show that a 10 percent increase in steam flow from full power can be accommodated without a reactor trip. If a 10 percent step load increase is postulated, feed flow will increase to match steam flow and maintain steam generator level. Depending on whether or not automatic rod control is available, a new steady state condition is established at the initial coolant average temperature or at a lower coolant average temperature. As is evident from the over-temperature T equation, more than 10 percent power margin in DNB is available. Repair of the steam generators resulting in units of similar physical size and tube structure could slightly affect the excessive load increase accident in that the higher (by about 5 percent) full power fluid inventory of the repaired steam generators could cause the transient to progress more slowly; however, the same endpoint equilibrium condition would still be eventually reached, since no reactor trips are encountered.

The turbine generator design analysis describes the turbine generator and its speed control and provides a discussion concerning the velocity and energy of postulated ejected parts from the turbine. This analysis is completely independent of the nuclear steam supply system and thus is not affected by the repaired steam generators.

It is apparent, therefore, that only those accidents which involve a primary-secondary interaction could potentially be affected by steam generator repair. Since the remaining accidents on the above list are generally concerned with primary coolant heatup or cooldown resulting from loss of secondary heat sink or excessive heat removal from the secondary side, they could potentially be affected by changes resulting from steam generator repair. These accidents are evaluated individually in the following sections.

5.1.2.1 Startup of an Inactive Reactor Coolant Loop

The cold leg temperature in the inactive loop will be identical to the cold leg temperatures of the active loops and to the reactor core inlet temperature. If the reactor is operated at power, there is a temperature drop across the steam generator in the inactive loop, and reverse flow would exist through the inactive loop thereby lowering the hot leg temperature of that loop below core inlet temperature. Administrative procedures require that the plant be brought to less than 25 percent load level and approximate temperature equilibrium between loops prior to starting the pump in the inactive loop.

The startup of an inactive reactor coolant loop accident occurs when a coolant pump in a loop, which contains water at a lower temperature than active loops, is started, causing a significant increase in water flow into the core. The decrease in core temperature due to the increase in flow and the injection of colder water causes a rapid core power increase due primarily to moderator reactivity feedback. Verification that safety criteria for this accident are not violated is accomplished by demonstrating that the DNB ratio is always greater than 1.30.

The analysis presented in the Turkey Point FSAR assumed that the inactive loop flow reversed and accelerated to its nominal full flow value instantaneously. The reactor coolant in that part of the inactive loop from the steam generator plenum to the reactor exit plenum (normally the hot leg) was assumed to be at a temperature equal to the saturation temperature of the secondary side. This assumption is independent of the heat transfer characteristics of the steam generator, and thus is not affected by the repaired steam generators. The primary side volume reduction for the repaired steam generators, would insig-
nificantly reduce the duration of the cold water slug (currently 15 seconds).
The delay for the slug to reach to core inlet (4 seconds) would remain un- 5
changed from the FSAR analysis. Therefore, the transient results presented in the FSAR would be essentially the same, and the accident criteria would continue to be met with the repaired steam generators.

5.1.2.2 Reduction in Feedwater Enthalpy Incident

The reduction in feedwater enthalpy is another means of increasing core power above full power. Such increases are attenuated by the thermal capacity in the secondary plant and in the reactor coolant system. The overpower-overtemperature protection (nuclear overpower and T trips) prevents any power increase which could lead to a DNBR less than 1.30.

An extreme example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the feedwater bypass valve which diverts flow around the low pressure feedwater heaters.

The function of the bypass valve is to maintain net positive suction head on the main feedwater pump in the event heater drain pump flow is lost, e.g., during a large sudden load decrease. In the event of an accidental opening of the feedwater bypass valve, flow is diverted around the low pressure feedwater heaters. This causes a sudden reduction in inlet feedwater temperature to the steam generators. This increased subcooling will create a greater load demand on the primary system which can lead to a reactor trip.

Two cases are analyzed to demonstrate the unit behavior in the event of a sudden feedwater temperature reduction resulting from accidental opening of the feedwater bypass valve. The first case is for the reactor in manual control with a zero moderator coefficient since this represents a condition where the unit has the least inherent transient capability. The second case is for the reactor in automatic control with a large negative moderator coefficient. Initial pressure coolant temperature and power conditions are assumed at extreme values consistent with steady state operation to allow for calibration and instrument errors. This results in minimum margin to core DNB limit at the start of the transient.

During the accidental opening of a feedwater bypass valve transient, the secondary heat extraction is greater than the core power generation. This causes the pressurizer pressure and coolant average temperature to decrease. Without automatic reactor control and a zero moderator coefficient of reactivity, the core power level increases slowly and eventually comes to equilibrium at a slightly higher power level. With automatic reactor control and a large negative moderator coefficient the negative coefficient causes the core power to increase rapidly. Steady state conditions are reached at a higher power level.

The analysis presented in the Turkey Point FSAR for the accidental opening of a feedwater bypass valve without reactor control and zero moderator coefficient shows that an equilibrium condition is reached without generating a reactor trip and the DNB ratio does not decrease below the initial value. For the case with automatic rod control and negative moderator coefficient the FSAR analysis shows that no reactor trip is generated and the DNB ratio remains well above 1.30. The replacement steam generator design has a larger full load steam generator mass, approximately 5 percent larger. This increased secondary side heat capacity would result in a slightly slower cooldown rate than in the FSAR analysis. The steady state conditions of the FSAR analysis would be reached at a slower rate. The same margins to reactor trip will exist. Therefore, the accident criteria would still be met with the repaired steam generators.

5.1.2.3 Loss of External Electrical Load

A loss of external electrical load may result from:

a. Abnormal variation in network frequency or other adverse network operating conditions

b. Trip of the generator or opening of the main breaker from the generator with failure of turbine trip. In this case the action of the turbine control system causes a large nuclear steam supply system load reduction

c. Trip of the turbine

The unit is designed to accept a step loss of load from 100 percent to 50 percent load without actuating a reactor trip. The automatic turbine bypass system, with 40 percent steam dump capacity to the condenser, is able to accomodate this load rejection by reducing the severity of the transient imposed upon the reactor coolant system. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift for the 50 percent step loss of load with steam dump.

In the event the turbine bypass valves fail to open following a large load loss or in the event of a complete loss of load with steam dump operating, the steam generator safety valves may lift. The reactor coolant temperature will increase rapidly and the DNB limit will be approached. The reactor is tripped on the following signals:

a. High pressurizer pressure signal

b. High pressurizer level signal

c. Overtemperature ΔT signal

The steam generator shell side pressure will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the reactor coolant system and steam generator against overpressure for all load losses without assuming the availability of the turbine bypass system. The steam dump valves will not be opened for load reductions of 10 percent or less. For larger load reductions, they may be open.

The most likely source of a complete loss of load on the nuclear steam supply system is a trip of the turbine-generator. In this case, there is a direct reactor trip signal (unless below approximately 10 percent power) derived from either the turbine autostop oil pressure or a closure of the turbine stop valves. Reactor coolant temperatures and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system are functioning properly. However, in the Turkey Point FSAR the behavior of the unit is analyzed for a complete loss of load from 102 percent of full power without a direct reactor trip due to a turbine trip primarily to show the adequacy of the pressure relieving devices and also to show that no core damage occurs. The reactor coolant system and main steam system pressure relieving capacities are designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or turbine bypass control systems.

In the Turkey Point FSAR, the following cases are analyzed for the loss of external electrical load accident:

a. BOL with pressure control and automatic rod control

b. EOL with pressure control and automatic rod control

c. BOL without reactor control and pressure control

d. EOL without reactor control and pressure control

It is shown in the FSAR that the accident criteria on system pressure and DNB are not violated in any of the loss of load cases.

The increase in full load mass of the repaired steam generators would provide additional heat capacity and reduce the heatup rate. Thus the conclusions of the FSAR remain valid.

5.1.2.4 Loss of Normal Feedwater

A loss of normal feedwater (from a pipe break, pump failures, valve malfunctions, or loss of outside ac power) could conceivably result in a loss in capability of the secondary system to remove the heat generated in the reactor core. Since the plant is tripped well before the steam generator heat

transfer capability is reduced and auxiliary feedwater flow initiated, the primary system variables never approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

a. Reactor trip on low-low water level in any steam generator

b. Reactor trip on main steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator

c. Three turbine driven auxiliary feedwater pumps (600 gpm each shared by Units 3 and 4) which are started automatically on:

   1. Low-low level in any steam generator, or

   2. Opening of both feedwater pump circuit breakers, or

   3. Any safety injection signal, or

   4. Loss of voltage on both 4 kV busses, or

   5. Manual

On the loss of normal feedwater transient following the reactor and turbine trip, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Making conservative assumptions on steam generator water level at time of trip, residual heat generation in core, number of auxiliary feedwater pumps available and reactor coolant flow (natural circulation flow is assumed), the FSAR analysis demonstrates the adequacy of the auxiliary feedwater system to remove stored and residual heat without water relief from the primary system.

The loss of normal feedwater is a loss of heat sink accident. The increased steam generator mass at full load of the repaired steam generators is a change in a favorable direction. The physical dimensions of the steam generator have not changed. Therefore, the conclusion that the tubesheet in the steam generators receiving auxiliary feedwater will always be covered and adequate heat transfer capability will be maintained remains valid.

5.1.2.5 Loss of All AC Power to the Station Auxiliaries (Blackout)

The loss of ac power to the station auxiliaries is analyzed to demonstrate long term heat removal capability by auxiliary feedwater and natural circulation reactor coolant flow.

In the event of a complete loss of offsite power and a turbine trip there will be a loss of power to the station auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. After a loss of ac power with turbine and reactor trip, the following events would occur:

a. Reactor coolant flow would coast down to natural circulation flow-rates. Main feedwater flow would stop and auxiliary feed pumps would automatically start.

b. The rise in steam system pressure following the trip would automatically open the steam system power operated relief valves. (If the condenser is not available, the steam will be vented to the atmosphere.) If the steam flow rate through the power relief valves is not adequate, the steam generator self-actuated safety valves would lift to dissipate the sensible heat of the fuel and coolant above no-load temperature plus the residual heat produced in the reactor.

c. As the no load temperature is approached, the steam system power relief valves are used to dissipate the residual heat and to maintain the plant at the hot shutdown condition.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. In the Turkey Point repaired steam generator design, the average tube height is unchanged from the existing design. The nominal pressure drop in the steam generator has decreased, reflecting a smaller frictional pressure drop which could also be extended to all flows. Therefore, it is concluded that the loss of ac power analysis in the Turkey Point FSAR is conservative for the repaired steam generators. 5

5.1.2.6 Rupture of a Main Steam Pipe (Steam Break)

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. An uncontrolled steam release, typically through a ruptured steam line or a defective valve, causes the secondary system temperature and pressure to fall and the heat transfer rate through the steam generator tubes to rise. Therefore, the heat removal rate from the reactor coolant system increases, and the core moderator temperature drops. As the core is cooled, the negative moderator temperature coefficient causes the core reactivity level to rise.

The FSAR analysis of an uncontrolled steam release was performed to demonstrate that:

a. Assuming a stuck control rod assembly with or without offsite power and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

b. There is no return to criticality for any single active failure in the main steam system. The single active failure is the

opening, with failure to close, of the largest of any single steam bypass, relief, or safety valve.

c. Energy release to the containment from the worst steamline break does not cause failure of the containment structure.

The following systems provide the necessary protection against the steam break accident:

a. Safety injection system actuation on:

   1. Pressurizer low pressure coincident with low pressurizer level

   2. Steam line differential pressure

   3. High steam flow coincident with low coolant temperature or low steam line pressure

   4. High containment pressure

b. Reactor trip on:

   1. Overpower reactor trips

   2. Reactor trip on safety injection signal

c. Feedwater isolation on safety injection signal

d. Steam line isolation on:

   1. High steam line flow coincident with low reactor coolant system average temperature or low steam line pressure

   2. High-high containment pressure coincident with high containment pressure

Major assumptions include use of end-of-life core kinetics parameters, assumption of the most reactive rod cluster control assembly stuck in the fully withdrawn position, and minimum safety injection capability due to a single failure in the system.

The cases considered are the complete severence of a main steam pipe upstream and downstream of the flow restrictor in the steam pipe, with and without the simultaneous loss of offsite power and steam release through a safety valve. All the cases assume initial hot shutdown condition since steam generator mass inventory is greatest at that condition. Should the reactor be just critical or at power at the time of the steam line break the reactor would be tripped by the normal overpower protection system and the additional stored energy would be removed by the cooldown before the no load condition and shutdown margin assumed above are reached. In addition, the greater steam generator mass at hot shutdown conditions increases the magnitude and duration of the cooldown.

The core power and reactor coolant system transients will not be affected by the repaired steam generators. The reasons for this conclusion include the following:

a. The key parameters which strongly influence the transient are performance of the emergency shutdown system and core reactivity coefficients. There are no changes to these parameters as they are used in the analysis due to repair of the steam generators.

b. The flow area of the main steam line is an important factor in determining the amount and rate of heat extracted from the reactor coolant. This flow area remains unchanged for the proposed modifications.

c. No changes are expected due to differences in initial conditions (zero load steam temperature and pressure are identical for the unit with repaired steam generators). The no load steam generator mass is unchanged.

The steam line break analysis reported in the Turkey Point FSAR utilized the Moody correlation with choked flow in the nozzle of the steam generator. Modifications to the internal components of the steam generator which result in only minor variations in the pressure drop through the secondary side of the steam generator will have no effect on the steam line break analysis. As noted previously, the no load masses for the steam generator remain the same; therefore, the FSAR analysis of containment pressure following a steam line break is unaffected.

Therefore the steam line break and containment pressure analyses presented in the FSAR are valid for the repaired steam generators.

5.1.3 Loss Of Coolant Accident Evaluation

In the event of a major reactor coolant system pipe break, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A safety injection system signal is actuated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50. Thereafter the core heat transfer is based on local

conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms. During the refill period, rod-to-rod radiation is the only heat transfer mechanism.

When the reactor coolant system pressure falls below 600 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break. This conservatism is again consistent with Appendix K of 10 CFR 50.

The reactor is designed to withstand thermal effects caused by a loss of coolant accident, including the double ended severance of the largest reactor coolant system pipe. The reactor core and internals together with the emergency core cooling system (ECCS) are designed so that the reactor can be safely shutdown and the essential heat transfer geometry of the core preserved following the accident.

The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the acceptance criteria (Reference 1).

Several large break loss of coolant analyses have been submitted as amendments to the Turkey Point FSAR. The most recent existing analysis to use as a
baseline for the evaluation of the steam generator repairs was performed with 5
19 percent steam generator tube plugging and F_q equal to 2.05.

This analysis utilized the Westinghouse, October 1975 ECCS Evaluation Model (References 2, 3 and 4). This version includes modifications to the models as specified by the NRC in Reference 5 and complies with 10 CFR 50.46 and Appendix K of 10 CFR 50.

The analysis was performed with a reactor vessel upper head fluid temperature
equal to the reactor coolant system hot leg temperature, and the latest method 5
for calculating T_{in}. The effect of using the hot leg temperature in the reactor vessel upper head is described in Reference 6.

Scoping analyses were performed to determine the effects on ECCS performance of the repaired steam generator parameters. These scoping analyses utilized the Westinghouse, October 1975 ECCS Evaluation Model (References 2, 3 and 4), which includes modifications to the model as specified by the NRC in Reference 5 and complies with 10 CFR 50.46 and Appendix K of 10CFR50. These scoping analyses were performed with the reactor vessel upper head fluid temperature equal to the reactor coolant system hot leg temperature, plus the latest method for calculating T_{in} as approved by the NRC. The basis for these scoping analy-
ses was to compare the original 44 series unplugged steam generator with the 5
repaired steam generator. The differences between the original steam generator and the repaired unit are:

a. The repaired unit will have approximately 1.5 percent fewer tubes. The average tube length will remain the same. The primary side volume and total primary to secondary side heat transfer area decreases.

b. The design of the tube ends is modified in the repaired units so that the total flow resistance is less than that of the original units.

c. Improvements in manufacturing tolerance on tube wall thickness
result in an increase of the overall heat transfer coefficient
for the repaired units. The tube wall thickness will be con-
trolled to a maximum mean thickness of 0.050 inches whereas
the original FPL steam generators used a mean wall thickness of 5
0.0525 inches, resulting in an increase of U of approximately
2.5%. This increase in heat transfer coefficient offsets the
decrease in heat transfer area of approximately 2.2%, therefore
the steam generator heat transfer area remains essentially
unaffected.

The impact of these changes on the limiting design basis LOCA is as follows:

The blowdown phase of the limiting break LOCA, with respect to the
steam generators, is basically dependent on the total coolant mass and
energy of the primary system and heat flow to and from that coolant.
Since the heat transfer remains unchanged due to item c above, the
only effect on the blowdown phase would be the slight decrease in pri-
mary side volume of the steam generator tubes. This results in a 5
slightly faster blowdown of the repaired unit and an earlier time to
end of blowdown.

The core reflood phase of the limiting break LOCA, with respect to the steam generators, is dependent only on the temperature of the water in the secondary side and on the total resistance to flow through the primary side. The secondary side pressure (temperature) will not change in the repaired unit. The overall primary side flow resistance of the repaired unit is less and the total flow area is slightly less. The overall effect of these changes is negligible and the reflood transients of the repaired and original unit are nearly identical.

The results of these scoping analyses showed that the original and repaired 5
units gave similar peak clad temperatures for the same peaking factor. The
scoping analyses of the original and repaired units at a zero plugging level
gave peak clad temperatures of 2189 and 2190 respectively at a peaking factor
of 2.20.

The scoping analyses show that the replacement steam generator's LOCA response compares closely with that of the original steam generators and the ECCS performance satisfies the current criteria.

The existing small break analysis for Turkey Point is described in Section 14.3 of the FSAR. This analysis is in conformance with 10 CFR 50.46 and Appendix K to 10 CFR 50. None of the parameters in the repaired steam generator has a significant effect on small break LOCA. Thus the effect on the small break analyses is negligible and the existing small break analyses in the FSAR are applicable to the plant with repaired steam generators.

The containment mass and energy release in the FSAR were performed in con-
formance with criteria existing at the time of the Turkey Point Operating
License submittal. This analysis for a LOCA is sensitive to reactor power
and primary system volume. Since the reactor power remains the same and the
change in primary system volume is small the containment mass and energy re- 5
lease would not be effected and the existing analysis is still applicable.

5.1.4 Steam Generator Tube Rupture

The improved manufacturing tolerance on the tube wall thickness will result in an increase in the tube inner diameter. This increase in diameter is small and the effects on the tube rupture analysis is negligible. Thus, the tube rupture analysis and consequences in the Turkey Point FSAR would be unchanged with the repaired steam generators and remain valid. 5

5.1.5 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471, April 1975 (Proprietary) and WCAP-8472, April 1975 (Non-Proprietary).

3. "Westinghouse ECCS Evaluation Model October 1975 Version," WCAP-8622 November 1975 (Proprietary), and WCAP-8623, November 1975 (Non-Proprietary).

4. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassallo of the Nuclear Regulatory Commission, Letter Number NS-CE-924, January 23, 1976.

5. "Supplement to the Status Report by the Directorate of Licensing in the matter of Westinghouse Electric Corporation ECCS Evaluation Model Conformance of 10 CFR 50 Appendix K," Federal Register, November 1974.

6. Letter from C. Eicheldinger of Westinghouse Electric Corporation to V. Stello of the Nuclear Regulatory Commission, Letter Number NS-CE-1163, August 13, 1976.

5.2 CONSTRUCTION RELATED EVALUATIONS

Rigging and transportation of heavy load requirements have been evaluated. Administrative procedures and controls will be established to minimize the likelihood of any mishap. This nonwithstanding, rigging and construction incidents have been postulated to occur. The evaluations below demonstrate that the in situ configuration, augmented where appropriate with temporary physical protection, can accommodate all events analyzed. In addition, to assess ultimate facility capability to accommodate these events, the functions potentially impacted by each construction event were assumed lost. The safe shutdown and maintenance of spent fuel cooling evaluation provided below demonstrates that for each event, loss of function could be acceptably accommodated.

The conclusion reached by the analysis of construction related incidents is that any loss of safety-related functions has been precluded and that

the assumed loss of potentially affected safety-related functions is acceptable. Therefore, there is no unreviewed safety question associated with the construction activity.

5.2.1 Handling of Heavy Objects

As described in Section 3.1.5, precautions will be taken to preclude the possibility that a rigging or transportation incident will damage any component, system or structure important to the nuclear safety of either unit. These precautions include training of equipment operating personnel, additional protection of buried piping and duct banks where necessary along the steam generator haul routes, controls on haul routes and equipment speed, and controls on lift heights, travel directions, location and swing arcs for both loaded and unloaded cranes. However, for the purpose of evaluation, certain construction related incidents were analyzed and the results of the analyses are summarized in the following subsections.

These analyses demonstrate that the spectrum of postulated events will not preclude the ability to achieve/maintain a safe shutdown condition or to cool the spent fuel pool. These events are not likely to occur. In all cases analyzed, these events have been precluded by design and/or temporary augmented protective measures.

5.2.1.1 Overturning of a Loaded Trailer

Analyses were performed to determine the conditions which would be required to overturn a typical trailer loaded with a steam generator lower assembly. The following assumptions were made for the purpose of these analyses:

a. A multi-axle, multi-tire trailer with a bed height of 4 feet above the ground

b. Trailer tire span width of 12 feet out-to-out of tires

c. A combined trailer weight and steam generator saddle weight of 70 tons with a center of gravity at 4 feet from ground elevation

d. A steam generator lower assembly weight of 205 tons with center of gravity at 12 feet from ground elevation

e. A worst case turn radius of 25 feet

Results of analyses indicate that the loaded trailer would become unstable and overturn if inclined beyond 31 degrees from the horizontal, or if the trailer exceeds a speed of 15 miles per hour in a turn radius of 25 feet.

For the following reasons, it is concluded that overturning of a typical trailer is highly unlikely:

a. Except for the containment access ramps, the haul routes do not have any banking; therefore, the trailer will not be inclined horizontally. In addition, as described in 3

Subsection 3.1.1.2, haul routes will be evaluated and upgraded, if necessary, to preclude the possibility of roadway collapse.

b. The turning radius of any haul road on the plant site will significantly exceed the conservatively assumed turn radius of 25 feet.

c. A speed less than 15 mph will be maintained by administrative controls.

5.2.1.2 Effect of Postulated Rigging Incidents on Safety-Related Plant Structures

Postulated Unloaded Crane Boom Drop

Analyses were performed to determine the ability of the following safety-related structures to withstand the impact of a freefalling unloaded crane boom. (See Figure 5.2-1 for locations):

LOCATION A - Unit 4 auxiliary building walls and roof

LOCATION B - Unit 4 buried intake cooling water (ICW) piping under Unit 4 equipment hatch construction area

LOCATION G - Unit 3 buried ICW piping under Unit 3 equipment hatch access road

LOCATION H - Unit 3 buried duct bank under Unit 3 equipment hatch construction area

LOCATION I - Unit 3 auxiliary building walls and roof

LOCATION J - Diesel generator building walls and roof

The following assumptions were made for the purposes of these analyses:

a. For above ground structures, a crane operator inadvertently orients the plane of boom motion at 90° to either the auxiliary building or the diesel generator building.

b. The crane boom is in an initial vertical position.

c. For buried structures, the crane is situated during the incident such that impact occurs directly over the structures.

d. A 70-foot long unloaded crane boom weighing 30,000 pounds falls through a vertical plane prior to impact (friction is neglected).

e. No additional protection, such as timber matting, structural bridging or added fill, is considered in the analyses for buried structures.

An analysis was conducted for a freefalling crane boom on the auxiliary building roof over the spent fuel pool. It assumed the boom impacts the auxiliary building at the parapet, a simple plastic hinge is formed in the boom, and the velocity of the boom extending beyond the parapet does not decrease after impact. Based upon these conservative assumptions, the structural response of the auxiliary building roof over the spent fuel pool was not considered adequate.

Since the analysis above did not consider any resistance that might be provided by the parapet or the formation of a moment-resisting plastic hinge, a more refined analyses was performed to take this into account. The refined analysis accounts for the fact that when the boom impacts the auxiliary building at the parapet, a moment-resisting plastic hinge will form in the lower chord of the boom at the point of impact. The resistance of the hinge was assumed to be in accordance with AISC Specification Part II, for plastic design. The motion of 1
the portion of the boom beyond the parapet is reduced by this moment resistance. The results of this analysis indicate that the structural response of the roof is acceptable.

All other structures evaluated were also found capable of withstanding the impact of the freefalling crane boom.

In analyzing the walls of structures for boom impact, the boom was assumed to impact at midheight of the wall. Using the energy balance method as described in BC-TOP-9A (Reference 1), a conservative maximum ductility of approximately 15 was calculated for the walls. Local effects, such as penetration, perforation and spalling were evaluated per BC-TOP-9A and found acceptable.

For the following reasons, it is concluded that the effects of an incident involving a freefalling boom require no further consideration:

a. The likelihood that a crane operator orients the plane of boom motion at 90° to building structures at the same time the crane boom system fails is remote.

b. The likelihood that the crane boom will be in an initial vertical position and strike the structures at an angle of 90° at the same time the events in a. above are occurring is remote.

c. The likelihood that the crane will be situated such that impact occurs directly over buried structures at the same time the crane boom support system fails is remote.

d. Even discounting the considerations in a., b., and c., results of these analyses indicate that all evaluated safety-related 1
structures will maintain structural integrity.

Postulated Steam Generator Lower Assembly Drop

Analyses were performed to determine the ability of the following safety-related buried structures to withstand the impact of a steam generator lower assembly dropped during the transfer from the service platform to a transporter (see Figure 5.2-1 for locations):

LOCATION B - Unit 4 buried ICW piping under Unit 4 equipment hatch construction area

LOCATION H - Unit 3 buried duct bank under Unit 3 equipment hatch construction area

The following assumptions were made for the purposes of these analyses:

a. The trailer transporter is not in position under the load at the time of rigging failure.

b. The 205-ton steam generator lower assembly drops 13 feet to the ground directly over the buried structure.

c. No additional protection, such as timber matting, structural bridging or added fill, is considered in the analyses.

Results of the analyses indicate that additional protection would be required for both the electrical duct banks and ICW piping. Therefore, protection will be provided over the buried structures to maintain structural integrity. This notwithstanding, the ability to accommodate failures of functions potentially impacted by steam generator drop is the same as discussed in Section 5.2.1.4 for transportation incidents.

It should be noted that the steam generator lower assembly will at all times be over either the service platform or the transporter. Therefore, in the event of a rigging failure, considerable energy will be dissipated prior to impact with the ground. Credit for this energy absorption was not assumed.

Other structures which could be impacted by a rigging incident have not been evaluated further because they do not perform safety-related functions during steam generator repair on that unit. These structures are as follows:

Containment	-	During the repair, fuel is removed from the affected containment.
Condensate Storage Tank	-	Since fuel cooling is accomplished entirely by the spent fuel pool cooling system, there is no need to maintain a supply of water for the auxiliary feedwater pumps.
Diesel Oil Storage Tank	-	More than 8000 gallons of fuel oil capacity are available within the diesel generator building. This is sufficient to maintain two units in hot shutdown for 69 hours and allow time to obtain additional fuel from local suppliers if the need arises. If the Unit 3 containment access ramp is removed as 1 described in Subsection 3.1.1.2, it may be necessary to drain the tank because the ramp forms a portion of the oil retention dike around the tank. Alternatively a temporary dike or other suitable temporary features would be installed to confine and/or divert the spill to a suitable location. If this tank is drained, the quantity of diesel oil required by the plant Technical Specifications will be stored elsewhere on site.

5.2.1.3 Postulated Transportation Incidents Involving Safety-Related Plant Functions

Although the precautionary measures described previously will preclude loss of function of safety-related equipment due to postulated transportation and

rigging incidents, an evaluation was performed conservatively assuming that
loss of function would occur. Locations for postulated transportation inci-
dents are provided below. Figure 5.2-1 illustrates these locations, as well
as the haul route from the barge slip northeast of Unit 1. The accident 7
locations evaluated are based on haul routes associated with each onsite
receipt and disposition alternative being considered for the lower assem- 3
blies. It must be noted, however, that these evaluations are provided to
illustrate capability only; actual construction practices will preclude
occurrence of these accidents.

LOCATION B - Unit 4 buried intake cooling water (ICW) piping under Unit 4 equipment hatch construction area

LOCATION C - Unit 4 buried ICW piping and duct bank under steam generator haul route

LOCATION D - Unit 4 buried duct bank under steam generator haul route

LOCATION E - Unit 3 buried ICW piping under steam generator haul route

LOCATION F - Unit 3 buried duct bank under steam generator haul route

LOCATION G - Unit 3 buried ICW piping under Unit 3 equipment hatch access road

LOCATION H - Unit 3 buried duct bank under Unit 3 equipment hatch construction area

Other plant equipment adjacent to the transportation haul routes was evaluated and determined to be either not required for safe shutdown or far enough from the haul route to preclude the potential for damage.

5.2.1.4 Evaluation of Postulated Transportation Incidents

The safety analysis for each location provided in this Subsection includes the analysis of the assumed loss of all safety-related functions at each location.

<u>Unit 4 Shutdown/Under Repair - Unit 3 Operating</u>

LOCATION B - Unit 4 Buried ICW Piping Under Unit 4 Equipment Hatch Construction Area

It was conservatively assumed that all Unit 4 ICW piping in this location was damaged, such that total cooling function to the Unit 4 component cooling water (CCW) heat exchangers was lost. Since, under these conditions, Unit 4 is shutdown, the only potential safety-related consequence is loss of cooling to the Unit 4 spent fuel pool.

Should this postulated incident occur, the Unit 3 CCW system would be aligned through presently installed permanent crossover piping to the Unit 4 CCW system thus reestablishing spent fuel pool cooling. The CCW system intertie capability has been evaluated and the CCW system of one unit is capable of removing the heat load of the operating unit and the additional heat load of the other's spent fuel pool. Availability of the crosstie is not affected by a loss of offsite power.

It has been determined that a postulated incident in this area will not impact the operating unit.

LOCATION C - Unit 4 Buried ICW Piping and Duct Bank Under Steam Generator Haul Rote

It was conservatively assumed that all Unit 4 ICW piping and power cable to the Unit 4 ICW pumps were damaged such that total cooling function to the Unit 4 CCW heat exchangers was lost. The only potential safety-related consequence is loss of cooling to the Unit 4 spent fuel pool. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish spent fuel cooling until necessary repairs are accomplished. 5

LOCATION D - Unit 4 Buried Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all the Unit 4 ICW local control cables were damaged. Following this failure it is expected that the ICW pumps will continue to operate. 1 However, in the event that a pump trips and cannot be restarted from the control room, it would be restarted at its associated 4.16-kV switchgear.

LOCATION E - Unit 3 Buried ICW Piping Under Steam Generator Haul Route

It was conservatively assumed that all Unit 3 ICW piping was damaged such that total cooling function to the Unit 3 CCW heat exchangers was lost. This would result in loss of all normal CCW functions to Unit 3.

Unit 3 would be tripped and maintained in a hot shutdown condition with secondary heat removal via the auxiliary feedwater system. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling as required. Under these conditions, the unit can be maintained at hot shutdown conditions using the auxiliary feedwater system and condensate storage tank makeup supplied by the water treatment plant. In the event that offsite power is unavailable, the water treatment plant would be loaded on a vital bus. There is additional secondary makeup water inventory available in the Unit 4 condensate storage tank, the Unit 3 primary water tank and the Unit 4 primary water tank.

This long term hot shutdown capability will allow sufficient time to affect necessary repairs.

LOCATION F - Unit 3 Buried Duct Bank Under Steam Generator Haul Route

This duct bank contains all power and control cables to the Unit 3 intake structure (e.g., ICW pumps and circulating water pumps). The evaluation

assumed total loss of function of this equipment.

The evaluation assumed total loss of ICW flow to the Unit 3 CCW heat exchangers. Unit 3 would be tripped and maintained in a hot shutdown condition as described above (Location E). The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling as required. 1

As noted above (Location E), there is sufficient time to affect necessary repairs.

Unit 3 Shutdown/Under Repair - Unit 4 Operating

LOCATION C - Unit 4 Buried ICW piping and Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all Unit 4 ICW piping and power cable to the Unit 4 ICW pumps were damaged such that total cooling function to the Unit 4 CCW heat exchangers was lost. This would result in loss of all normal CCW functions for Unit 4.

Following reactor trip, the unit would be maintained in the hot shutdown condition with secondary heat removal via the auxiliary feedwater system. The Unit 3 to Unit 4 CCW intertie would be utilized as discussed above to 1
reestablish component cooling to equipment as required. Under these conditions the unit can be maintained at hot shutdown condition indefinitely using the auxiliary feedwater system with condensate storage tank makeup supplied by the water treatment plant. In the event that offsite power were unavailable, the water treatment plant would be loaded on a vital bus. In addition, available inventory in the Unit 3 condensate storage tank, the Unit 3 primary water tank and the Unit 4 primary water tank can be made available for secondary heat removal.

This long term hot shutdown capability will allow sufficient time to affect necessary repairs.

LOCATION D - Unit 4 Buried Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all the Unit 4 ICW local control cables were damaged. Following this failure it is expected that the ICW pumps will continue to operate. However, in the event that a pump trips and cannot be restarted from the control room, it would be restarted at its associated 4.16 kV switchgear.

LOCATION E - Unit 3 Buried ICW Piping Under Steam Generator Haul Route

It was conservatively assumed that all Unit 3 ICW piping in this location was damaged such that total cooling function to the Unit 3 CCW heat exchangers was lost. Since, under these conditions, Unit 3 is shutdown, the only potential safety-related consequence is loss of cooling to the Unit 3 spent fuel pool.

Should this postulated incident occur, the Unit 4 CCW system would be aligned through permanent crossover piping to the Unit 3 CCW system as discussed above (LOCATION B).

It has been determined that a postulated incident in this area will not impact the operating unit.

LOCATION F - Unit 3 Buried Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all power and control cables to the Unit 3 ICW pumps were damaged such that total cooling function to the Unit 3 CCW heat exchangers was lost. This would result in loss of cooling of the Unit 3 spent fuel pool.

Should this postulated incident occur, the Unit 4 CCW system would be intertied with the Unit 3 CCW system as discussed above. (LOCATION B).

This long term ability to cool the spent fuel pool will allow sufficient time to affect necessary repairs.

Loss of function of the remaining cables in this duct could require that Unit 4 be brought to the hot shutdown condition until an alternate power feed is reestablished to the Unit 4 circulating water pumps. Following this, Unit 4 can be returned to full power operation. 5

LOCATION G - Unit 3 Buried ICW Piping Under Unit 3 Equipment Hatch Access Road

The potential consequences of failing the ICW piping in this area are discussed above (LOCATION E).

LOCATION H - Unit 3 Buried Duct Bank Under Unit 3 Equipment Hatch Construction Area

It was conservatively assumed that all power and control cables to the Unit 3 ICW pumps, spent fuel pool cooling (SFPC) pumps and CCW pumps were lost. This would result in loss of cooling of the Unit 3 spent fuel pool.

Should this postulated incident occur, the Unit 4 CCW system would be crosstied with the Unit 3 CCW system as discussed above (LOCATION B). A temporary power supply can be connected to the Unit 3 SFPC pumps to reestablish spent fuel pool cooling. It must be noted, however, that reliance on the availability of the temporary power supply is not necessary. If it were not provided, fuel pool boiling, with establishment of a makeup source to accommodate boiloff, would constitute an acceptable temporary operating mode.

This long term ability to cool the spent fuel pool will allow sufficient time to affect necessary repairs.

The potential consequences of damaging the cables to Unit 4 are discussed above (LOCATION F).

Unit 3 and Unit 4 Operating

It may be necessary to transport the new steam generators over the portion of the haul route encompassing postulated incident locations C, D, E and F prior to the repair outage with both units in normal operation.

LOCATION C - Unit 4 Buried ICW Piping and Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all Unit 4 ICW piping and power cable to the Unit 4 ICW pumps were damaged such that total cooling function to the Unit 4 CCW heat exchangers was lost. This would result in loss of all normal CCW functions for Unit 4.

As discussed above in the scenario with only Unit 4 operating, Unit 4 would be maintained in the hot shutdown condition following reactor trip with secondary heat removal via the auxiliary feedwater system. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling to the Unit 4 spent fuel pool and other equipment as required. The Unit 3 component cooling water system has sufficient capacity to supply its own operational cooling needs and to supply the hot shutdown requirements of Unit 4. 5

LOCATION D - Unit 4 Buried Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all of the Unit 4 ICW local control cables were damaged. As discussed above, it is expected that following this failure the ICW pumps will continue to operate. However, in the event that a pump trips and cannot be restarted from the control room, it would be restarted at its associated 4.16 KV switchgear.

LOCATION E - Unit 3 Buried ICW Piping Under Steam Generator Haul Route

It was conservatively assumed that all Unit 3 ICW piping was damaged such that total cooling function of the Unit 3 CCW heat exchangers was lost. This would result in loss of all normal CCW functions to Unit 3.

As discussed above in the scenario with only Unit 3 operating, Unit 3 would be maintained in the hot shutdown condition following reactor trip with secondary heat removal via the auxiliary feedwater system. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling to the Unit 3 spent fuel pool and other equipment as required. The Unit 4 component cooling water system has sufficient capacity to supply its own operational cooling needs and to supply the hot shutdown requirements of Unit 3.

LOCATION F - Unit 3 Buried Duct Bank Under Steam Generator Haul Route

It was conservatively assumed that all power and control cables to the Unit 3 ICW pumps were damaged such that total cooling function to the Unit 3 CCW heat 5
exchangers was lost.

Unit 3 would be tripped and maintained in a hot shutdown condition. The Unit 3 to Unit 4 CCW intertie would be utilized to reestablish component cooling as required.

Loss of function of the remaining cables in this duct could require that Unit 4 also be brought to hot shutdown condition until an alternate power feed is reestablished to the Unit 4 circulating water pumps. Unit 4 ICW system would remain functional to supply the Unit 3 and Unit 4 hot shutdown and spent fuel pool cooling requirements. Following this, Unit 4 can be returned to full power operation.

5.2.1.5 References

1. "Design of Structures for Missile Impact," BC-TOP-9A, Revision 2, 1
September 1974

5.2.2 Radioactive Releases and Dose Assessment

Radioactive airborne and liquid releases have been evaluated for the repair effort using conservative, bounding parameters and assumptions. In order to assess the significance of these releases, they were compared with the radioactive releases at Turkey Point and are shown on Tables 5.2-3 and 5.2-7. 7

5.2.2.1 Airborne Releases

Airborne effluent releases to the environment resulting from the repair effort have been estimated using the following assumptions and parameters.

a. Airborne releases are expected to occur mostly during the decontamination, and cutting of the channel head and the divider plate. Cutting and removal of concrete necessary for this operation are expected to result in insignificant releases of radioactivity.

b. The repair effort is assumed to start after approximately 7 and 9 7
years of commercial operation for Unit 4 and 3, respectively.

c. The channel head and divider plate are expected to be contaminated primarily by deposited corrosion products. Typical radioisotopic activities expected on the channel head surface after 9 years of commercial operation are given in Table 5.2-1. Divider plate activities are expected to be approximately 2 times the activities given in Table 5.2-1 for Unit 4, the activities on the channel head and divider plate are bounded by those for Unit 3.

d. Fifteen days of radioisotopic decay prior to decontamination and cutting is assumed.

e. For the purpose of analysis, it is assumed that the channel head internal sufaces are decontaminated by some method (as described in Section 3.3.5.3.2) which yields a DF of 12 and, then, the steam generator midsection is cut and removed from the channel head. Airborne radioactivity concentrations are then estimated assuming that all of the activity remaining on the channel head and divider plate surfaces of three steam generators, after this initial decontamination, becomes airborne due to further decontamination and cutting. 7

f. During decontamination and cutting, the channel heads will be surrounded by contamination control envelopes where atmosphere will be exhausted through a HEPA filter. The HEPA filter is assumed to be 99 percent efficient.

Radioactive airborne effluent release to the environment based on the above assumptions are calculated to be 4.59×10^{-2} Ci per unit. Details of airborne effluent release by isotopes are given in Table 5.2-2.

5.2.2.2 Environmental Consequences of Airborne Releases

The critical organ (lung) and whole body doses for a teenager (critical age group) at the worst site boundary location resulting from the calculated airborne effluent releases during the repair effort were evaluated using a ground level release atmospheric dispersion factor of 1.02×10^{-6} Sec/Meter^3 (Turkey Point FSAR Chapters 2 and 11) and the dose models and dose factors given in Regulatory Guide 1.109. The critical organ and whole body doses are estimated to be 5.92×10^{-3} and 9.14×10^{-6} mrem, respectively, per unit during the repair effort.

5.2.2.3 Comparison with Estimated Doses During Normal Operation

Observed gaseous effluent releases during the period July 1978 through June 1979 are compared with estimated releases during the repair effort in Table 5.2-3.

The critical organ dose (thyroid) and the whole body dose for a child (critical age group) at the worst site boundary location due to the release of gaseous effluents for 1978-1979 were calculated to be 0.02 and 0.09 mrem/unit, respectively. The estimated critical organ (lung) dose for the repair effort is 32 percent of the calculated critical organ dose for 1978-1979. The estimated whole body dose for the repair effort is 0.01 percent of calculated whole body dose during 1978-1979. 7

5.2.2.4 Liquid Effluent Releases

Since the steam generator channel head primary side surfaces are planned to be decontaminated by a method that may generate liquid waste, the liquid effluent release for the channel cut approach will increase by approximately 0.015 Ci per unit. With this minor addition, the discussion below and the discussion in Section 5.2.2.5, originally for the pipe cut approach, are applicable to the channel cut approach.

Liquid effluent releases resulting from the repair effort were estimated using the following parameters and assumptions:

a. The reactor coolant system is drained 15 days after reactor shutdown and the reactor coolant is subsequently discharged after

processing through a mixed bed demineralizer and through the boric acid recovery evaporator as required. Laundry waste water is discharged without processing.

b. The decontamination factors for processing equipment are listed below and are in accordance with NRC NUREG-0017:

Processing Equipment	Decontamination Factors		
	Iodines	Cs&Rb	Others
Mixed bed demineralizer	10	2	10
Boric acid recovery evaporator	100	1000	1000

c. Reactor coolant concentrations are given in Table 5.2-4 and are based on radiochemical analyses taken during 1976 at Turkey Point.

d. The mass of the reactor coolant discharged after processing is 4.08×10^5 lbs.

e. Laundry releases were estimated using the expected specific activities in the laundry waste water given in Table 5.2-5 and assuming approximately 22,000 gal/day of laundry waste water will be discharged for approximately 300 days during the repair effort for one unit. (It is expected, however, that on the average only 10,000 gal/day of laundry waste water will be discharged during this period.)

The total radioactive liquid effluent release based on the above assumptions is estimated to be approximately 0.55 Ci/unit, excluding tritium and dissolved gases, and approximately 185 Ci/unit of tritium. Details of this release by isotope are given in Table 5.2-6.

5.2.2.5 Comparison with Observed Radioactive Liquid Releases During Normal Operation

Estimated radioactive liquid releases during the repair effort are compared with the observed liquid waste releases during the period 1978 through 1979 in Table 5.2-7. The estimated total radioactive liquid release per unit (excluding tritium and dissolved gases) during the repair effort is seen to be about 1.8 times the observed total liquid waste release per unit (excluding tritium and dissolved gases) during 1978-1979. The estimated tritium release per unit during the replacement effort is about 40 percent of the observed tritium release per unit during 1978-1979. 7

TABLE 5.1-1

COMPARISON OF OPERATING PARAMETERS FOR ORIGINAL AND REPAIRED STEAM GENERATORS

Nominal power/SG	Unchanged
Nominal primary flow/SG	Unchanged
Nominal hot leg temperature, ^{o}F	Unchanged
Nominal cold leg temperature, ^{o}F	Unchanged
Feedwater temperature, ^{o}F	Unchanged
Reactor Coolant System pressure, psia	Unchanged
Nominal steam pressure, psia	Unchanged
Nominal fluid mass/SG, lb^{m}	Increased 5%
No load fluid mass/SG, lb^{m}	Unchanged
No load temperature, ^{o}F	Unchanged
Steam flow	Unchanged

TABLE 5.1-2

COMPARISON OF DESIGN PARAMETERS FOR ORIGINAL AND REPAIRED STEAM GENERATORS

Primary Side Volume	Decreased by 1%	5
Number of Tubes	Decreased by 46	
Tube O.D.	Unchanged	
Wall Thickness (Nominal)	Unchanged	5
Primary Pressure Drop	Decreased by 0.7 psi	5
Fouling Factor	Unchanged	
Heat Transfer Area	Decreased by 2.2%	5
Flow Area	Decreased by 1.5%	
Equivalent Length	Unchanged	5

TABLE 5.2-1

ESTIMATED CORROSION PRODUCT ACTIVITIES ON STEAM GENERATOR PRIMARY SIDE PLENUM (1) (2) (3) (4) (5) (6) 7

Isotope	Activity (Ci/cm^2)	Isotope	Activity (Ci/cm^2)
Co-58	26.03	I-131	4.5×10^{-1}
Co-60	18.1	I-132	4.5×10^{-1}
Mn-54	2.03	Te-132	4.5×10^{-1}
Fe-59	4.65×10^{-1}	Cs-137	6.74×10^{-2}
Cr-51	1.35	Ba-140	4.5×10^{-1}
Zr-95	2.25	La-140	1.35
Nb-95	3.15	Ce-141	7.5×10^{-1}
Mo-99	4.5×10^{-1}	Ce-144	3.72
Tc-99m	2.25×10^{-1}	Np-239	4.8
Ru-103	1.5		

Notes

(1) The activities are based on actual Turkey Point data.

(2) Activities listed are extrapolated to 9 years of commercial operation.

(3) For Unit 4 (approximately 7 years of commercial operation) activities are bounded by those for Unit 3 (approximately 9 years of commercial operation).

(4) The activities shown are for 90 hours after shutdown.

(5) Multiplication Factors for Isotopic Concentrations for Components in the steam generator: 7

Component	Relative Concentration Factor	Area (Cm^2)
Tubes	0.12	4.1×10^7
Divider Plate	2.0	7.2×10^4
Tube Sheet	2.0	3.8×10^4
Rolled Tube Ends	45/tube end	6520 tube ends
Channel Head Bowl	1	1.5×10^5

(6) The amount of each isotope, in curies, can be obtained by decaying the isotope for 80 days (the estimate for the earliest that the generators can be removed from the containment); by multiplying the surface area for each component by the primary side concentration and the relative concentration factor; and by summing for all components.

TABLE 5.2-2

ESTIMATED AIRBORNE RELEASES TO ENVIRONMENT DURING THE STEAM GENERATOR REPAIR (1) (2)

Isotope	Release (Ci
Co-58	1.65 E-2
Fe-59	2.72 E-4
Co-60	1.33 E-2
Mn-54	1.45 E-3
Cr-51	6.75 E-4
Zr-95	1.42 E-3
Nb-95	2.12 E-3
Ru-103	8.52 E-4
I-131	9.1 E-3
I-132	1.39 E-3
La-140	1.71 E-4
Ce-141	4.02 E-4
Ce-144	2.65 E-3
Pr-144	2.65 E-3
TOTAL(3)	4.59 E-2

7

NOTES: (1) Repair effort releases are given for Unit 3. For Unit 4, the releases are bounded by those for Unit 3 since Unit 4 will have less operating time.

(2) Releases less than 10^{-4} are of little significance and, therefore, are not listed.

(3) Includes the unlisted isotopes.

TABLE 5.2-3

COMPARISON OF GASEOUS EFFLUENT RELEASES

Isotope	Average July 1978-June 1979 Release/Unit (Ci)	Estimated Release/Unit During the SG Repair Effort (Ci) (1)
Noble gases	11300	Negligible
Iodines	0.055	0.0032
Particulates	0.032	0.0428
Tritium	0.5	Negligible

7

NOTE: (1) Repair effort releases are given for Unit 3. For Unit 4, the releases are bounded by those for Unit 3 since Unit 4 will have less operating time.

TABLE 5.2-4

TYPICAL REACTOR COOLANT SYSTEM
RADIOCHEMICAL ANALYSES
SUMMARY SHEET

Non-Volatile Fission and Activation Products

Nuclide	Unit 4 Specific Activity μ Ci/ml	Unit 3 Specific Activity μ Ci/ml
Ag-110m	2.2×10^{-5}	2.0×10^{-5}
Ba-139	2.6×10^{-4}	1.9×10^{-2}
Co-57	1.9×10^{-6}	-
Co-58	8.8×10^{-4}	3.6×10^{-3}
Co-60	5.8×10^{-5}	8.2×10^{-5}
Cr-51	2.0×10^{-4}	1.4×10^{-4}
Cs-134	1.5×10^{-3}	-
Cs-137	-	1.3×10^{-3}
Cs-138	4.0×10^{-2}	1.7×10^{-1}
F-18	2.1×10^{-1}	1.2×10^{-1}
Fe-59	9.1×10^{-6}	2.5×10^{-3}
I-131	1.5×10^{-2}	9.4×10^{-3}
I-132	8.2×10^{-3}	1.4×10^{-1}
I-133	1.8×10^{-2}	9.6×10^{-2}
I-134	8.3×10^{-3}	2.2×10^{-1}
I-135	1.0×10^{-2}	1.4×10^{-1}
Mn-54	1.7×10^{-5}	-
Na-24	9.8×10^{-3}	4.7×10^{-3}

Nb-95	2.6×10^{-5}	2.4×10^{-5}
Nb-97	4.2×10^{-5}	3.4×10^{-5}
Sb-124	1.2×10^{-5}	-
Sr-89	2.6×10^{-5}	1.7×10^{-4}
Sr-91	-	1.3×10^{-4}
Te-132	-	6.1×10^{-4}
Zr-95	1.7×10^{-5}	-
Zr-97 - Nb-97m	3.4×10^{-6}	2.5×10^{-5}

Gaseous Fission and Activation Products

Nuclide	μ Ci/ml	μ Ci/ml
Ar-41	1.9×10^{-4}	-
Kr-85	-	1.1×10^{-2}
Kr-85m	4.2×10^{-3}	-
Kr-87	5.3×10^{-3}	2.1×10^{-2}
Kr-88	6.2×10^{-3}	1.8×10^{-2}
Xe-133	5.6×10^{-2}	6.4×10^{-2}
Xe-133m	1.5×10^{-3}	-
Xe-135	2.2×10^{-2}	6.8×10^{-2}
Xe-135m	4.2×10^{-3}	3.6×10^{-2}
Xe-138	-	5×10^{-2}
Total Non-Gas	0.322	0.928
Total Gaseous	0.100	0.268
Total Isotopic	0.422	1.2

TABLE 5.2-5
ESTIMATED SPECIFIC ACTIVITIES OF LAUNDRY WASTE WATER

Isotope	Specific Activity (1) μ Ci/cc
Co-58	6.7×10^{-6}
Co-60	5×10^{-6}
Cs-137	5.4×10^{-6}
Cs-134	6.5×10^{-7}
Mn-54	7.3×10^{-7}
I-131	1.06×10^{-7}

Note

(1) Time-averaged specific activity during a period of 300 days

TABLE 5.2-6

ESTIMATED RADIOACTIVE LIQUID EFFLUENT RELEASES DURING THE STEAM GENERATOR REPAIR

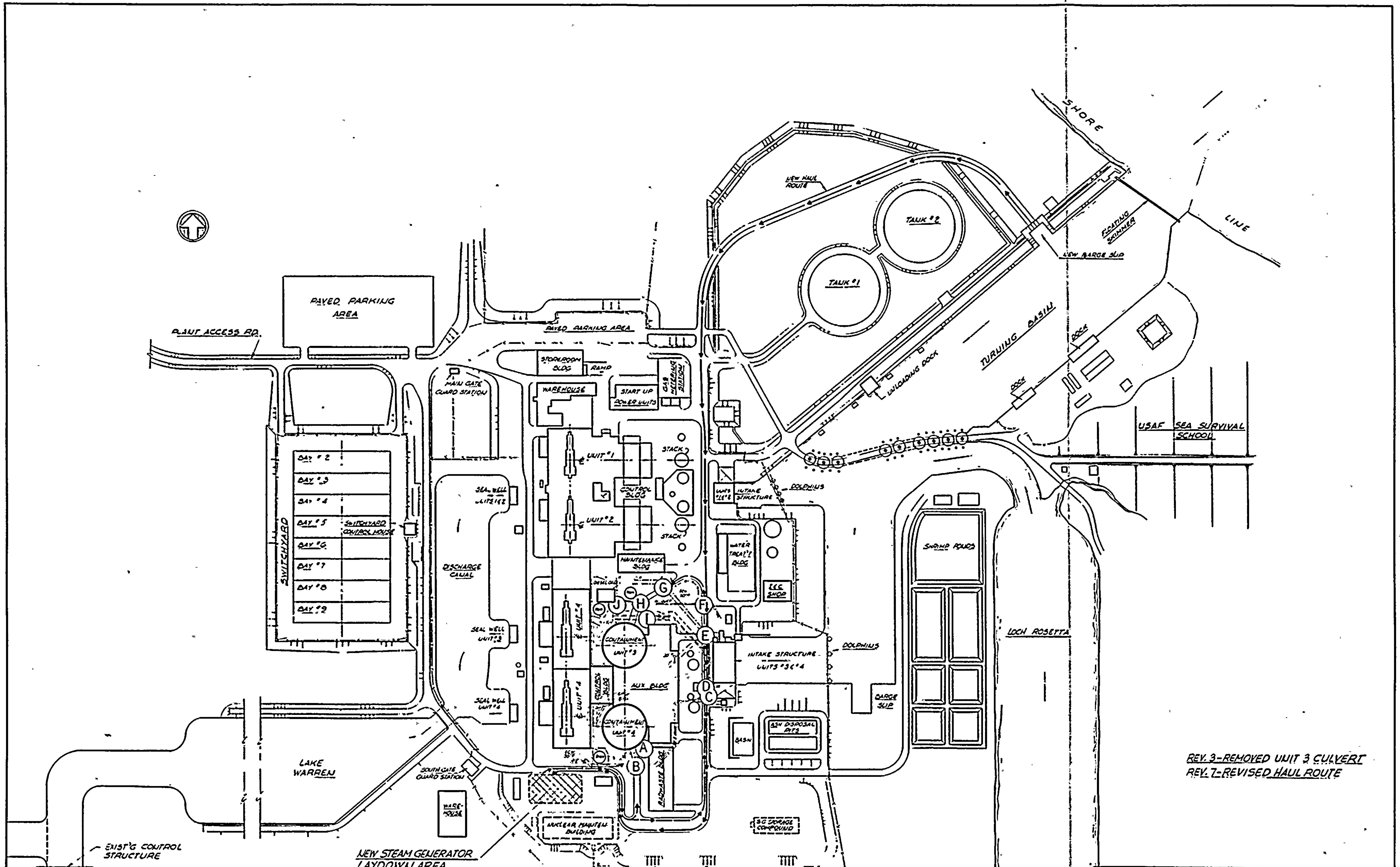
Isotope	Release/Unit (Ci)
Mn-54	1.8×10^{-2}
Co-58	1.7×10^{-1}
Fe-59	1.7×10^{-5}
Co-60	1.25×10^{-1}
Cr-51	3.7×10^{-4}
Cs-134	3×10^{-2}
Cs-137	1.35×10^{-1}
Sr-89	4.8×10^{-5}
I-131	7.6×10^{-2}
Total	5.54×10^{-1}
Tritium	185

TABLE 5.2-7

COMPARISON OF RADIOACTIVE LIQUID EFFLUENT RELEASES

Isotope	Average 1978-1979 Release/Unit (Ci)	Estimated Release During the S.G. Repair Effort/Unit (Ci)
Total (excluding tritium and dissolved gases)	0.305	0.55
Tritium	458	185

7



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

CONSTRUCTION INCIDENT LOCATIONS
FIGURE 5.2-1

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March 1980

6.0 ENVIRONMENTAL ASPECTS OF THE REPAIR EFFORT

6.1 GENERAL

This section evaluates environmental effects relevant to the repair effort and demonstrates that the repair activity will not present any additional environmental impacts as compared to normal plant operation. Construction activities will be carried out in conformance with local, state and federal regulations. When the facility is returned to service subsequent to repair, water use, occupational exposures and radiological releases are expected to be less than those associated with current facility operations.

6.2 RESOURCES COMMITTED

6.2.1 Non-Recyclable Building Materials

The steam generator repair effort at Turkey Point Plant requires the commitment of various irretrievable building materials. Table 6.2-1 provides numerical estimates of material resources committed for the repair of a single Turkey Point unit and for a typical new 1000 MWe PWR plant. The building materials to be used for the repair effort are small compared to material resource commitments for a typical nuclear power plant construction.

6.2.2 Land Resources

The repair effort will have minimal impact on the existing site plan layout 1
in terms of required new foundations. There will be no additional virgin land committed due to the repair and onsite storage.

6.2.3 Water Resources

During the repair effort, construction water will be supplied from existing Turkey Point Plant water sources. No requirements for commitments of new water sources have been identified for the repair effort. Since water consumption during an extended shutdown is expected to be less than during plant operation, water consumption during the repair effort will likely result in a reduction in plant water usage.

6.3 WASTE WATER

Sanitary and laundering operation discharges during the repair effort are the only potential waste water sources of significance. Their additional impact is considered negligible because the number of additional people required for repair (about 300) is typical of the number of additional people required for refueling and major maintenance activities associated with normal operation. Waste water generated by task force personnel will be handled as discussed below.

6.3.1 Sanitary Facilities

Since the repair activities will take place in locations (e.g., the containment and laydown area) near which permanent sanitary facilities are not readily accessible, portable units will be used. There will be no substantial modification to existing sanitary facilities as a result of the repair activity.

6.3.2 Laundering Operations

Laundry waste water generated during the repair activities will originate from facilities built for the proposed effort. If required, laundry waste water will be directed to the liquid radwaste system for processing (see Subsection 3.3.6.3). Additional information on the expected quality and quantity of laundry waste water from the steam generator repair program is provided in Subsection 5.2.2.4 of this report.

6.4 CONSTRUCTION

Construction activities at the time of the repair effort will satisfy applicable laws that are in force at that time. These activities will have a negligible effect on noise levels, dust or smoke.

6.4.1 Noise

To examine order of magnitude sound pressure levels (SPL), typical construction noise sources were assumed and their SPL at the site boundary (0.8 miles) was calculated. The calculation utilized the methods described in References 1 and 2. Attenuation results are summarized in Table 6.2-2. Based on the results of Table 6.2-2 and based on the location of the site in a low population area and the limited amount of construction equipment required, noise resulting from the repair program for the steam generators is expected to have negligible, additional impact on the local area.

To protect personnel located on the site, Occupational Safety and Health Administration Standards (OSHA) will be followed. 1

6.4.2 Dust

Dust created by movement of vehicular traffic in an unpaved area, if any, will be abated by periodically spraying with water. The frequency of spraying and the quantity of water sprayed will be determined by visual inspection of the areas and will vary with the weather conditions.

6.4.3 Open Burning

Open burning is not anticipated during the steam generator repair effort. However, should the necessity arise, applicable county and state regulations for open burning will be followed.

6.5 RADIOLOGICAL MONITORING

The estimated releases of radioactive airborne and liquid effluents during the repair effort are found to be smaller than observed effluent releases for the operating plant during the period July 1978 through June 1979. The comparison is shown in Section 5.2.2. The radioactive effluent release points during steam generator repair activities will be the same as during normal plant operations; therefore, the plant radioactive process monitor locations will not be affected. 7

Since releases of radioactive effluents during the repair program will be small relative to the operating plant and their potential exposure pathway will be the same as for the existing plant operation, these effluents will be monitored in accordance with FPL's existing environmental monitoring program.

6.6 RETURN TO OPERATION

6.6.1 Water Use

Water consumption during post repair plant operation is expected to be considerably less than water consumption during current plant operation. Currently, frequent shutdowns of the nuclear units to perform steam generator tube plugging and/or eddy current inspection result in a significant water consumption. Steam generator filling and draining operations are required to locate the leaky tubes prior to plugging, for hydrostatic testing of the steam generators after plugging, and to maintain the other generators in wet lay-up. These operations require significant quantities of water. For example, filling and draining to locate leaks and to perform hydrostatic testing require 60,000 gallons of water per generator requiring tube plugging, plus 15,000 gallons for wet lay-up of the other two steam generators, for a total of approximately 75,000 gallons of water. An outage to perform the currently required periodic steam generator inspections consumes approximately 167,000 gallons of water. If tube plugging is required at the time of the inspection, an additional 40,000 gallons of water will be expended.

Following repair of the steam generators, it is expected that the steam generator tubes will remain intact; therefore, no unit shutdowns are anticipated for steam generator tube plugging and requirements for periodic inspection should be reduced significantly. 1

6.6.2 Operational Exposures

Section 3.3.7 discusses the reduction in man-rem exposure associated with repair. Due to the expected elimination of the necessity to plug steam generator tubes in the repaired steam generators, approximately 235 man-rem 7
will be saved per year after implementing the repair. Thus, after 9 years of operation post-repair, the savings in man-rem will exceed the man-rem expended during the repair.

6.6.3 Radiological Releases

Secondary plant activity results from primary to secondary leakage. The repaired steam generators will result in enhanced tube integrity thus reducing secondary plant releases.

6.7 REFERENCES

1. "Standards Publication for Gas Turbine Sound and its Reduction," NEMA SM 33-1964.

2. L. L. Beranek, "Noise and Vibration Control," McGraw Hill Book Company, 1971, Chapter 7, Pp 164-166.

3. R. H. Bryan and I. T. Dudly, "Estimated Quantities of Materials Contained in a 1000 MWe PWR Plant," Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 1974, (ORNL-TM-4515). 1

4. "Sound and Vibration Control," November 1970

TABLE 6.2-1
NUMERICAL ESTIMATES OF MATERIAL
RESOURCE COMMITMENTS

Material	Estimated Quantity (Metric Tons Except Where Noted) Steam Generator Repair (Per Plant)	Typical 1000 MWe PWR Model Plant (Reference 3)
Aluminum	Negligible	18
Antimony	Negligible	Negligible
Asbestos	Negligible	138
Chromium	54.5	415
Copper	1.5	726
Iron	490	34,662
Lead	Negligible	47
Manganese	1	467
Molybdenum	Negligible	164
Nickel	244	484
Silver	Negligible	1
Tin	Negligible	2
Titanium	Negligible	Negligible
Zinc	Negligible	2
Magnesia	Negligible	783
Cement	225	30,133
Aggregate (Coarse)	625	90,361
Aggregate (Fine)	480	45,855
Wood	3.0×10^4 (board feet)	4.8×10^6 (board feet)

TABLE 6.2-2

SITE BOUNDARY NOISE LEVELS FOR TYPICAL SOURCES

SOURCE	SOURCE SPL (DECIBELS, dBA)	ESTIMATED SITE BOUNDARY SPL* (DECIBELS, dBA)
Dynahoe Hydraulic Backhoe	79	42
Concrete Truck, 14-Yard Capacity	108	45
Air Compressor, 100 psig 150 CFM Capacity	100	40
Pile Driver, 20,000 ft. lb.	101	58
Front End Loader	77	43

* For the purpose of comparison, the following typical levels are given (Reference 4): 1

Passenger Car, 65 mph, measured at 25 feet - 77 dBA

Living Room Music - 76 dBA

Air Conditioning Condensing Unit, measured at 20 feet - 60 dBA

7.0 EVALUATION OF ALTERNATIVES

7.1 INTRODUCTION

Repair, the replacement of the steam generator lower assemblies with new shop-fabricated lower assemblies, has been selected as the optimum solution at hand to correct denting. The discussion that follows demonstrates this. It also vividly indicates that the cost associated with the outage is the overriding consideration that governs any cost benefit evaluation.

The discussion that follows is based on the current state-of-the-art. It assumes that either one or both nuclear units must be shut down or that corrective action is required to ensure an acceptable level of system reliability. It must be noted that the technology as it relates to steam generator corrosion, electrical system requirements and economics are dynamic factors that directly impact the analyses provided below. At the shutdown of Unit 3 and/or Unit 4, evaluations will be updated as required to ensure that FPL embarks on the optimum approach to accommodate outage of the unit(s).

Loss of capacity from one or both Turkey Point units would require the addition of replacement capacity from new generating facilities and/or the purchase of firm power. The cost of new facilities can be compared with the cost of repair. However, the availability of firm power for purchase must be periodically re-evaluated to reflect current conditions. 1

Derating of one or both Turkey Point units is an alternative to the repair that cannot be addressed quantitatively at this time. Parametric studies can be performed assuming various derating conditions to determine the economics of repair versus derating. However, at this time corrosion rates and the likelihood of achieving a corrosion plateau cannot be quantified with precision. Accordingly, economic evaluations of derating do not presently provide a sufficiently reliable prediction of real world events. Should the evolving technology yield suitable corrosion models, further evaluation of derating would be warranted.

Potentially, there are several alternatives to the repair that could accommodate denting, viz., (1) arresting the corrosion phenomenon, (2) in-place restoration of dented tube areas (sleeving), and (3) in-place steam generator refurbishment (retubing). As discussed infra, the ability to sleeve is moot unless tube support plate (TSP) corrosion can be arrested. The ability to arrest TSP corrosion to ensure long term (30 to 40 year) operation without repair is not at hand.

In addition there are several potential means to accomplish the repair activity, viz:, steam generator removal from and entry to containment via (1) the equipment hatch, or (2) the containment wall, or (3) the containment dome.

The viability of each alternative to repair must be determined primarily by its present state of development. Alternatives that require research and development (R&D) to demonstrate feasibility are incompatible with the earliest potential shutdown date for initiation of repair activities. 7

Table 7.1-1 is a summary comparison of the repair options. 2

7.2 ARRESTING TSP CORROSION

Steam generator tube deformation currently being observed in the Turkey Point units results from a continuing corrosion of the TSP. Corrosion products that have a volume appreciably greater than that of the carbon steel support plates are confined in the crevice between tube and TSP. Once the crevice is filled, forces due to the material volumetric change are exerted on the tube and TSP. These corrosion-product-related forces are of sufficient magnitude to cause distortion of the tube. This corrosion-related phenomenon is referred to as denting.

The denting phenomenon has been reproduced in the laboratory. The corrosion process is quite complex and not fully quantified. Synergisms between species present in the crevice are indicated, but to date all possible corrosion-related synergisms have not been identified. Tests to date also indicate that there is some possibility for the successful development of corrosion inhibitors which could arrest or greatly reduce the rate of TSP corrosion.

Laboratory test programs will in time more precisely define the corrosion processes and, hopefully, will develop inhibitors to arrest TSP corrosion. Even if inhibitors are eventually developed, several technical issues must be resolved before they could be utilized with confidence, e.g.,

a. What is the effect of adding inhibitors to steam generators that have experienced TSP corrosion?

b. What are the long term effects of inhibitors in the steam generators? 1

c. Will the inhibitor be compatible with the plant steam cycle?

Based on the current state-of-the-art, it is not possible to predict the likelihood of achieving a saturation level of denting, i.e., an upper limit of TSP corrosion. Accordingly, there is no evidence to date to suggest that denting would or would not plateau at some level which would permit long term plant operation at reduced power.

There are two pivotal factors necessary to cause denting, viz., a TSP corrosion product whose volume is appreciably greater than the parent material, and the existence of crevices wherein chemicals can hide and corrosion products can be confined.

These causal factors can be eliminated by current state-of-the-art designs, which is the approach followed in repair by utilizing new steam generator lower assemblies. The new tube support plate material has a corrosion product of volume essentially equal to that of the parent material, the fully rolled tube eliminates the tube-to-tube sheet crevice, and the quatrefoil TSP minimizes the extent of areas of close tube to TSP clearance and allows for higher sweeping velocities between the tube and TSP which minimizes steam formation and chemical concentrations in this region.

In summary, for plants experiencing appreciable denting, incorporation of state-of-the-art designs appears to currently offer the only viable long term solution to denting.

7.3 IN-PLACE TUBE RESTORATION

The feasibility of locally repairing dented tubes to restore the tubes structural integrity via sleeving has been considered. Sleeving is the insertion of a thin-walled tube insert that is positioned in the vertical section of a tube and hydraulically expanded in place. This method has been utilized in a current test program to restore tube strength for tubes subject to external thinning. The expanded joint may experience minor leakage from 1 to 10 cc/min.

If the cause of external tube damage is eliminated, then sleeving may offer a means of restoring damaged tubes provided that there is no tube deformation or tube diameter reduction (a close tolerance between tube ID and sleeve OD is required for sleeving).

Methods such as sleeving are not applicable to tube restoration associated with the denting phenomenon and are not a viable alternate to repair because:

a. As stated above, there currently is no means available to arrest denting. Sleeving does not address the fundamental cause of denting, viz., TSP corrosion.

b. Sleeving could enhance tube resistance to denting. However, TSP corrosion would continue, thus the initiation of tube denting would, at best, be prolonged by sleeving.

c. Sleeving cannot be utilized for tubes that are deformed, i.e., dented.

d. Since a sleeve cannot be inserted beyond an expanded sleeve, 1
sleeves would have to be installed at the uppermost support plate first and sequentially downward until all support plate intersections are sleeved. There are 3260 U-tubes and six support plates per steam generator. Thus, approximately 39,000 sleeves per steam generator would be required to sleeve all TSP/tube intersections. This would result in an appreciable man-rem burden without compensating benefit since sleeving offers at best an interim fix -- the evidence currently available indicates that repair would eventually be required.

e. Sleeving would require an R&D effort to develop and qualify the sleeving process.

f. Reduced primary side flow resulting from the increased tube resistance and the increased resistance to heat transfer at the sleeve locations would require evaluation to ascertain the effect on unit performance.

In summary, it is concluded that in-place tube restoration via sleeving is currently not a viable alternative to the repair.

7.4 IN-PLACE STEAM GENERATOR REFURBISHMENT

In principle, the methodology exists to refurbish the steam generators in-place. Although much of the technology exists, a comprehensive program of development and testing would be required to provide a basis for cost, time and personnel exposure comparisons.

To refurbish a steam generator, the upper or dome portion of the steam generator would be removed; the lower assembly internals and tubes would be removed and replaced with state-of-the-art internals and tubes; then the dome would be welded back in place. The in-place refurbished steam generator would be equivalent to the new shop-fabricated lower assemblies utilized in the repair effort.

Since both in-place refurbishment and repair result in equivalent steam generators, the viability of refurbishment must be based on economics, the availability of required tooling, and man-rem considerations. Refurbishment requires an R&D effort to develop (1) tools for retubing activities with a high enough production rate to reduce the unit outage to an economically acceptable level and (2) means to reduce personnel exposure to tolerable levels. This R&D effort is not compatible with the earliest potential outage date. 1

A comparative evaluation has been conducted to determine whether or not retubing could reduce the unit outage time to a point where it would be desirable to pursue the requisite R&D activities. This evaluation was based on the current state-of-the-art and assumed production rates for retubing activities believed to be achievable via a relatively short term R&D effort. The outage for refurbishment was estimated to be longer than the time required for replacement.

In-place refurbishment would require work within the steam generators to remove and replace tubing, etc. Shielding techniques and/or decontamination would be required to reduce radiation exposure to personnel. An R&D effort would be required to develop the means to reduce radiation fields to levels compatible with project manloading requirements. Even if exposure mitigating techniques are successfully developed, man-rem exposures associated with refurbishment are expected to exceed those associated with repair.

In summary, retubing does not currently offer an economic alternative to repair. An evaluation of advantages/disadvantages and a cost breakdown estimate of in-place steam generator refurbishment are contained in comparison Tables 7.4-1 and 7.4-2, respectively. 7

Subsequent to the original preparation of this section, Westinghouse Electric Corporation has completed more research and development on in-place steam generator refurbishment (retubing). Westinghouse has submitted a report describing their efforts to the NRC.

FPL has reviewed the in-place steam generator refurbishment method and has noted that considerable uncertainty still exists relative to expected outage length and expected total man-rem incurred for this method. FPL has concluded that sufficient schedular and man-rem .

savings cannot be assured so as to offset the greater capital cost associated with the in-place refurbishment method when compared with 7 the planned repair method.

7.5 ALTERNATIVE REPAIR METHODS

Removal of the steam generators from containment and entry of the shop-fabricated units into containment can be accomplished in one 1 of three ways, viz.,

Scheme I

In this scheme, the steam generator lower assemblies would be removed and replaced through the existing equipment hatch. The upper assemblies (steam dome) remain inside the containment. Following installation of the new lower assemblies, the original upper assemblies will be welded to the new lower assemblies to complete the repair. An evaluation of advantages/ disadvantages and a cost breakdown estimate for Scheme I are contained in 2 comparison Tables 7.4-1 and 7.4-2, respectively.

Scheme II

In this scheme, entire steam generators would be removed and replaced through a construction opening in the containment wall located above the equipment hatch, between elevations +58' and +98'. The opening would be about 20 feet wide and 40 feet high.

Scheme III

In this scheme, entire steam generators would be removed and replaced through a construction opening in the containment dome. The opening would be located above the steam generator closest to the equipment hatch. The opening would be about 20 feet in diameter.

Scheme III requires the use of an elevated rolling gantry on top of the containment. The gantry would move on two track beams (50 tons each). Depending 1 on the orientation of the rolling gantry, a postulated, though unlikely, collapse of the gantry and/or its rigging could impact safety-related plant functions. Thus, this scheme should appropriately be eliminated based on nuclear safety considerations. Schemes I and II do not result in any potentially unacceptable construction-related nuclear safety considerations.

Capital costs for repairing units 3 and 4, as well as a single unit's outage time (from shutdown to commencement of power operation) are as follows: 2

Alternative	Cost ($\$10^6$)	Outage Duration (days)
Scheme I	119	270 *
Scheme II	145	340
Scheme III	135	300

Replacement power costs during the repair (based on fuel differential cost between nuclear and oil) are expected to be about $522,000/day for Unit 4 7 assuming an October 1980 outage. The replacement power costs is expected to be $546,000/day for Unit 3 assuming an October 1981 outage.

*The outage duration period includes the time period to perform the normal refueling operations and maintenance. This also includes the time necessary to perform the appropriate testing and activities required prior to commencement of power operations. The actual duration of steam generator repair activities is expected to be 217 days.

Based on the above, it is clear that Scheme I is the obvious repair alternative. Since there are no nuclear safety issues associated with Scheme I and other non-economic considerations, i.e., man-rem exposures are comparable for Schemes II and III, the unit down-time cost consideration dramatically favors this alternative.

7.6 MAN-REM CONSIDERATIONS

The preceding discussion demonstrates that repair appears to be the only long term method currently available to correct appreciable denting in steam generators. Of the three schemes potentially available to effect the repair, the man-rem exposure will be essentially the same since they differ only in the means of steam generator removal from containment. In-place refurbishment (retubing), although currently not a viable alternative, would likely involve a higher man-rem burden than the repair activity, based on today's state-of-the-art. 7

Since the need for extensive steam generator inspection and tube plugging operations will be obviated by the repair, yearly exposures associated with these steam generator operations will be significantly reduced. After approximately 9 years of operation post-repair, the savings in man-rem gained from the repair will exceed the man-rem expended during the repair. 7
The net result is that there will be a savings in man-rem over the life of the plant (see Section 3.3.7).

7

7.7 REPLACEMENT CAPACITY

If FPL is required to permanently shut down one or both Turkey Point nuclear units, it would have to replace this capacity to ensure adequate electrical system reliability. New coal units to replace the Turkey Point nuclear units would take seven years to complete following a 7
decision to construct them; new nuclear units not less than ten years. New combustion turbines could be constructed in three years following such a decision. The nuclear units at Turkey Point are used for base load operation. Combustion turbines are used to supply peaking power, and thus are not suitable as replacement capacity for these units due to their high fuel and operating costs. In addition, the use of combustion turbines as replacement capacity would increase the use of petroleum products, which is contrary to what we perceive to be the National Energy Policy.

The repairs to the steam generators at Turkey Point are justified when the associated costs of repair are compared to the cost to build replacement capacity. The steam generator repairs are estimated to cost 89.3 \$/kW which must be compared to the costs for replacement capacity provided below. This comparison establishes repair as the current cost-effective choice for satisfying FPL's electrical system requirements.

TYPE OF UNIT	COST (\$/kW)	LEAD TIME (years)
Gas Turbine	370	3
Coal	1357	7
Nuclear	1979	10

7

7.8 DERATION

Section 7.2 discusses TSP corrosion. It indicates that it is not currently possible to predict whether or not a corrosion saturation level or plateau will occur. Should such a plateau become predictable, it would be possible to define a power condition associated with the corrosion plateau. Economic studies could then be conducted to assess the propriety of repair.

Section 7.2 also addresses inhibitors. Should these be successfully developed, rates of TSP corrosion may be reduced dramatically. This could allow for a ramped or linear rate of possible deration over many years. Economic studies could be conducted to determine whether ramped deration or repair is the optimum choice. Alternately, repair might be delayed until a more attractive date. 1

Unfortunately, inhibitors and the ability to predict a TSP corrosion plateau with precision are not at hand. Thus, it is not possible to conduct meaningful deration studies at this time.

7.9 CONCLUSIONS

Repair of the steam generators via the containment equipment hatch pathway is the preferred choice. The dramatic economic advantage due to reduced outage time offsets any potential advantages associated with other viable containment removal schemes.

The ability to arrest TSP corrosion at Turkey Point is not within today's state-of-the-art. Sleeving does not offer any potential benefit without the ability to arrest TSP corrosion. In-place refurbishment (retubing) requires R&D to develop the tooling necessary to make this alternative economically competitive and R&D to develop means to reduce man-rem exposures to acceptable levels. There is currently no suitable alternative to the repair of the Turkey Point steam generators.

There are two principal societal considerations associated with repair, viz., the duration of the unit outage and man-rem exposure. Repair of a unit could involve an estimated exposure of 2084 man-rem. 7

A 270-day outage at about a $535,000/day replacement power cost has a worth of about $145,000,000. Clearly then, any emphasis for reducing societal costs should be focused on reducing unit unavailability. This is reinforced due to the fact that man-rem associated with repair will be offset by a substantial reduction in operating man-rem subsequent to repair 7
with a net man-rem societal savings over the lifetime of both Turkey Point units.

Both near term barge shipment and on site storage and decommissioning of the existing steam generators meet the guidelines of the ALARA principle. Chemical decontamination of the entire steam generator or reactor coolant system does not meet the guidelines of this principle.

TABLE 7.1-1

COMPARISON OF THE REPAIR OPTIONS

	Repair Steam Generators	In-place Refurbishment	Continuing Operation in Present Mode	Continuing Operation with Corrosion Inhibitor	Derating	Replacement with Fossil Units	Replacement with New Nuclear Units	
Comparative Capital Costs	$ 89.3 $/kW	272 $/kW	A dollar value has not been estimated for this option.	The state-of-the-art for corrosion inhibitors must be further developed by technical research before meaningful evaluations can be made with respect to effects on cost, reliability, reserve margin, and public health and safety.	A dollar value has not been estimated for this option.	1357 $/kW - Coal 370 $/kW - Gas Turbine (The installation of gas turbines would increase the nation's reliance on petroleum products, which we perceive to be contrary to the National Energy Policy.)	1979 $/kW	7
Effect on Reliability	Following the repair, the steam generators will have state-of-the-art internals and tubes. Reliability, therefore, would be enhanced because shutdowns for tube plugging would likely be eliminated. Shutdowns for tube inspection will be reduced significantly.	Following the refurbishment, the steam generator will have state-of-the-art internals and tubes. Reliability, therefore, would be enhanced because shutdowns for tube plugging would likely be eliminated. Shutdowns for tube inspection will be reduced significantly.	Outages will be required for inspecting and plugging steam generator tubes. It is conceivable that continuation of the corrosion mechanism could eventually result in unacceptable inspection intervals and reductions in unit power.		Even though derating occurs, outages will be required for inspecting and plugging steam generator tubes.	The reliability of large coal units is comparable to that of nuclear units of similar size. Other units, such as combustion turbines, are not considered suitable for base load service.	Reliability would be enhanced because shutdowns for tube plugging would likely be eliminated. Shutdowns for tube inspection will be reduced significantly.	2
Effect on Reserve Margin	Reserve margin would be reduced during the outage period.	Reserve margin would be reduced during the outage period.	Continued steam generator corrosion could result in reduction in unit power or, possibly, total unit shutdown. Should either occur, system generating capacity reserves would be reduced and replacement power would eventually be required. (Currently, firm purchase power is not expected to be available in necessary quantity after 1981.)		As the units are derated and possibly, eventually, shut down, system generating capacity reserves would be reduced and replacement power would eventually be required. (Currently, firm purchase power is not expected to be available in necessary quantity after 1981.)	Reserve margin could be reduced between the time that the nuclear units would be derated or shut down and the replacement fossil units are fully operational.	Reserve margin could be reduced, and replacement power may be required between the time that the nuclear units would be derated and possibly, eventually, shut down and the replacement nuclear units are fully operational. (Currently, firm purchase power is not expected to be available in quantity after 1981.)	
Effect on Public Health and Safety	During and following the repair, radioactive releases should be less than experienced in the period July 1978 through June 1979 at Turkey Point Units 3 and 4. A net savings of man-rem is expected to be attained after 9 years of operation post-repair.	Further developmental work is required for this option to accurately predict radioactive releases during the outage. However, it is expected that radioactive releases and man-rem doses over the life of the plant will be reduced by implementing refurbishment.	The inspection and plugging criteria for steam generator tubes and the reduction in unit power, if required, are established to ensure that there are no undue risks to the public. While some small amount of radioactivity is released as a result of primary to secondary leakage, all plant parameters and radioactive releases will continue to be within Technical Specification limits.		The inspection and plugging criteria for steam generator tubes and the reduction in unit power are established to ensure that there are no undue risks to the public. While some small amount of radioactivity is released as a result of primary to secondary leakage, all plant parameters and radioactive releases will continue to be within Technical Specification limits.	There are some public health and safety effects associated with fossil power plants due mainly to the release to atmosphere of combustion exhaust products.	With new nuclear units, radioactive releases and man-rem doses will be reduced significantly.	7

TABLE 7.4-1

COMPARISON OF IN-PLACE
STEAM GENERATOR REFURBISHMENT TO STEAM GENERATOR
LOWER ASSEMBLY REPLACEMENT

	In-place Refurbishment	Lower Assembly Replacement	
Advantages	1. No reactor coolant system cuts are required.	1. The repair can be accomplished using existing technology.	
	2. Less containment internal concrete must be removed/replaced when compared to lower assembly replacement.	2. Lower cost is incurred than for in-place refurbishment.*	
	3. Rigging and transport loads are lighter than those for lower assembly replacement.	3. Radioactive steam generator lower assemblies may be easily sealed and stored as a unit thereby eliminating the cut up, packaging, and shipping of radioactive steam generator materials.	7
	4. The repaired steam generators will have state-of-the-art internals and tubes.	4. Lower airborne activity levels may occur since no cutting of contaminated tubes is required.*	
		5. The repaired steam generators will have state-of-the-art internals and tubes.	2
Disadvantages	1. Higher cost is incurred than for lower assembly replacement.*	1. Reactor coolant system cuts are required.	
	2. Considerable difficulty is associated with handling, storage and ultimate disposition of the individual contaminated tubes.*	2. More containment internal concrete must be removed/replaced when compared to in-place refurbishment.	7
	3. Higher airborne activity levels may occur due to the cutting of the individual contaminated tubes.*	3. Rigging and transport loads are heavier than those for in-place refurbishment.	

* These advantages/disadvantages are based on the current state-of-the-art. Research and development activities may eventually eliminate these current advantages/disadvantages.

TABLE 7.4-2

ESTIMATED CAPITAL COSTS ASSOCIATED WITH IN-PLACE STEAM GENERATOR REFURBISHMENT AND LOWER ASSEMBLY REPLACEMENT

	Refurbishment* Cost (dollars)	Repair Costs (dollars)
Cost of Steam Generator Equipment	$ 20,600,000	$ 25,000,000
Development Program for Retubing	82,400,000**	-0-
Engineering Cost	7,700,000	5,200,000
Field Costs	29,000,000	58,300,000
Power Resources Costs	3,200,000	3,200,000
FPL Support Costs	4,300,000	4,300,000
Construction & Engineering Fees	1,400,000	2,400,000
Allowance for funds used during construction	17,100,000	11,000,000
Contingency	14,900,000	9,900,000
TOTAL	$180,600,000	$119,300,000

7

* Scoping estimate based on current available technology.

** This amount also includes field use of tooling and technical support of retubing.

8.0 COST BENEFIT ANALYSIS FOR THE REMOVAL, STORAGE AND DISPOSITION OF THE LOWER ASSEMBLIES CONSIDERING ALARA

8.1 INTRODUCTION

Steam generator lower assembly removal, storage, and disposal were evaluated. The cost of each was determined and compared with benefits gained in the total man-rem reduction to workers in accordance with the philosophy of reducing worker dose to levels which are ALARA. The following are evaluated herein: (a) In-place decontamination, and (b) steam generator storage and disposal methods.

a. The potential benefits of in-place decontamination of the steam generators must be derived from the increased cost for the decontamination plus the increased outage time, weighed against the net savings in man-rem as well as any potential gain in labor productivity.

b. The optimum method for steam generator storage and/or disposal must be derived from the cost associated with the storage facility, activities related to preparation of the lower assemblies for shipment, e.g., possible decontamination, cut up, packaging, etc., shipment to a disposal site, and burial, 1
weighed primarily against the net savings in man-rem.

8.2 STEAM GENERATOR IN-PLACE DECONTAMINATION

If the steam generators were to be decontaminated prior to removal, the reduction in worker dose is expected to offset the cost, schedule, licensing, and other considerations associated with the decontamination activities. Although decontamination of commercial reactor systems has been accomplished successfully by chemical means at Shippingport and electro-chemical means at Surry, the state-of-the-art for these processes for an entire steam generator has 7
not convincingly demonstrated that the benefits outweight the problems and costs. FPL plans to only decontaminate those portions of the primary system that will result in an overall man-rem reduction during the replacement operation. The decon methods described in 3.3.5.3.2 can be man-rem effective in the decontamination of a channel head or of coolant pipe, but are not adaptable to decontamination of the steam generator tube bundle.

8.3 STEAM GENERATOR STORAGE AND DISPOSAL

The steam generator lower assemblies will be stored onsite, at least temporarily, prior to eventual disposal. The alternatives of storage and disposal of the steam generator lower assemblies following their removal from the containment are addressed in this section. The evaluation was divided into two basic categories: (1) method and time period for storage, 7
and (2) preparations and methods for disposal.

8.3.1 Lower Assembly Storage and Disposal

Three basic concepts were evaluated: (1) storage until the end of the plant lifetime then disposition during decommissioning, (2) interim short term storage prior to cut up and shipment to a burial site, and (3) immediate 7
barge shipment to a licensed land burial site.

8.3.1.1 Long Term Lower Assembly Storage

As discussed in Section 3.4.6, the Units 3 and 4 lower assemblies will be sealed prior to storage. This results in a complete encapsulation of residual contamination inside the steam generator. Release of airborne or
liquid radioactive materials will not occur; therefore, enclosed storage is 1
not required (i.e., floor and roof need not be provided). However, shielding would still be provided around the assemblies and access control measures would be provided during the storage period.

At the end of the plant lifetime, disposition of the steam generators will be accomplished in conjunction with plant decommissioning.

8.3.1.2 Lower Assembly Storage with Provisions for Interim Short Term Cut Up and Shipment

If the lower assemblies are to be cut up for shipment, additional contamination control measures, as well as shielding, would have to be employed as discussed in Subsection 3.4.3.1. For this option, an enclosure with appropriate controls for airborne and liquid effluents will be required in addition to the requirements set forth in Section 3.4.2.

8.3.1.3 Immediate Barge Shipment

As discussed in Subsections 3.4.3.2 and 3.4.3.3, the Units 3 and 4 lower assemblies will be sealed prior to removal from the containment. This results in a complete encapsulation of residual contamination inside the steam
generator. Release of airborne or liquid radioactive materials will not 7
occur. Upon leaving the containment, each steam generator will be taken to a temporary laydown area, placed into a licensed Type A cask, then placed onto a barge and shipped, one at a time, to a licensed land burial site. Temporary shielding will be utilized as necessary to reduce radiation exposure.

8.3.2 Cut Up Preparation and Method of Disposition

The lower assemblies will be stored and prepared for disposition by one of
the methods discussed in Sections 3.4.2 or 3.4.3. 1

8.3.2.1 Near Term Cut Up and Disposal Without Decontamination

Man-rem burdens associated with cut up and disposal of the lower assemblies without decontamination, as shown in Section 3.4.4, are relatively high. Techniques for field cut up and packaging are not well established which makes accurate cost prediction uncertain.

8.3.2.2 Near Term Cut Up and Disposal with Injection of Solidification Agent

For the purposes of cut up, the contamination encapsulated in the lower assemblies can be "fixed" during cutting operations by injecting a solidification agent. This is as an alternate to decontamination of the lower assemblies in storage.

The lower assemblies would be prepared by cutting the nozzle closure plates and welding inlet and outlet connections prior to the injection of the 1 solidification materials. The materials would be pumped into the steam generator channel head and tubes and allowed to solidify.

Injecting the solidification agents into a complex geometry such as the steam generator and having assurance that most of the 3260 tubes will get filled has a large uncertainty at this time. A research and development (R&D) program would be required to define materials, methods and equipment requirements to achieve the desired results. Without this R&D program, precise cost and man-rem predictions are difficult to assess.

8.3.2.3 Near Term Cut Up and Disposal with Chemical Decontamination

Decontamination of the lower assemblies while in storage can be done similar to the in place decontamination discussed in Section 8.2. However, it can be performed with less cost and less worker dose in storage since more space is available to set up equipment, shielding, etc. Additionally, the task can be deferred to take credit for the additional decay time of radioisotopes.

As with the case of solidification, only partial success may be achieved. This is due to the fact that some of the tubes are already plugged. Difficulty is also expected in wetting all of the tube surface area since it may not be possible to force all of the air out of the tubes.

The cost estimate includes the decontamination process plus the cut up, packaging, shipment and disposal. It was assumed that the fewest number of cuts would be made which would meet size, weight and radioactive shipping restrictions. Torches or cutting machines would be employed in a semi-automatic manner as appropriate to minimize worker dose.

8.3.2.4 Long Term Storage with Deferred Cut Up and Disposal

The lower assemblies can be stored sealed in open storage until the end of the plant lifetime. At that time the radioactivity will be less than 1 percent of the levels expected when the lower assemblies are first placed in storage. Cut up and disposal can be accomplished by one of the methods discussed above. At that time, temporary contamination control measures, as discussed in Subsection 3.4.3.1, would be employed.

8.3.2.5 Lower Assembly Disposition During Decommissioning

If the lower assemblies are stored until the plant is decommissioned, radioactive contamination will be less than 1 percent of the present levels. Disposal during decommissioning will therefore result in lower man-rem and lower costs for contamination control and disposal efforts. The costs associated with this option include the surveillance program outlined in Section 3.4.6. Decommissioning costs approximate those in the AIF decommissioning study (Reference 1). 7

8.3.2.6 Immediate Barge Shipment

The lower assemblies will be removed from containment, placed in a barge and shipped to a licensed land burial facility. Shielding will be provided to meet DOT transportation regulations.

8.3.2.7 Conclusions 7

The evaluations presented in this section are based on the steam generator lower assembly configuration which results if the repairs were conducted by means of the pipe cut approach. If the channel cut repair approach were utilized, the resulting lower assemblies would be smaller and contain fewer curies of activity, thus requiring fewer cuts and less handling during disposition. Hence, man-rem and radioactive release effects resulting from the disposition of the lower assemblies removed by the channel cut repair approach are bounded by the discussions below.

The cost and man-rem estimate (see Section 3.4.4) for each method of lower assembly disposition are summarized below. It is seen that the lowest cost is associated with near term barge shipment, and the lowest man-rem burden 7
is associated with the long term, onsite storage and decommissioning of the lower assemblies.

		Cost	Man-Rem
a.	Cut up and disposal near term with no decontamination	$5,320,000	1500-3050
b.	Cut up and disposal near term with solidification agent	$4,930,000	800-1650 7
c.	Cut up and disposal near term with decontamination	$5,540,000	250-1150
d.	Long term storage with deferred cut up and disposal	$3,470,000	70-90
e.	Near term barge shipment	$2,620,000	53-56
f.	Long term storage with disposition during decommissioning	$3,000,000	51-53

As indicated in Section 3.4.4, the occupational exposure associated with onsite storage of six steam generator lower assemblies, without decontamination, for a period of 35 years and subsequent disposition, is less than 90 man-rem. On the other hand, the exposure associated with near-term cut up and disposal is greater than 250 man-rem. Therefore, it is readily apparent that near-term 7
cut up and disposal of the lower assemblies is not cost beneficial when compared with long-term storage or near-term barge shipment and in fact is detrimental from a man-rem exposure standpoint.

The steam generator lower assemblies will ultimately be disposed of in a licensed land burial site or decommissioned with the plant. Radiological considerations associated with short-term barge shipment and long-term 7
storage and decommissioning are acceptable. Thus, the final decision on the alternatives to be employed for storage, handling, packaging and shipping of the lower assemblies will be based on economics and man-rem exposure.

8.3.3 References

1. AIF Decommissioning Study, AIF/NESP-009, November 1976, Chapter 3 and Appendix A. 1

8.4 SINGLE UNIT CONSIDERATIONS

The discussions above are based upon repair of both nuclear units at Turkey Point. For the reasons cited below, the conclusions reached remain valid if only one unit is repaired.

Costs per steam generator will be greater if only one unit is repaired since fixed costs associated with a given operation are spread over three rather than six steam generators. For example, equipment to process decontamination fluid must be installed whether or not three or six steam generators are decontaminated.

1. If safety-related equipment can be affected by the postulated toppling of a crane, provide the criteria that will be applied to cranes regarding seismic events and high winds.

RESPONSE

All areas of the plant where crane usage is anticipated have been evaluated for rigging incidents resulting from mechanical failure, operator error, or crane toppling. The evaluation, summarized in Subsection 5.2.1.2, concludes that buried electrical duct banks and intake cooling water piping in the potential strike zone can tolerate a postulated lower assembly drop with some additional protection and that the other analyzed structures can tolerate the postulated incidents with no additional protection. Safety-related functions will not be adversely affected by a postulated toppling of a crane. Therefore, special seismic/high wind criteria which exceeds normal construction practices are not required.

2. Discuss the potential for a rigging incident occurring such that either of the Refueling Water Tanks, Primary Water Storage Tanks, or H_2 and N_2 Building are impacted and removed from service. Can a postulated drop of a steam generator lower assembly adversely affect any of these structures?

RESPONSE

Rigging operations, including steam generator lower assembly handling, will be conducted in areas sufficiently removed from the Refueling Water Tanks, the Primary Water Storage tanks, and the H_2 and N_2 Building to preclude damage to these structures due to postulated rigging incidents. Furthermore, to preclude damage to these structures due to a postulated unloaded crane boom drop during onsite crane movements, crane booms will be in the lowered position when traversing the East Plant Road in the vicinity of the nuclear units.

3. A discussion has been presented addressing possible transportation incidents involving safety-related buried piping, duct work, and other structures outside containment. It is stated in Section 5.2.1.2 that rigging incidents involving the containment itself need not be evaluated since all the fuel is removed from the core during the repair. Are there any components or systems inside the containment where the steam generators are being repaired that, if damaged or removed from service, could have any affect on the operating unit? Since there will be a considerable amount of construction activity and lifting and movement of heavy components and equipment taking place inside the containment, discuss whether a postulated accident inside the containment can adversely affect the operating unit.

RESPONSE

A review of Unit 3 and 4 safety-related systems indicates that no postulated rigging incident inside the containment of the shutdown unit could adversely affect the safety-related functions of the operating unit.

4. Indicate whether a postulated equipment handling accident inside containment could compromise the leak tight integrity of the spent fuel pool.

RESPONSE

A postulated equipment-handling incident inside containment could not compromise the leak-tight integrity of the spent fuel storage pool.

During the repair outage, all of the fuel in the reactor will be transferred to the spent fuel storage pool which is located outside the containment in the spent fuel pool area of the auxiliary building. No fuel will remain in the containment. Therefore, no postulated rigging incident inside containment could result in an object impinging on the fuel or the spent fuel storage pool.

With regard to the fuel transfer tube, in addition to the fuel transfer tube flange and cover plate in the refueling canal inside containment, there are two barriers located outside containment that will prevent the spent fuel storage pool from draining:

a. The fuel transfer tube isolation valve, located at the termination of the fuel transfer tube in the auxiliary building fuel transfer canal, will be closed.

b. The spent fuel pool seal gate, which is removed from the spent fuel storage pool keyway during refueling to allow movement of the fuel under water between the pool and the auxiliary building fuel transfer canal, will be in place.

5. If the Intake Cooling Water (ICW) lines of one reactor unit fail, does the ICW line of the other reactor have sufficient capacity to provide the necessary cooling required for operation of the spent fuel pool of both units? Please explain.

RESPONSE

If the intake cooling water (ICW) lines of one reactor unit fail, the ICW lines of the other unit have sufficient heat removal capacity to cool the spent fuel pools for both units. With one unit under repair and the other unit operating the CCW heat loads and flow requirements, assuming total loss of ICW capability on the operating unit, are as follows:

Heat Source	Heat Load (BTU/hr.)	Flow to Remove Heat Load (gpm)	
	Unit Under Repair		
Spent fuel pool-conservatively assuming 4 - 2/3 cores are stored and accident occurs 10 days after shutdown for full core off-load	$18.6X10^6$	2800	5
	Operating Unit Brought to Hot Shutdown		
Spent fuel pool-conservatively assuming 4 - 2/3 cores are stored and the accident occurs 150 hours after shutdown for normal refueling	$8.8X10^6$	2225	5
Containment air coolers	$1.6X10^6$	560	
Miscellaneous	$0.1X10^6$	415	
Total	$29.1X10^6$	6000	

A single ICW/CCW train for either unit (one ICW pump at 16,000 gpm, one CCW pump at 7500 gpm and two CCW heat exchangers) has the capability of removing $39X10^6$ BTU/hr. of heat load with 7500 gpm of CCW flow. This is sufficient cooling capability to supply the cooling needs of its own spent fuel pool in addition to supplying 3200 gpm of CCW through the Unit 3 to Unit 4 CCW intertie to provide the cooling needs of the affected unit.

6. (a) Under accident analysis, evaluate the possibility of a steam generator striking the radwaste building or fuel handling building, including its cooling capability, and the resulting radiological consequences. (b) Describe the route which will be taken by each steam generator from both containments to the temporary storage area and from the temporary storage area to the site boundary. (c) Describe the hauling device which will be used to transport the steam generators and the loading and unloading of the steam generators. (d) Evaluate the consequences of the crane operator becoming incapacitated while lifting a steam generator outside containment. (e) Evaluate the potential consequences of the crane tipping with a steam generator at its highest lift position. (Parentheses added.)

RESPONSE

a. As discussed in Subsection 3.1.4.2, a suitable lifting device would be utilized at the containment equipment hatch to transfer the lower assembly from the temporary access platform to a transporter. During the transfer onto the transporter the lower assembly movement would be confined to a zone either directly above the platform or directly above the transporter. At no time would the radwaste building or fuel handling buildings come within the strike zone for a postulated lower assembly drop, therefore precluding either a radiological release or the loss of spent fuel pool cooling, due to a lower assembly striking these structures.

An alternate approach for lower assembly ingress and egress from containment has also been evaluated for transferring the steam generator lower assemblies at the equipment hatch to the trailer transporter. This approach would not require the use of cranes for loading and unloading operations in the equipment hatch construction area. The alternate is based upon the following:

1. In lieu of moving the steam generator assemblies through the hatch onto a service platform and then using cranes for transfer to the transporter, the assemblies would be moved through the hatch directly onto the transporter.

2. The existing containment access ramps would be modified as shown in Figures A.6-1 and A.6-2 to provide a level work area. This would allow the positioning of the transporter bed at the elevation of the equipment hatch to permit simple transfer of the lower assembly thru the equipment hatch directly onto the transporter, as shown in Figure A.6-3.

The figures are provided to demonstrate feasibility of implementation. Engineering details may change as the design is finalized; however, any changes will not alter the envelope

of construction incidents postulated for this alternate approach. Construction related events and incidents were analyzed based upon this alternate approach and the results of the analyses are summarized in the following paragraphs.

Protection of Buried Facilities

Due to the fact that the alternate approach does not require total removal of the containment access ramp, the distribution of the transporter surcharge loads, through the existing fill for the ramp, was analyzed. The results indicate that the transporter surcharge loads will provide higher factors of safety at locations B and H (see Figure 5.2-1) than previously analyzed incidents. Factors of safety for previous analyses are shown in Tables 3.1-1 and 3.1-2.

Postulated Steam Generator Lower Assembly Drop

Analyses were performed to examine buried safety related facilities in the highly unlikely event that a steam generator lower assembly dropped at locations B and H during the transfer from the equipment hatch to the transporter (see Figure 5.2.1 for locations).

The following assumptions were made for the purpose of these analyses: 5

1. During handling, the lower assembly is dropped from the transporter approximately 5' to the ground directly over the buried structure.

2. No additional protection, such as timber matting, structural bridging or added fill is considered in the analyses.

3. Ramp elevation is at approximately 25'-0".

Results of the analyses indicate that these buried facilities are capable of withstanding the postulated steam generator lower assembly drop without additional protection.

Overturning of a Loaded Transporter

Analyses were performed to determine the capability of the following structures to withstand the impact caused by an overturning trailer (see Figure 5.2.1 for locations).

LOCATION B (approx.) - Radwaste building wall

LOCATION I - Unit 3 auxiliary building walls (spent fuel pool area).

The following assumptions were made for the purpose of these analyses:

1. The structure is located within impact range of the loaded transporter.

2. The ramp fill collapses causing the loaded transporter to tilt.

3. The lower assembly drops from the transporter at the top of the ramp then rolls off the edge of the ramp and impacts the structure.

Results of these analyses indicate that the analyzed structures are capable of withstanding the impact loads without loss of integrity.

Postulated Loaded Transporter Brake Failure

Analyses were performed to determine the capability of the radwaste building to withstand the impact caused by a postulated brake failure of a loaded transporter traveling down the ramp. Due to the orientation of the transporter at the equipment hatch, neither the Unit 3 auxiliary building nor the diesel generator building could be impacted by this postulated incident.

The following assumptions were made for the purpose of these analyses:

1. The brakes fail on a loaded transporter at the top of the ramp.

2. The loaded transporter travels from the top of the ramp to the bottom of the ramp prior to impact.

3. The loose fill embankment extending the full length of the radwaste building, as shown in Figure A.6-2, will be impacted prior to impact with the radwaste building wall.

Results of these analyses indicate that the radwaste building is capable of withstanding the impact loads, which are distributed through the embankments to the walls, without loss of integrity.

b. The Unit 4 removed lower assemblies will be transported south on the containment access road, around the southern end of the radwaste building, then east along the road south of the radwaste building to the temporary onsite storage facility which will be located in the laydown area shown on Figure 3.1-2. 7

The Unit 3 removed lower assemblies will be transported northeast on the containment access road, south along the east plant road, to the storage facility. 7

Should it be decided at some future time to transport the removed lower assemblies (either intact or cut up) to a land burial site, the following routes may be utilized:

1. Transport from the storage facility north along 7

the east plant road to a point north of Unit 1. At this point the lower assemblies could be loaded on barges in the turning basin or transported off site through the north gate.

2. Transport from the storage facility west along the road south of Unit 4, across the discharge canal bridge, then offsite. This route does not require transport of the lower assemblies over any safety related piping or duct banks. 7

c. There are two types of hauling devices under consideration for steam generator transport:

1. A flat-deck, multi-axle, 250 ton-capacity trailer with jeep dolly. The trailer would be supported by 4 rear axles with 8 tires each, 2 jeep axles with 8 tires each and 2 prime mover axles with 4 tires each. The bed would measure approximately 12' X 42'.

2. A 250 ton-capacity dolly system utilizing steel framing for lower-assembly support. The framing would be supported at each corner by a 16-wheel dolly set; the prime mover would be separate. The dolly set gauge would be adjustable to provide acceptable bearing pressure.

d. As discussed in the response to Question 1 of this Appendix, the results of analyses summarized in Subsection 5.2.1.2 are applicable to all crane incidents including those caused by operator malfunction.

e. As discussed in the response to Question 1 of this Appendix, the results of the analyses summarized in Subsection 5.2.1.2 are applicable to all crane incidents including those caused by a crane tipping with a lower assembly at its highest lift position.

7. Provide technical justification for not complying with the recommendations of Regulatory Guide 1.71. Welder Qualifications for areas of Limited Accessibility, or indicate alternative procedures that will be used to meet the intent of these recommendations.

RESPONSE

Westinghouse practice does not require qualification or requalification of welders for areas of limited accessibility as described by Regulatory Guide 1.71. Experience shows that the current Westinghouse shop practice produces high quality welds. In addition, the performance of required nondestructive evaluations provides further assurance of acceptable weld quality.

Limited accessibility qualification or requalification, in excess of ASME Section III and IX requirements, is unduly restrictive for component fabrication, where the welders' physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised.

In lieu of welder qualification in restricted access areas, field welds associated with the repair will be given 100 percent radiographic examination in accordance with the codes noted in Section 3.2.4.

8. Provide information to ensure that the on-site quality assurance requirements for repair related components that are part of the primary reactor coolant pressure boundary will meet the appropriate recommendations of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.

RESPONSE

FPL and Bechtel comply with the intent of Regulatory Guide 1.37 with minor deviations as stated in the "FPL Topical Quality Assurance Report" (FPLTQAR 1-76A), Revision 2, September 8, 1977, and the "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report BQ-TOP-1, Revision 2A, July, 1977, respectively. 7

9. Provide additional information on the repair and inspection procedures that are intended to be used, especially regarding the welding, heat treatment, and examination procedures for the upper to lower assembly shell weld.

RESPONSE

The repair and inspection procedures for the upper to lower assembly shell weld are under development and will incorporate the applicable requirements of the ASME Boiler and Pressure Vessel Code.

10. In Section 2.1.4, a discussion of Regulatory Guide 1.44 indicates that Westinghouse weld procedures will prevent sensitization of stainless steel. Provide a detailed description of the field cutting and welding and/or post-weld heat treatment procedures that will be used to ensure that the stainless steel piping will not become sensitized during the repair procedure. Provide a weld procedure qualification test program that will employ the same weld procedures to be used in the field repair. This test qualification program should incorporate Position C.3 of Regulatory Guide 1.44.

RESPONSE

Field welding and post-weld heat treatment procedures for stainless steel piping will comply with Regulatory Guide 1.44 as discussed in revised Section 3.7.

Field cutting of stainless steel piping may employ either thermal or mechanical cutting techniques. Mechanical cutting techniques would not result in the time/temperature conditions for sensitization. If thermal cutting techniques are employed, the cut would be made as near to the steam generator as possible. Any sensitized material would be removed during the preparation of the piping for welding to the repaired steam generator.

Stainless steel welding procedures to be employed on the reactor coolant system are qualified to the requirements of ASME Section IX.

Corrosion testing to ASTM A262 is not necessary since temperatures of 800-1500° will not be present during the repair work except in the weld zone itself. Established procedures control the heat input at welds by restricting the size of the electrodes used and by limiting the interpass temperature to 350°F maximum. Experience has shown that when these procedures are employed, sensitization of stainless steel in the heat affected zone is minimal and insufficient to cause stress corrosion cracking.

11. Paragraph 2.1.4 of the Steam Generator Repair Report specifies the regulatory guide applicability to the fabrication of the replacement lower assemblies. Reference is made to the Westinghouse QA Plan WCAP-8370, Revision 7A and also Revision 8A. It is not understood why the applicability should vary from one regulatory guide to another. A commitment to the latest revision, 8A, would be acceptable for all references or if the assemblies have been fabricated to the previous revision, then the commitment to Revision 7A is also acceptable. Please clarify.

RESPONSE

This question is addressed in revised Section 2.1.4.

12. Describe the FPL QA program applicable to the field operations affected by the replacemnt of the steam generators. A commitment of FPL Topical QA Report FPLTQA 1-76A, Revision 2 is acceptable.

RESPONSE

The FPL QA program applicable to field operations is described in the FPL Topical QA Report (FPLTQAR 1-76A), Revision 2. See revised Section 3.6.1.

13. Regulatory Guide 1.38 "QA Requirements for Packaging, Shipping, Receiving, Storage, and Handling for Nuclear Power Plants," has not been included in the list of applicable regulatory guides specified in paragraph 2.1.4. And, while not yet endorsed by a regulatory guide (but acceptable to NRC), the Requirements for Auditing of QA Programs for Nuclear Power Plants, ANSI N45.2.12-Draft 3, Revision 4, February 22, 1974, have also not been included in the list. Please include these in the list or provide information as to why they should not be included.

RESPONSE

FPL and Bechtel comply with the intent of Regulatory Guide 1.38 and ANSI N45.2.12 with minor deviations as stated in the "FPL Topical Quality Assurance Report," (FPLTQAR 1-76A), Revision 2, September 8, 1977, and the "Bechtel Quality Assurance Program for Nuclear Power Plants," Topical Report BQ-TOP-1, Revision 2A, July 1977, respectively. 7

The Westinghouse position on Regulatory Guide 1.38 presented in revised Section 2.1.4 of this report. The Westinghouse position on ANSI N45.2.12 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

14. The new steam generator lower assembly will have two 2-inch Schedule 40 Inconel internal blowdown pipes and a 3/8-inch primary shell drain. Also, a 2-inch nozzle is being added to the upper shell to facilitate wet layup conditions. Because of these changes, indicate whether there will be any additional piping, valves or other components external to the steam generator or modifications to existing components. If so, describe the interface with existing systems, provide quality group classifications, and discuss any stress analyses that must be performed.

RESPONSE

The two 2-inch Schedule 40 Inconel internal blowdown pipes will allow higher capacity blowdown from the steam generators as compared to the present internal blowdown pipe. As provided in the response to Question 29 of this Appendix, this design refinement would facilitate increasing the steam generator blowdown system capacity in the future if it is determined to be cost beneficial or desirable for operational flexibility. Any changes to the steam generator blowdown system will be done in accordance with existing plant change/modification procedures. Similarly, any changes related to either the 2-inch nozzle added to the upper shell or the 3/8-inch primary shell drain, will be done in accordance with existing plant change/modification procedures.

15. Present your preoperational testing program and your startup testing program for placing a unit back in service with the modified steam generators. Identify all the systems and instrumentation to be tested or recalibrated.

RESPONSE

The preoperational and startup test program is still being developed at this time and thus, certain details remain to be determined. The objective of the test program is to ensure that the plant is returned to safe and reliable full power operation. To meet this objective, the program will require testing of all newly installed equipment, as well as testing of those pieces of equipment which have been affected by the repair. In addition, the program will include testing of safety-related equipment in accordance with the Technical Specifications requirements and other testing that is normally performed between routine fuel loadings and operation at full power.

The test program, as currently envisioned, can be summarized as follows:

a. Perform "pre-operational" checks and inspections to ensure that all newly installed safety-related equipment and equipment that has been affected is ready to be tested functionally. This will include such things as flushing and cleaning, leak checks, wire continuity checks, calibration checks, verification of valve lineups, etc..

b. Perform functional checks of equipment that has been newly installed. This will include such things as steam generator water level instrumentation checks, steam generator thermal performance test, steam generator blowdown performance test, etc.

c. Perform functional checks of any equipment that has been affected by the repair. This will include items such as starting and running pumps, operating valves, etc.

d. Perform normal surveillance of equipment in accordance with Technical Specifications during the outage, such as valve operability, pump operability, etc.

e. Startup testing, which includes those checks, tests and operations that are normally performed between routine fuel loadings and operation at full power, such as rod drop tests, low power physics tests, etc.

16. Discuss the possible effect the new support plate design and/or material could have on the ability to perform eddy current inspection and to interpret the results at the tube/tube support plate intersections. Also, how will full depth expansion of the tubes in the tubesheet affect inspection in this area?

RESPONSE

The use of ferritic stainless steel (Type 405) is not expected to result in any loss of defect visibility since both 405 and carbon steel are magnetic. Two factors in the design of the quatrefoil support will increase the visibility of tube defects in the support plate region. The first of these factors is the increase in the material thickness from 3/4 inch used for carbon steel plates to 1-1/8 inches. The increased thickness will reduce the interference associated with the edges of the plate as the probe moves through the plate. This effect is favorable in increasing the ability to "see" defects although it is a minor influence. The second effect is associated with the effective average distance of the support plate from the tube. For the drilled carbon steel support plate design, the carbon steel surrounds the tubes at a distance of a few mills. The quatrefoil plate will be in close proximity to the tube only at the four land areas. The increased average distance of the support plate from the tube will reduce the interference from the support plate and allow some increased visibility of any possible tube defects.

Full depth expansion of the tubes in the tubesheet will not reduce the ability to ascertain the tube integrity. The effect of the transition from the expanded portion of the tube in the tubesheet to the unexpanded region at the top of the tubesheet is predictable and can be accounted for in eddy current signal analysis. The transition region exists on current units with partial expansion; therefore, there is no net effect on eddy current inspection from the full depth expansion.

17. Provide the secondary water chemistry specifications you intend to use and the corrective plant operator actions to be taken should these expected chemistries be violated. Specifically, describe operator action following the discovery of condenser leakage.

RESPONSE

Following repair of the steam generators, FPL will maintain steam generator water chemistry in accordance with FPL-approved Turkey Point Units 3 & 4 AVT specifications and procedures. These specifications and procedures will reflect the steam generator manufacturer's recommendations and FPL's extensive operating experience. They will also provide for a proper level of equipment protection consistent with FPL's commitment to provide electrical service to its customers.

18. In Section 2.2.1.7 it is stated that the corrosion of SA-240 results in an oxide with approximately the same volume as the parent material. Provide the data bases for this statement.

RESPONSE

In the initial corrosion tests in acid chloride environments, it was observed that the Type 405 specimens formed a thin, tight, adherent corrosion film with little change in overall specimen dimension, implying that the volume of the oxide formed was essentially equivalent to that of the parent metal consumed. This judgement was confirmed by careful metallographic examination: portions of 405 and carbon steel coupons were nickel plated and then immersed for 40 days in a 0.16 $NiCl_2$ solution (5800 ppm Cl) at 475 F (an accelerated corrosion environment typical of those used in corrosion testing). The nickel plate served two purposes: it marked the original surfaces or metal-reactant interface and it protected the underlying metal from attack. The adjoining surfaces which were unplated were, of course, attacked by the solution. Subsequent metallographic examination showed that the 405 corrosion product (magnetite spinel enriched in chromium, chemically and metallurgically similar to that which is expected in the event of Cl inleakage) in the unplated region did not extend above the original surface of the specimen, as indicated by the adjacent nickel marker. Similar regions in the carbon steel specimen showed a build-up of corrosion product extending well beyond the nickel marker.

19. Indicate what actions you plan to take to minimize condenser leakage or to improve the performance of your condenser.

RESPONSE

The main condensers at Turkey Point, Units 3 & 4, have been retubed with titanium tubing. Based upon industry and FPL experience, FPL believes that titanium tubing is the best choice of tubing material currently available for approaching a leak-free condenser.

An Amertap on-line condenser cleaning system has been recently installed in both Unit 3 (in service 2/78) and Unit 4 (in service 9/79). This system has improved condenser thermal performance and is expected to reduce tube corrosion by eliminating calcium scale buildup. 7

20. Indicate your plans, if any, to make modifications to reduce the amount of copper (or other metal ions) in the condensers and heaters in feedwater and condensate system in view of the Westinghouse findings on the role of Cu^{++}(or other metal ions) on denting.

RESPONSE

Investigation of the denting phenomena has shown that the denting was caused by the production of a magnetite (Fe_3O_4) corrosion product in the tube/tube support plate annulus of the currently employed carbon steel support plates. Lab tests indicate that Cu ions enhance the carbon steel corrosion rate.

However, the replacement steam generators are being built with SA-240 type 405 SS tube support plates employing a quatre-foil design. Corrosion tests and careful metallographic examinations have confirmed that SA-240 Type 405 SS in metallic chloride environments forms a thin, tight, adherent corrosion film with little change in overall specimen dimension. The test results to date indicate that SA-240 Type 405 SS has exhibited no denting under identical test conditions where carbon steel has produced significant denting.

It should be noted that the following will reduce the amount of copper and metal ions within the steam generators:

1. An improved internal blowdown design and improved steam generator thermal hydraulics have been incorporated into the replacement steam generator and are expected to improve the efficiency for blowdown removal of sludge and corrosion products.

2. Additional access ports will be provided to enhance the sludge lancing capability of the replacement steam generators.

3. Planned condenser modifications as provided in the response to Question 19 will significantly reduce copper bearing materials in the condensate systems.

4. The feedwater heaters are being retubed with stainless steel to reduce copper bearing materials in the feedwater system.

5. The moisture separator/reheaters are planned to be retubed with stainless steel to reduce the amount of copper bearing material in the secondary side of the plant. 7

6. A full flow condensate polishing demineralizer system is planned to be installed which will demineralize the condensate flow to the steam generators. The Powdex process used by the planned demineralizer will effectively remove dissolved cations such as magnesium, sodium, calcium, copper, iron and nickel, and anions such as sulfate, chloride, and dissolved silica. In addition, the polisher will remove suspended and colloidal impurities such as silica, copper, iron and nickel.

21. We are beginning to hear of denting in the tube-to-tube sheet crevices. How does this affect the projected performance of the new design as described in Section 2.2.1.42. Although these new steam generators will have the tube expanded to the full depth of the tube sheet, there is still a potential for crevice corrosion of the carbon steel tube sheet at this location. Thermal ratcheting could serve to open up these crevices. Provide experimental results that shown that full depth expansion of the tubing into the tube sheet eliminates crevice corrosion of carbon steel in the presence of an acid chloride environment.

RESPONSE

The potential for corrosion cracking in the tube-to-tube sheet crevices would exist if both the geometry and the available superheat permit the concentration of aggressive contaminants. The repair lower assemblies for Turkey Point have been designed to minimize these conditions by permitting no more than a 1/16-inch-deep crevice to remain at the top of the secondary side of the tube sheet. This configuration has been evaluated by model boiler tests in caustic environments at steam generator operating temperature and pressure. The test results for Inconel 600 tubes in carbon steel tube sheets were as follows:

a. No caustic attack occurred in crevices intentionally formed 1/8 inch deep.

b. Caustic attack and occasional cracking occurred in crevices 2-3 inches deep.

The results were further confirmed by operating plant experience:

a. Plants utilizing phosphate or AVT chemistry control with partially expanded tube-to-tube sheet joints have observed some tube deformation but no through-wall defects at the tube sheet.

b. Plants utilizing AVT chemistry control with full expanded tube-to-tube sheet joints have observed no tube deformation at the tube sheet.

The possibility of thermal ratcheting has been evaluated by analyses which considered system pressure, thermal expansion of the tube and tube hole, tube sheet bending, and thermal gradients in the tube wall. The following results were obtained for the bounding cases studied:

a. Normal operating transients did not cause crevices to open.

b. A feedwater transient which injected 32 F water into the secondary side did not cause crevices to open.

The results of these analyses indicate that the design acceptably accommodates the thermal ratcheting potential.

22. Provide the corrosion test data for SA-240 Type 405 stainless steel and the alternative support plate materials that were evaluated as discussed in Section 2.2.1.7.

RESPONSE

Candidate support plate materials were evaluated for a variety of attributes including corrosion resistance. The corrosion tests that were performed compared the candidate material, Type 405, with carbon steel, under four basic conditions.

a. Isothermal Autoclave Exposure

Temperature:	$475^{o}F$ to $600^{o}F$
Environment:	AVT, concentrated sea water
Chloride Level:	0.15 ppm to 100,000 ppm
No. of Specimens:	250
Total Exposure Time:	14,400 hrs

b. Heater Crevice Tests

Temperature:	$540^{o}F$
Environment:	200 ppm Cl plus CuO or $CuCl_2$
Total Exposure Time:	5040 hrs

c. Heated Capsule Tests

Temperature:	$450^{o}F$ to $600^{o}F$
Environment:	$NiCl_2$, $CuCl_2$ (5800 ppm Cl)
No. of Specimens:	48
Total Exposure Time:	5040 hrs

The test results to date may be summarized as follows:

a. Isothermal Autoclave Exposure

At the lower temperatures, both carbon steel and 405 have low rates of corrosion. As the temperature increases, the carbon steel is attacked more severely. The 405 is superior by a factor of 3 to 40, depending on the specific environment and specimen configuration. (Figures A.22-1 and -2 are typical findings.)

b. Heated Crevice Tests and Heated Capsule Tests

Type 405 exhibited no denting, whereas carbon steel produced significant denting under identical test conditions (Figure A.22-3).

Other attributes of importance included machinability, weldability, availability in required sizes and quantity, the coefficient of thermal expansion, fretting and wear characteristics, and the mechanical properties. When evaluated against all these criteria, SA-240 Type 405 material was selected.

23. Operating experience suggests that some erosion/cavitation type degradation may be occurring downstream of clover leaf type support plate holes. How will this erosion problem be avoided in the new steam generators with the quatrefoil tube support plates.

RESPONSE

This phenomenon is associated with once-through steam generators which produce superheated steam. The area of the tube bundle specifically impacted by this attack is in the superheat region. The attack experienced is more clearly described as erosion-corrosion of the tubes due to impingement of contaminants in the high-velocity steam. This problem was not experienced in all liquid or two-phase regions of the steam generators. Since the recirculating type steam generator operates with liquid or two-phase fluid, this type of degradation is not anticipated and has not been observed in recirculating type steam generators.

24. In Section 2.2.1.2 it is stated that the flow distribution baffle plate material is ferritic stainless steel. How does ferritic stainless steel cope with Cl^- regarding stress corrosion cracking or Fe_3O_4 formation?

RESPONSE

Stressed U-bends of Type 405 plate were exposed for 12 weeks in an extremely aggressive environment of 100,000 ppm Cl^- at 550°F. Stress corrosion cracks were found only in the heat-affected zone of certain welded specimens which were not stress relieved after welding. No cracking occurred in the unwelded areas or in specimens which were stress relieved at 1350°F after welding. As a result of these tests, any limited welding which may be performed on the tube support or baffle plates will be followed by the appropriate stress relief.

In the large number of corrosion tests which were performed in acid chlorides at the temperatures of interest, 540° and 550°F, the corrosion product rate of formation on the Type 405 was less than that of carbon steel by factors of 3 to 40, depending on the specific environment and specimen geometry. The corrosion product was generally characterized as tight and adherent as compared to the loose, flaky, voluminous oxide on the carbon steel.

25. Section 2.2.1.5 discusses the thermal treatment of the Inconel-600 tubes. When and how is the Inconel thermally treated to relieve and/or to give resistance to primary side stress corrosion cracking? How will this heat treatment affect secondary side behavior? Also, provide the test results relative to intergranular attack in environments containing O_2 and/or sulphur. Does the thermal treatment affect the pitting resistance of Inconel in Cl media?

RESPONSE

The thermal treatment was designed to improve the corrosion resistance against secondary side attack, specifically caustic stress corrosion. The thermal treatment is performed as a final operation on the straight length tubing, after the straightening and polishing. It consists of heat treating the tubing in vacuum for a nominal temperature and time of 1300 F and 15 hours. This treatment results in a microstructure which consists of semi-continuous grain boundary precipitates but with little or no chromium depletion adjacent to the boundaries. This structure in conjunction with the almost complete relief of residual stresses, provides the corrosion resistance benefit. After bending the thermally treated tubes, the small diameter bends (up to 20" diameter) are re-treated at 1300° for several hours to reduce the stresses caused by the bending operation.

Grain boundary carbide precipitation, while beneficial in caustic and other species, may be detrimental in certain oxygenated or sulfur bearing environments if there is a chromium depleted region adjacent to the boundaries; this is called "sensitization." The special thermal treatment is specifically designed to avoid this condition; the temperatures and time selected results in a "de-sensitized" structure toward intergranular attack in environments which otherwise could be aggressive. This was demonstrated during the development and qualification phase of the program, where reference treated material was shown to be resistant to cracking in polytheonic acid and to exhibit very low weight loss in the nitric acid Huey tests. Routine testing is performed during production to assure that the material is de-sensitized.

The thermal treatment was not expected to significantly affect the pitting resistance in chlorides for better or worse. After 7 months of exposure to acid chlorides in heated capsule tests, no pitting attack has been observed in thermally treated material. In an accelerated electro-chemical test at 630 F, 3600 ppm Cl and +150 mV applied potential, stressed C-rings of mill annealed material cracked whereas the reference treated material was unattacked. Hence the thermal treatment may also be beneficial in chloride environments.

26. Describe the layup procedures and startup procedures that will preclude or minimize the introduction of corrosion products into the steam generator that may have accumulated in the steamlines and balance of plant piping during the extended layup.

RESPONSE

The response to Question 17 of this Appendix is also applicable to layup and startup procedures.

27. If decontamination is necessary, demonstrate that the methods and the decontamination solutions will not degrade or adversely affect the reactor coolant piping or components which are part of the primary system pressure boundary. Further show that the decontamination solutions will not have deleterious latent effects in subsequent plant operations.

RESPONSE

In-place chemical decontamination of the reactor coolant piping is not planned for the steam generator repair effort. Decontamination of the steam generator channel heads may be performed by one of the methods described in Subsection 3.3.5.3.2. 7

As indicated in Subsection 3.3.5.3 and in the response to Question 40 (c) of this appendix, local/spot decontamination of the steam generators and reactor coolant piping will be performed as required. Local decontamination will consist of cleaning of loose surface contamination with either soap and water of RADIAC solutions. Past industry experience has shown that these standard solutions are not delerious to the piping or components involved.

28. Describe your plans, if any, for periodic chemical cleaning in the future on the secondary side of the new steam generators. Indicate current research of efforts to qualify solvents for use in the future.

RESPONSE

FPL does not have any specific plans to chemically clean any of their steam generators at this time. FPL is, however, aware of the potential for chemical cleaning and is, therefore, continuing to remain abreast of industry development in steam generator chemical cleaning. As such, FPL has a representative on the Steam Generator Chemical Cleaning Review Board which is sponsored by the U.S. Department of Energy in conjunction with Consolidated Edison. FPL is also following steam generator cleaning work sponsored by EPRI and the Steam Generator Owner's Group.

29. Section 2.2.1.3 describes the Improved Internal Blowdown Design and states that "The improved blowdown system will allow higher capacity blowdown in comparison with the present blowdown arrangement." What is anticipated from this increased blowdown in future operation of the plant?

<u>RESPONSE</u>

As discussed in Subsection 2.2.1.3, each replacement steam generator is designed with an improved internal blowdown system which will allow higher blowdown efficiency and also allow higher blowdown capacity in comparison with the present blowdown arrangement. The current steam generator blowdown system external to the steam generators, however, does not have the capacity for increased blowdown. If it is later determined to be cost beneficial to increase the capacity of the existing steam generator blowdown system, it will be done in accordance with existing plant change/modification procedures.

As stated in Section 1.1.5 and further discussed in Section 6.6.3, the steam generator repair will, in effect, provide a reduction in future radioactive effluents from the plant because the repair will result in enhanced steam generator integrity, thus reducing primary to secondary leakage. Therefore, in the event that chemical control by increased blowdown is employed, it is not anticipated that an increase in radioactive effluents will result. Furthermore, regardless of ultimate steam generator blowdown system design, the plant will operate within the Technical Specifications, thus providing further assurance that there will be no adverse environmental impact associated with increased steam generator blowdown.

30. Section 3.3.6.3 identifies three potential sources of radioactive liquid wastes. Describe the liquid radwaste system's ability to store and process these wastes. What is the estimated quantity and radioactivity of each of these wastes? Describe the criteria which will be used to determine if the wastes will be processed thru (1) filters, (2) evaporators and/or (3) demineralizers prior to release to the environment. For each of the three potential sources of radioactive liquid, evaluate their environmental impact. Identify the equipment, including both permanently installed or mobile units, which will be available to process the wastes. Evaluate the possibility of the local decontamination solution(s) having deleterious effect(s) on the equipment employed to process it.

RESPONSE

Subsection 3.3.6.3 identifies the following three potential sources of radioactive liquid to be disposed of during the repair effort:

a. Water drained from the reactor coolant system
b. Laundry waste water
c. Local decontamination waste fluids

The reactor coolant will be processed by the chemical and volume control system (CVCS), as described in Section 9.2 of the Turkey Point FSAR. Storing and, subsequently, processing an entire reactor coolant system volume of water is a normal mode of operation for the CVCS. The holdup tanks within the system have the capacity for four reactor coolant system volumes.

As described in Subsection 3.3.6.3, the laundry waste water may be discharged without processing due to the low activity level, as indicated by the estimated laundry waste water specific activities given in Table 5.2-5. Laundry waste water which requires processing is not expected to be a normal occurrence but is within the capability of the liquid radwaste processing scheme. Information regarding sampling and criteria for processing laundry waste water is provided in response to Question 32 of this Appendix.

As indicated in Subsection 3.3.6.3, liquid waste generated as a result of local decontamination is expected to be minimal and will be treated as part of the normal liquid radwaste processing scheme. Decontamination wastes will be collected and sampled. If analyses show that the radioactivity exceeds the discharge limits of the Technical Specifications, the solutions will be processed; otherwise, they will be discharged without processing.

The following liquid waste processing systems are available to store and process radioactive liquids, excluding reactor coolant which is processed by the CVCS, as required:

a. The in-plant system consists of two 15-gpm waste evaporators, associated filters and demineralizers, waste holdup tanks with a total capacity of 35,000 gallons, and associated waste monitor tanks.

b. The in-plant system has connections installed to allow hookup to a commercially available mobile processing unit.

The estimated quantities and radioactivity of the liquids (excluding local decontamination waste fluids which will be minimal) to be disposed of are given in Subsection 5.2.2.4. As described in Subsection 5.2.2.5, the estimated total radioactive liquid release per unit (excluding tritium and dissolved gases) during the repair effort is expected to be about 13 percent of the observed total liquid waste per unit during 1976. The estimated tritium release per unit during the replacement effort is about one half the observed tritium release per unit during 1976. Based on these results, the environmental impact for each of the three potential sources of radioactive liquid is expected to be less than would occur during normal plant operation.

Local decontamination solutions will either be soap and water or a chemical agent such as RADIAC. Plant experience has shown that neither of these have any deleterious effects on the waste processing system equipment.

31. Section 3.3.6.1 describes the disposal of concrete removed from the containment. The section implies that radiological controls for removed concrete will only be established if "transferable contamination" exceeds 2200 dpm/100 cm^2. The section does not address concrete which has fixed (not loose) contamination which exceeds 2200 dpm/100 cm^2. You should clearly describe what criteria will be used for disposal of the concrete (as well as other waste materials) including unrestricted release and burial at a licensed burial site.

RESPONSE

Current Turkey Point practices will be used for disposal of concrete. These practices involve the following:

a. A frisker type instrument is used to conduct radiation surveys on an area. If the frisker indicates levels greater than 100 cpm/probe area (HP-210 probe) a Beta-gamma swipe analysis is performed. The radiation survey and the swipe analysis are used to determine the amount of fixed and transferable contamination.

b. If the frisker indicates levels (fixed and transferable) less than 100 cpm/probe area (HP-210 probe) the concrete is handled as nonradioactive waste. If the frisker indicates levels greater than 100 cpm/probe area and a swipe analysis indicates transferable contamination is present, the concrete is either decontaminated to below 100 cpm/probe area and handled as nonradioactive waste, or disposed of in shipping containers and transported to a licensed burial site. If the frisker indicates that fixed contamination is present in excess of 100 cpm/probe area the concrete is disposed of in shipping containers and transported to a licensed burial site.

Other contaminated or otherwise radioactive wastes will be packaged and shipped to a licensed burial site.

In all cases, radioactive wastes will be packaged and shipped in accordance with applicable state and federal regulations, including 10 CFR 71 and 40 CFR 170-178.

32. Section 3.3.6 states that sampling of laundry wastes will dictate whether or not processing is necessary. Describe the sampling program and identify which isotopes will be specifically assayed. Are there any plans to specifically measure beta emitters such as Fe-55 and Ni-63? If not, provide justification for not analyzing these and other beta emitters which could be present.

RESPONSE

Laundry waste water is sampled by grab sampling. The sample may be analyzed, either for gross activity, or for isotopic content and concentrations by a gamma spectrometer. If either the gross radioactivity measurement shows radioactivity greater than 3×10^{-8} μCi/cc, or isotopic concentrations are in excess of 10^{-5} μCi/cc, laundry water is processed as described in the response to Question 30. An isotope listing for the laundry waste water at Turkey Point is given in Table 5.2-5.

There are no plans to specifically measure beta emitters, such as Fe-55 and Ni-63, in the laundry waste water. In laundry waste water, gamma emitters are predominant and are consequently measured. This is further evidenced by the listing of expected isotopes for laundry waste water effluents provided in Table 2-20 of NUREG-0017, April 1976 which does not include Fe-55 and Ni-63 in discharged untreated PWR liquid detergent wastes. Furthermore, it is highly unlikely that these isotopes would enter the laundry waste water in significant quantities. Fe-55, Ni-63 and other corrosion products form a tenacious film on reactor coolant system primary side surfaces. This film would become dislodged only during cutting and weld prepping operations. During these operations, workers will wear 2 sets of protective clothing. The outer set of protective clothing will be removed when leaving the work area and deposited in a container. The inner set is not expected to be contaminated and is, therefore, currently planned for laundering and reuse.

Weekly and monthly composite samples for the liquid waste disposal system, which includes laundry waste water, are taken and analyzed for Sr-89 and Sr-90 using a chemical separation and subsequent beta determination with a 2π gas flow proportional counter. Both Fe-55 and Ni-63 are low energy beta emitters. In lieu of analyzing for these weak beta emitters, the higher energy beta emitters, Sr-89 and 90, are assayed. This approach is consistent with the guidelines of Regulatory Guide 1.21.

33. Section 3.4.5.1 states that the lower assemblies will be cut in a temporary facility which is provided with HEPA filters in the exhaust line but no mention is made of airborne radiation control for the containment as a whole. Describe the modes of operation of the containment ventilation systems during replacement activities including use of filters (ESF and non-ESF), effluent and local monitoring systems, and other features considered to control radioactive materials. Specifically, specify whether the containment will always be kept at a slight negative pressure to prevent exfiltration through the equipment hatch.

RESPONSE

This question was addressed in Revision 1 of Section 3.3.3.

34. Estimate the airborne radiation source associated with the cutting and removal of concrete and the methods and criteria that will be used in reducing these airborne concentrations.

RESPONSE

As discussed in Subsection 3.3.6.1, the majority of the concrete to be removed from the containment internal walls and floors has an insignificant amount of contamination; therefore, it is expected that the airborne radiation source associated with the cutting and removal of concrete will also be insignificant. Furthermore, as described in Section 3.3.3, airborne radioactivity inside containment during the repair effort will be controlled, monitored, and ultimately released via the plant vent stack using the containment purge exhaust system.

Proposed concrete cutting methods are intended to reduce the generation of airborne dust particles. For example, the drilling technique described in Subsection 3.2.5.1 is a water-cooled process with the benefit that the removed concrete material from the drilled hole is carried away in the form of a slurry. Retention dams and splash shields will contain and direct the slurry to maintain containment cleanliness and to control potential contaminants. The subsequent splitting process does not generate a significant amount of dust. Furthermore, dust which may result during the splitting process will generally originate from the interior of the structure; any contamination is expected to be near the concrete exterior surface.

Based on the expected concrete cutting method, it has been conservatively estimated that only 3.1 μCi would be released to the containment as a result of concrete cutting.

35. Section 3.4.5.1c assumes that all activity present in the vicinity of the cut becomes airborne. What area is considered in this assumption and what is its justification?

RESPONSE

The following kerf widths were conservatively assumed in calculating the radioactive releases and resultant doses presented in Subsection 3.4.5.2:

a. Tubesheet - kerf width of 2 inches (yielding 9.2 sq. ft. per lower assembly)

b. Channel head - kerf width of 2 inches (yielding 5.6 sq. ft. per lower assembly)

c. Tube bundle - kerf width (each tube) of 1/2 inch (yielding 235 sq. ft. per lower assembly)

Advanced cutting techniques would reduce kerf widths below that assumed in the analysis. The kerf widths listed above were assumed for the purposes of conservatively bounding potential releases.

36. Describe the program which will be implemented after the steam generator replacement, if approved, is complete which will requalify the ESF filters inside containment. Describe the program which will assure that the activated carbon in the ESF filter units will still have the capability to absorb radioactive iodine.

RESPONSE

The HEPA and activated charcoal sections will be sealed as required with plastic sheeting to prevent possible contamination during the repair effort.

During the steam generator repair outage, laboratory tests of the charcoal surveillance specimens will be continued quarterly to assure that they remain within laboratory acceptance criteria for iodine removal.

Following completion of the steam generator repair work, a full system operating test and the refueling performance test will be performed in accordance with established plant procedures.

37. Describe the procedures and methods which will be followed in painting or repainting those surfaces which, due to the replacement program, have had their surface coatings removed.

RESPONSE

All coatings removed during the repair will be replaced with equivalent or better coatings utilizing accepted FPL/Bechtel procedures.

38. Describe the procedures and methods which will be followed and post repair testing which will be performed to assure that the floor drain system which is necessary for operation of the ESF liquid systems does not become clogged with construction debris.

RESPONSE

The floor drain system inside containment necessary for operation of ESF liquid systems will be protected during the repair outage so as to prevent their becoming clogged with construction debris. On completion of the repair, the drains will be inspected and cleaned as required.

39. Provide a detailed example of dose calculations for a particular procedure. This should include:

    A. A detailed procedure outline.
    B. The time and personnel involved in this job.
    C. Actual dose rate measurements taken in the area where the job will be performed.
    D. Specific procedures and techniques used to reduce the radiation exposure during the job.

RESPONSE

A. Detailed procedures for the various tasks associated with the repair will be developed, as required, as part of the overall design and planning effort for the job. These procedures will be written with consideration for keeping radiation exposure to personnel as low as is reasonably achievable. However, such detailed procedures are not required to estimate the man-rem associated with the repair. As discussed in Subsection 3.3.7.2, the man-rem predictions reflect the best estimates that can be made at the early planning phase of the job, i.e., job average man-rem predictions.

B. Manhours involved in major tasks associated with the repair are listed in Table 3.3-2.

C. Typical job area dose rates are shown in Figures 3.3-1 through 3.3-7 and in Table 3.3-2.

D. See the response to Question 43 of this Appendix.

40. Section 3.3.5.1 states that the containment model (1/24 scale) will be used in arriving at decisions for (1) temporary shielding; (2) local decontamination; (3) work planning and (4) craft training. Please (a) Describe any other models or mock ups available for the craft personnel involved in cutting the primary system or other high exposure operations, (b) Provide the criteria to be used in establishing the need for temporary shielding, (c) Provide the criteria to be used for establishing the need for local decontamination and (d) Describe the methods of local decontamination on the integrity of the primary system.

RESPONSE

a. In addition to the 1/2" = 1'-0" scale model of the containment it is planned to construct 1/2" = 1'-0" scale models of:

1. Laydown space requirements inside containment

2. Steam generator and internals details

The laydown space requirements model will consist of detailed models of all components requiring laydown space, cribbing arrangements, component lifting points, and hydraulic cranes. These models will be used in conjunction with the containment model to prepare a photographic sequence of component removal, laydown and replacement.

The steam generator and internals model will consist of snap-together sections of the upper and lower assemblies and all internal parts that are to be removed as separate pieces or assemblies. All parts will have their access and lifting points detailed, and the upper and lower assemblies will be complete in external details including manways and portions of the reactor coolant piping beyond the cut points.

All of these models will be available to craft personnel for work planning and training.

b. Health physics procedure HP-55 provides the criteria to be used in establishing the need for temporary shielding. Each case in which temporary shielding will be used, will be treated individually. When determining the need for temporary shielding the estimated exposure that would be received by personnel performing the installation and the removal of the shielding will be evaluated against the estimated exposure to be received for the job to be performed without the use of shielding. Also taken into account in evaluating the need for temporary shielding will be the dose rate, the time required to install and remove the temporary shielding, and weight and space limitations. If it is determined that installation of temporary shielding is desirable, the type and amount of shielding will be determined by health physics personnel.

c. The criteria for establishing the need for local decontamination is similar to that for temporary shielding. The exposure from direct radiation or potentially airborne activity from a non-decontaminated area will be compared to the exposure that would be received during the local decontamination effort. Also taken into account will be the time required to perform the local decontamination.

d. See the response to Question 27 of this Appendix.

41. Section 3.3.5.4 stated that "low background radiation waiting areas will be established." Provide the criteria to be used in establishing these areas. Describe how the airborne radioactive contaminants will be controlled in these areas.

RESPONSE

Low background radiation waiting areas will be established using the criteria that the radiation field within the area will be no greater than Zone III ($\leq$15 mr/hr). These waiting areas will be established by the Turkey Point Health Physics staff based on radiation surveys taken periodically by Health Physics personnel throughout the repair effort.

As described in Section 3.3.3, airborne radioactivity inside containment during the repair effort will be controlled, monitored, and released via the plant vent stack by operating the containment purge system. In addition, radioactivity generated during decontamination and cutting of the reactor coolant system will be contained within specially designed contamination control envelopes, which will provide high efficiency filtration. These measures will maintain airborne radioactivity concentrations at acceptably low levels in the waiting areas. 7

42. Section 3.3.7.2 states that detailed radiation surveys have already been made near the steam generators and notes that the reactor coolant system was drained. Was the secondary system also drained during the survey? If not, what is increased dose rate without the secondary side water? In assessing the general area dose rates in the vicinity of the steam generators shown in Figure 3.3-7, what contribution to the level of activity comes from components and structures other than the steam generators? In removing the steam generators, what criteria will be used for shielding craftsmen from other radioactive components and structures? What criteria will be used for shielding craftsmen from other components and structures during preparation for and installation of a new steam generator? Specifically identify by individual tasks (see Section 3.3.7.2, items a through k, for example of tasks), estimated exposure time and radiation field, those operations which contribute more than 10% to your total estimated exposure for the steam generator replacement, including ultimate disposal.

RESPONSE

As stated in Subsection 3.3.7.2, radiation surveys were taken with a teletector in June 1977 with the steam generators drained. Both the primary and secondary sides of the steam generators were drained.

As depicted on Figures 3.3-1 and 3.3-4, the general area dose rates in the vicinity of the reactor coolant loops are relatively constant throughout the 14'-0" elevation indicating essentially equal contributions to the level of activity from the reactor coolant piping, the steam generators, and the reactor coolant pumps. As one gets closer to a a steam generator, the contribution from it will increase until essentially the only contributor to the level of activity is from the steam generator.

The general shielding criteria for the repair effort is given in response to Question 40(b) of this Appendix. In particular, special consideration has been given to shielding the exposed tube bundle and the channel head nozzles. Prior to removing the lower assemblies from the containment, the openings in the lower assemblies will be sealed with shield covers to reduce the radiation exposure to the workers.

A task description, radiation field, estimated manhours, and man-rem dose for each major activity during the repair are contained in Table 3.3-2.

43. Describe how you will use special tools, such as remote cutting and welding apparatus, to help maintain occupational dose As Low As is Reasonably Achievable (ALARA) in your steam generator replacement program.

RESPONSE

Special tools, such as remote cutting and welding apparatus, will be used to the maximum extent practicable to:

a. Reduce the manhours required to perform a specific task, and/or

b. Allow the workers to be further away from the radiation source, and/or

c. Allow the workers to remain behind a shield wall while the task is being performed by an automated device.

The state-of-the-art for remote cutting and welding apparatus is continuously changing throughout the industry. The developments in the field will be followed and techniques will be evaluated using the following considerations:

a. Manhours required to set up the equipment

b. Manhours required to perform the task

c. Experience with the use of the proposed equipment

d. Cost and schedule impact associated with the equipment.

44. Describe the program which will be implemented after the steam generator replacement, if approved, is complete which will assess the actual man-rem exposures for the job relative to the estimates provided in the submittal.

RESPONSE

The existing personnel exposure monitoring program will be utilized to assess the actual man-rem exposures. Each worker on the steam generator repair project will have his exposure record updated on a daily basis. When the total job is complete, the exposure results can be reviewed and it will be possible to determine the actual total man-rem exposures attributed to the repair outage.

45. Section 3.3 frequently references the FPL Health Physics Manual and its implementing procedures. Describe how the FPL Health Physics Manual and procedures take into account ALARA principles contained in Reg. Guide 8.8 and 8.10.

RESPONSE

The Health Physics Manual is built around the philosophy of ALARA and contains sections that fully address the guidance given in Regulatory Guides 8.8 and 8.10. The manual is included in the plant's procedures, which have been reviewed by the NRC.

46. Section 3.4.2 states that a temporary onsite storage area will be provided for the storage of the steam generator lower assemblies. Details of the storage area, including its location, design bases, and radioactivity control systems, should be presented. How long will the steam generators remain at the temporary storage facility? Evaluate the effect of decay time (e.g., 0, 1, 5, 10, 25 years) of the radioactivity in the steam generators relative to the selection of the time of ultimate disposal of the steam generators and man-rem exposures. Clearly indicate how and when the steam generators will be disposed of.

RESPONSE

The removed Unit 4 lower assemblies will be transported south on the containment access road, around the southern end of the Radwaste Building, then east to the storage facility. The removed Unit 3 lower assemblies will be transported northeast on the containment access road, south along the east plant road to the storage facility. In each case, the lower assemblies will be lowered into the facility by cranes with subsequent installation of the roof. A center wall in the facility allows for storage of Unit 4 assemblies on one side of the facility and storage of the Unit 3 assemblies on the opposite side of the facility.

The facility will be founded on engineered fill at finished grade elevation +17'-6" MLW in the existing laydown area approximately 150 feet south of the ash disposal pits and 290 feet east of the Radwaste Building. The elevation of the existing laydown area ranges from +6 to +9 feet MLW. At the storage facility location, the existing surface layer consists of 4 feet of limerock fill, underlain by about 6 feet of muck. Beneath the muck, Miami limestone extends 20 feet, underlain by Key Largo limestone to about elevation -100 feet. 7

Prior to construction of the facility, the existing limerock fill and muck will be removed from below the potential zone of influence of the building foundation, and replaced with compacted crushed limerock fill up to elevation +17'-6" MLW. The existing muck and fill will be excavated to a minimum distance of 15 feet beyond the edge of the building. The building will be at least 65 feet back from the top of the compacted crushed limerock fill boundary slope. This slope will be 1-vertical on 3-horizontal.

The crushed limerock fill which will support the facility will be quarried from local Miami and Key Largo limestone formations. Maximum size will be about 6 inches, with up to 20 percent passing the No. 200 sieve. The crushed limerock will be stockpiled on site to drain effectively to the optimum moisture range between 7 percent and 14 percent. The fill will be placed using a maximum loose thickness of 12 inches. The fill will then be compacted with a vibrating drum roller to obtain a minimum dry density of 110 lb. per cubic foot. An extensive series of laboratory tests on the crushed limestone compacted to 110 lb. per cubic foot dry density has indicated effective strength parameters of conservatively 3 kips per square foot for cohesion and 39 degrees for internal angle of friction.

The compacted crushed limerock fill provides an allowable bearing capacity over 5 kips per square foot, including a factor of safety of 3.

The Miami and Key Largo limestone formations underlying the compacted crushed limerock fill have allowable bearing capacities of over 30 kips per square foot (including a factor of safety of 3).

The fill area will be designed for a 100 year flood level of +12.7 feet MLW as the design storm surge required for buildings in southern Florida. This 100 year flood level is per the Code of Metropolitan Dade County Florida.

The steam generator storage facility will be designed in accordance with the following current codes and standards:

South Florida Building Code

Code of Metropolitan Dade County Florida

Building Code Requirements for Reinforced Concrete (ACI 318)

American Institute of Steel Construction Manual of Steel Construction and Specification for the Design, Fabrication and Erection of Structural Steel for Buildings 7

American Welding Society Structural Welding Code (AWS D1.1)

The facility will be designed for a hurricane wind velocity of 120 miles per hour with application of shape factors in accordance with the American Society of Civil Engineers Paper No. 3269.

The structure will consist of 2'-0: thick reinforced exterior concrete shield walls sized to maintain a direct gamma dose rate of less than or equal to 2.5 mr/hr at exterior wall surfaces. The facility will be approximately 130 feet by 42 feet with centerline along its length oriented in the East to West direction. A 2'-0" thick reinforced interior concrete wall will be provided through the center of the structure for the full length of the facility. The interior wall will provide roof support and separation between the Unit 3 and Unit 4 steam generator lower assemblies. All walls will be founded on continuous strip footings with bases at approximately elevation +15"-6" MLW, 2 feet below the finished grade elevation. A maze shielded entrance with a door through the exterior wall will be provided to allow for periodic surveillance of lower assembly seal integrity. Each lower assembly weighing approximately 186 tons (Specific Gravity $\approx$ 1.88) with two steel support saddles will be stored in the facility on reinforced concrete bearing pads. A 6 inch thick reinforced concrete floor will be provided. Top of floor and pad elevation will be +18'-0" MLW.

The facility roof will be designed to be watertight. The major roof components will consist of precast concrete roof panels with concrete topping for a thickness of approximately 1 foot. The elevation of top of roof will be approximately +39'-6" MLW.

Wall footings and bearing pads will be designed to maintain a maximum allowable soil bearing pressure of 5000 pounds per square foot.

Other design loads for the structure will be in accordance with the South Florida Building Code. Concrete will have a minimum compressive strength of 3000 psi at 28 days. Reinforcing steel and structural steel will be in accordance with ASTM A615 (Grade 60) and ASTM A36 respectively. Concrete and steel allowable stresses will be in accordance with Building Code Requirements for Reinforced Concrete (ACI 318) and Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (AISC).

Since the steam generators will be welded in addition to being in a facility having a watertight concrete roof and reinforced concrete walls, there are no potential means to transport the surface contamination from the lower assembly surfaces. Therefore a floor, sumps and/or air filtration units 7
are not required. However, as previously stated, a floor will be provided.

As described in Section 8.3, an evaluation was performed to determine the most man-rem effective and cost-beneficial method for disposition of the removed lower assemblies. It was concluded in Subsection 8.3.2.7 that the lowest cost and the lowest man-rem burdens are associated with (1) long-term, onsite storage and disposal during decommissioning and (2) immediate barge 7
shipment to a licensed land burial facility.

Figure A 46-1 is a typical decay curve of percent steam generator gamma activity versus time following reactor shutdown. The initial activities used to generate the decay curve are given in Table 5.2-1. With this decay curve, the effect of lower assembly radioactive decay can be directly related to the time of ultimate steam generator disposal and to the associated man-rem exposures.

47. In addition to the steam generator lower assemblies, is there any other contaminated piping which is to be disposed of? If so, identify each major item and its associated radiation level and content and its final disposition. Will any portion of the steam generators or other contaminated piping be decontaminated and released for unrestricted use? If so, identify the item(s) the procedures used for decontamination and the criteria used for final disposition.

RESPONSE

The present plan for accomplishing the repair will not require the removal or disposal of contaminated piping.

It is not planned to decontaminate the lower assemblies and release them for unrestricted use.

48. Provide the curie quantities of transuranic and long lived fission product (e.g., I-129) contamination in the steam generators. Describe the criteria, methods of measurement and disposal of materials which have transuranic contamination.

RESPONSE

Two experimental measurement programs have been conducted on Turkey Point steam generators, i.e., a. gamma spectral measurements of steam generator manway diaphragm samples and b. steam generator dose rate distribution measurements.

a. The gamma spectral measurements were conducted on Turkey Point Unit 3 steam generator manway diaphragm samples using standard Ge(Li) detector techniques 90 hours following unit shutdown. The data received from these measurements was reduced and extrapolated out to 7 years and is provided in Table 5.2-1.

b. The dose rate distribution measurements were conducted in a Unit 4 steam generator channel head using shielded thermoluminescent dosimeters (TLD's). The directionally shielded TLD measurements provided sufficient data so that fractional isotopic distributions could be estimated in the steam generator, e.g., isotopic fraction in the channel head, tube sheet, generator tubes, etc.

The only detectable gamma emitting transuranic isotope found in the manway diaphragm samples was Np-239 (a relatively short lived isotope with a half life of only 2.35 days). Other potential transuranic isotopes were not readily identifiable during these measurements because of their low concentrations and/or weak radioactive emissions.

Analyses have since been performed in order to estimate the transuranic isotopic inventory in a typical Turkey Point Unit 3 steam generator lower assembly. The analyses accounted for the two primary sources of transuranics, i.e., activation of tramp uranium and fuel clad leakage. The tramp uranium concentrations used were typical of those found in Turkey Point type fuel and the fuel leakage source term was approximated based on Unit 3 operating history. Based on these calculations the expected activity of the total transuranic isotopic inventory in a typical Unit 3 lower assembly 3 months following shutdown is estimated to be only 0.28 curies.

Using the steam generator fractional isotopic distribution results obtained from the shielded TLD measurement study (discussed above), the following transuranic distributions were calculated:

TOTAL TRANSURANIC ACTIVITIES (nanocuries/gram)

Component	Time After Shutdown		
	90 Days	1 Yr.	5 Yrs.
Tubes (Excluding Tube End)	5.1	4.6	3.7
Divider Plates	4.1	3.6	2.9
Channel Head	0.39	0.35	0.28
Tube Sheet (Including Tube Ends)	1.0	0.91	0.73
Lumped Average of Above Components	2.7	2.4	1.9
Total Average for Steam Generator Lower Assembly (Total Weight 402,000 lbs.)	1.5	1.4	1.1

These estimates of the expected transuranic activities per gram mass of lower assembly component are well within the private licensed burial site transuranic criteria of 10 nanocuries per gram of waste.

Lower assembly long lived fission products typically associated with shielding, worker dose and solid waste disposal are provided in Table 5.2-1. Measurement and identification of other long lived fission products have not been conducted on the Turkey Point steam generators. It is believed that further measurements of other long lived fission products would not be productive because of their low concentrations and/or weak radioactive emissions. As an example, using the analytical techniques discussed above, it is estimated that only 0.002 curies of I-129 would be present in a typical steam generator. Hence, further analytical or measurement programs on long lived isotopes of this type are not required.

The discussion above concludes that the transuranic concentrations in the removed lower assemblies would be small. This is also evidenced by the findings in EPRI NP-613, Interim Report, January 1978, "Study of Transuranic Concentration Levels in Solid Radioactive Waste from Commercial Power Reactors." Furthermore, as discussed in Section 3.3.3 and in the response to Question 32 of this Appendix, precautions will be taken to prevent the spread of waste materials resulting from pipe cutting and prepping operations. Also, the expected transuranic activities per gram mass of this waste material are well within the private licensed burial site transuranic criteria of 10 nanocuries per gram of waste.

Therefore, it is concluded that existing plant procedures for disposal of solid radioactive waste material are appropriate and will be utilized during the repair.

49. For the entire steam generator replacement program, estimate the total volume and curies of solid wastes which are to be disposed of including transuranics, Iodine 129, Iron 55 and Nickel 63.

RESPONSE

The major source of solid radioactive waste associated with the repair effort is the steam generator lower assemblies. Each lower assembly has an approximate volume of 3500 ft^3 and an approximate radioactivity content of 250 curies of deposited gamma activity as stated in Section 3.4.

Should decontamination of the channel head as discussed in Subsection 3.3.5.3.2 be utilized, up to 45 curies/SG of deposited gamma activity (volume 420 ft^3) would require disposal as solid radioactive material. 7

As discussed in Subsection 3.3.6.1, approximately 60 cubic yards of concrete per unit will be disposed of during the repair effort. It is expected that the radioactivity content in this concrete will be only a fraction of a curie.

Reactor coolant system waste generated during the curring and weld end prep operations will be disposed of as radioactive material. It is estimated that these shavings may amount to approximately 7 cubic feet and contain approximately 1 curie of radioactivity per unit. 7

There will also be some additional low-level radioactive materials. The quantities involved are expected to be the same order of magnitude as the quantities disposed of following a refueling outage. Typical quantities and types of radioactive wastes for a refueling outage are as follows:

	Waste Material	Quantity (ft^3)	Activity (Ci)
a.	Rags, paper and clothing	22,000	~50
b.	Sand, concrete, tools, and scaffolding (packaged as is)	3,000	~50
c.	Evaporator bottoms	700	~ 5*
d.	Spent resin	100	~25*

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*These values were obtained by utilizing typical radioactive concentrations per unit of waste volume from Table 2-4, A.E. Ackens, Jr., et al., "Migration of Radionuclides at Low-Level Waste Burial Grounds," Dames & Moore Report for The AIF, December 1977 (in press).

50. Provide a man-rem analysis for each alternative approach to the steam generator problem (i.e., continue to operate with the present steam generators or retube the steam generators in place). For each alternative, list the major tasks involved, an estimated total occupational radiation exposure associated with each alternative approach.

RESPONSE

This question was addressed in Revision 2 of Sections 3.3.7 and 7.4.

1. During steam generator replacement it is anticipated that particulate effluents will be significant. Therefore, describe the provisions (and acceptance criteria) to pre-test the efficiency of all HEPA filter systems which will be used to reduce particulate radioactive effluents during the steam generator replacement program with DOP.

RESPONSE

It is not anticipated that particulate effluents will be significant during steam generator repair activities because of the measures to be taken during the repair, as discussed in SGRR Section 3.3.3.

During steam generator repair local high efficiency filtration will be provided for the specially designed contamination control envelopes. The efficiency of these HEPA filters will be pre-tested in the supplier's shop with dioctyl-phthalate (DOP) in accordance with the Institute of Environmental Sciences (IES) Standard for HEPA filters, CS-1, paragraph 7.b, or other equivalent acceptable industry standard. Acceptance criteria will be in accordance with IES CS-1, paragraph 4.a, as required by ANSI N509-1976, paragraph 4.1, or equivalent.

2. The objectives of Regulatory Guide 8.8, Rev. 2, for assuring that occupational exposures at nuclear power plants will be kept as low as is reasonably achievable (ALARA) are applicable to steam generator replacement programs. Consequently, the regulatory positions in Paragraph C of Regulatory Guide 8.8, Rev. 2, should be in effect in all areas pertinent to the steam generator replacement program. We consider the following sections of Paragraph C to be pertinent for steam generator replacement activities:

C.1. Program for Maintaining Station Personnel Radiation doses ALARA.

All sections of a thru d

C.2. Facility and Equipment Design Features sections

a all subsections

b (1), (2), (5b), (7)

c (2)

d (1), (3), (4)

i (10), (12)

C.3. Radiation Protection Program

All sections a thru c

C.4. Radiation Protection Facilities, Instrumentation and Equipment

(4) b thru e

You should commit to implementing the above and provide a brief summary of how you plan to implement each of the above sections or discuss the reasons that the positions are not applicable to the repair program.

RESPONSE

a. The Florida Power & Light Health Physics Manual was written to provide a program that would assure that occupational exposures at our nuclear power plants are kept as low as is reasonably achievable (ALARA). In

the preparation of the FPL Health Physics Manual, Regulatory Guide 8.8, Rev. 1 was one of the documents used as a guideline and basis to assure that exposures are ALARA. Regulatory Guide 8.8, Rev. 1 states that:

"Effective control of radiation exposure involves the following major considerations:

1. Management commitment and support;
2. Careful design of facilities and equipment; and
3. Good radiation protection practices, including good planning and the proper use of appropriate equipment by qualified, well-trained personnel."

Each one of these considerations is addressed specifically and in detail in the FPL Health Physics Manual.

b. The basic difference between Rev. 2 and Rev. 1 of Regulatory Guide 8.8 is that Rev. 2 provides more detailed specifications for implementing the major considerations of Rev. 1. The FPL Health Physics Program is defined by the Health Physics Manual and implementing procedures and complies with the intent of Regulatory Guide 8.8, Rev. 2 as discussed below.

The Health Physics Manual and implementing procedures reflect the management policy that the exposure of station personnel to radiation will be ALARA. The Health Physics Supervisor at each plant has a direct line of communications to the Plant Manager and the Corporate Health Physicist. The Corporate Health Physicist reports to the Power Resources Supervisor - Nuclear, who is in charge of the Nuclear Support Staff. The Power Resources Supervisor - Nuclear reports to the Manager of Power Resources - Nuclear who is responsible for all matters concerning the operation and maintenance of nuclear power plants. The Manager of Power Resources - Nuclear reports to the Vice President, Power Resources, who is responsible for all power plant operation and maintenance.

The FPL radiation protection policy and program is established by the Corporate Health Physicist and approved by the Manager of Power Resources - Nuclear in conformance with applicable regulations. The detailed procedures for implementing the program at each plant are established by the Health Physics Supervisor. The Health Physics Supervisor at each plant is responsible for ensuring the exposures are ALARA. The Corporate Health Physicist is responsible for ensuring that the ALARA policy is implemented.

Most personnel whose duties require working with radioactive materials are trained in the fundamentals of radiation protection and in station exposure control. Those who are not trained are escorted by trained individuals while in the Radiation Controlled Area.

New planned or modified designs and the selection of new equipment is reviewed by both plant and corporate health physics personnel, to ensure proper designation of radiation zones, adequate shielding and proper equipment and support facilities to ensure ALARA. Instrumentation, monitors and controls requiring frequent maintenance or access are located remotely in low radiation zones whenever feasible. When design changes are not feasible for control of airborne contamination, auxiliary ventilation systems are used.

The Health Physics Manual and implementing procedures provide for planning, operation, and evaluation of operations involving radioactive materials including training, monitoring of areas and operations, and monitoring of personnel exposure. Multi-range and multipurpose portable instruments are used to provide adequate surveys and monitoring. Hand and foot monitors, portal monitors, TLD's, direct-reading pocket dosimeters, "friskers" and whole body counters are used to adequately monitor personnel.

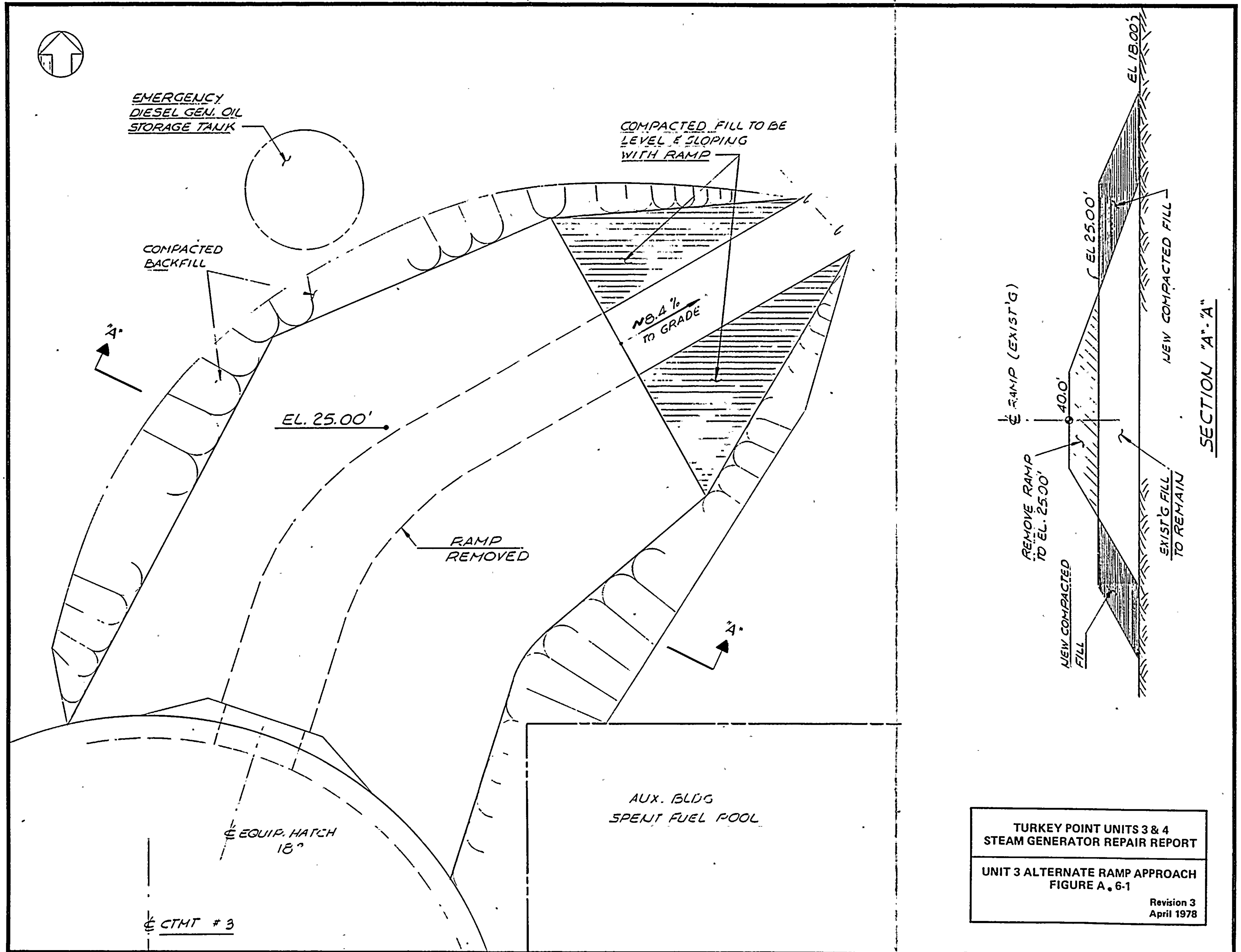
Protective clothing and respiratory protection equipment are used extensively where needed to protect personnel from exposure to radioactive materials. Decontamination areas, repair facilities, change rooms, and instrument calibration provide the support for a well-rounded radiation protection program that fully complies with the intent of Regulatory Guide 8.8, Rev. 2.

c. During the steam generator repair every reasonable effort will be made to keep radiation exposures to plant personnel ALARA in accordance with 10CFR20.1(c). The methods of achieving this objective are discussed in Sections 1.1.4, 2.2, 3.3, 8.1, 8.2 and 8.3, and in the responses to Questions 40 through 46, Appendix A. These methods are summarized below.

The radiological protection program to be implemented during the repair effort will be in accordance with the FPL Health Physics Manual which is responsive to Regulatory Guides 1.8, 8.2, 8.8 and 8.10. As appropriate, craft personnel will be given a comprehensive course in radiological protection. Radiation surveys will be taken throughout the repair and appropriate access controls will be established. Low background radiation waiting areas will be established where workers must wait between tasks. Additional facilities will be provided including craft change areas, radiation control points, protective clothing change areas and supplemental Health Physics areas. Additional facilities for laundering of protective clothing and cleaning of respiratory equipment will also be provided. Bulk containment atmosphere control of airborne radioactivity will be achieved by means of the containment purge exhaust system; local control of airborne radioactivity will be achieved by enclosure envelopes and high efficiency filters during cutting of the reactor coolant pipes.

All personnel entering the radiation controlled areas will be provided with TLD's, and all personnel entering a high airborne radioactivity area will be given an initial whole body count or bioassay at the start of employment and subsequently, as necessary. Portable survey and monitoring equipment will be available throughout the repair effort for area and airborne surveys and for personnel "frisking". Workers will be provided with one or two sets of protective clothing and respirators, as required. Treatment of contaminated personnel will follow established FPL Health Physics procedures.

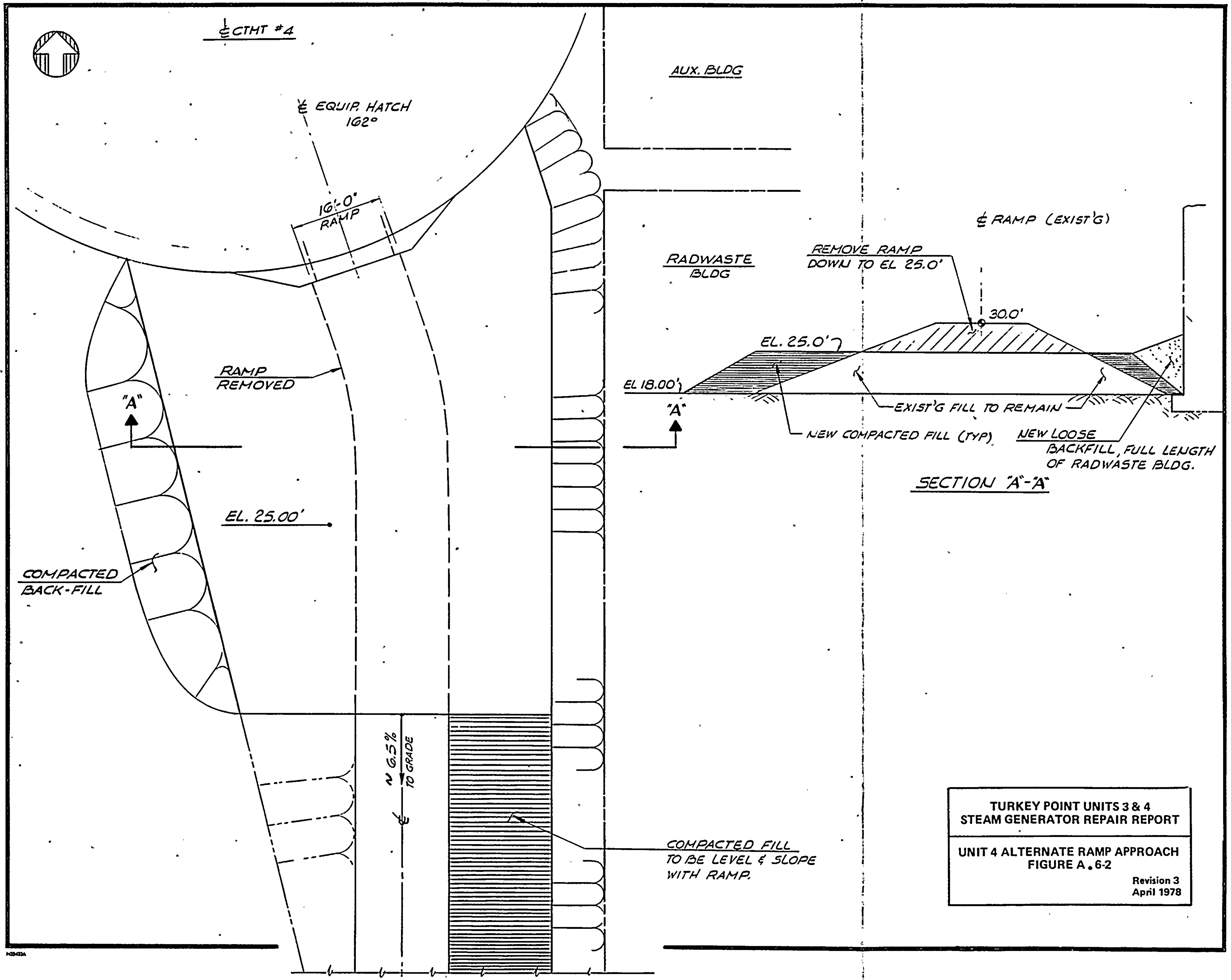
Additional ALARA consideration will be given to the planning for and the implementation of the repair. The replacement steam generator lower assemblies will incorporate design changes expected to enhance tube integrity; therefore, radiation exposure presently attributed to steam generator maintenance would be reduced over the remaining operating life of the plant. Special tools, such as remote cutting and welding apparatus, will be used to the maximum extent practicable. Temporary scaffolding, platforms and lighting will be employed to provide ready access to work areas. The containment model, the laydown space model and the steam generator model will be utilized extensively for work planning and craft personnel training; additionally, the models will be utilized to study temporary shielding requirements. Health Physics personnel will work with job supervisors to assure that activities are performed in accordance with ALARA practices. The method of storage and disposition of the removed steam generator lower assemblies will be selected to minimize exposure to personnel.



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

UNIT 3 ALTERNATE RAMP APPROACH
FIGURE A.6-1

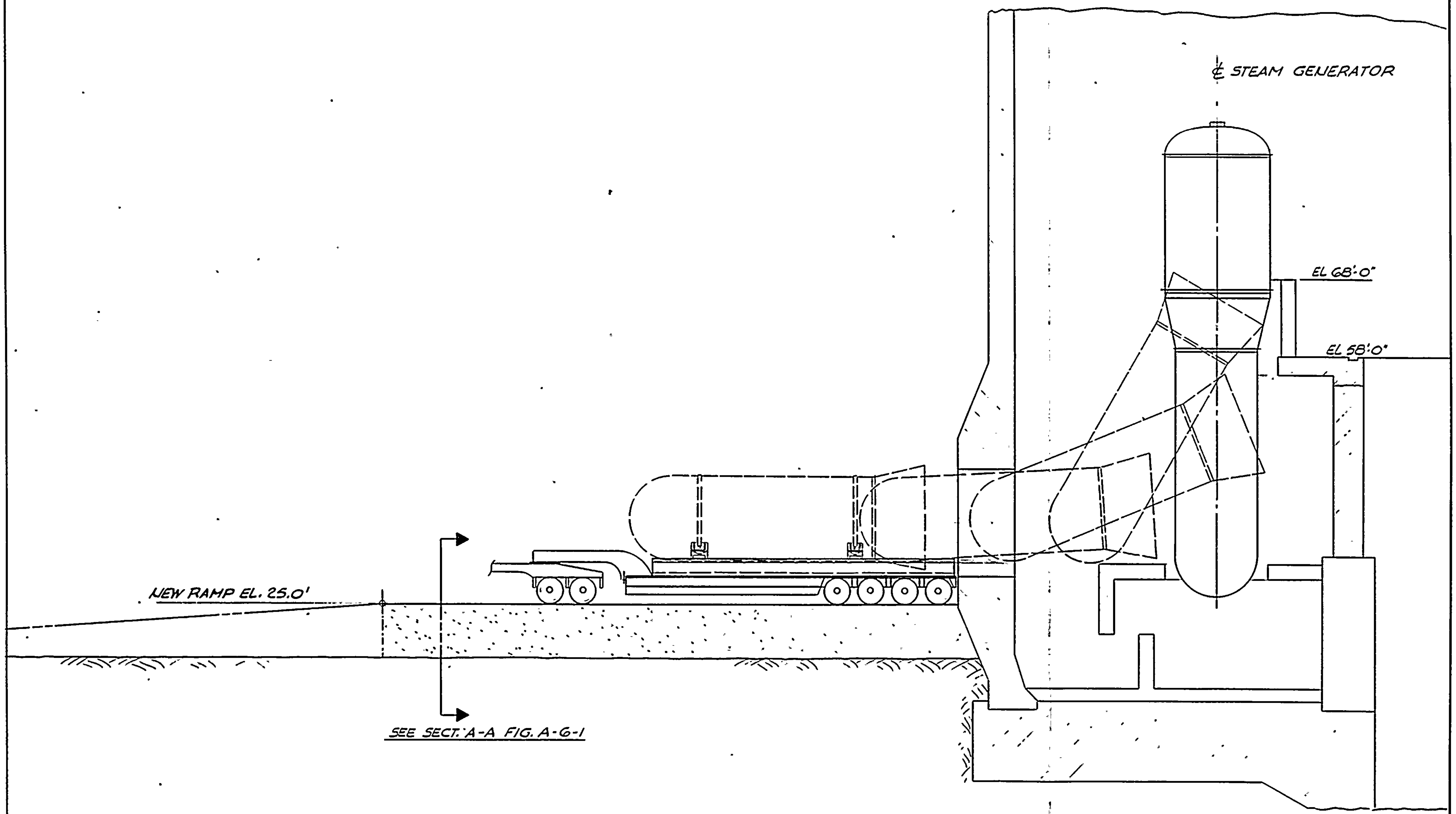
Revision 3
April 1978



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

UNIT 4 ALTERNATE RAMP APPROACH
FIGURE A.6-2

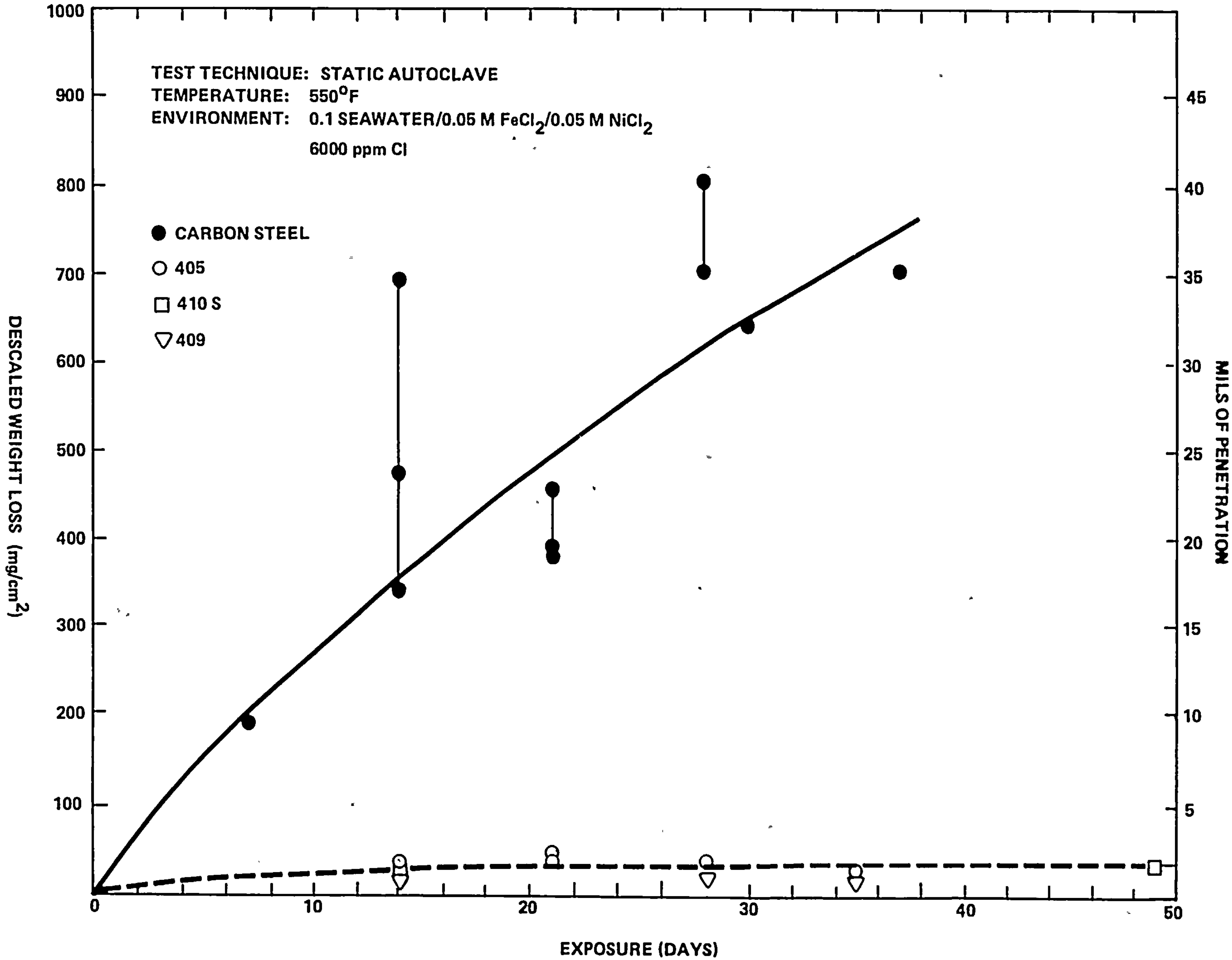
Revision 3
April 1978



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

ALTERNATE RAMP APPROACH
SEQUENCE
FIGURE A.6-3

Revision 3
April 1978

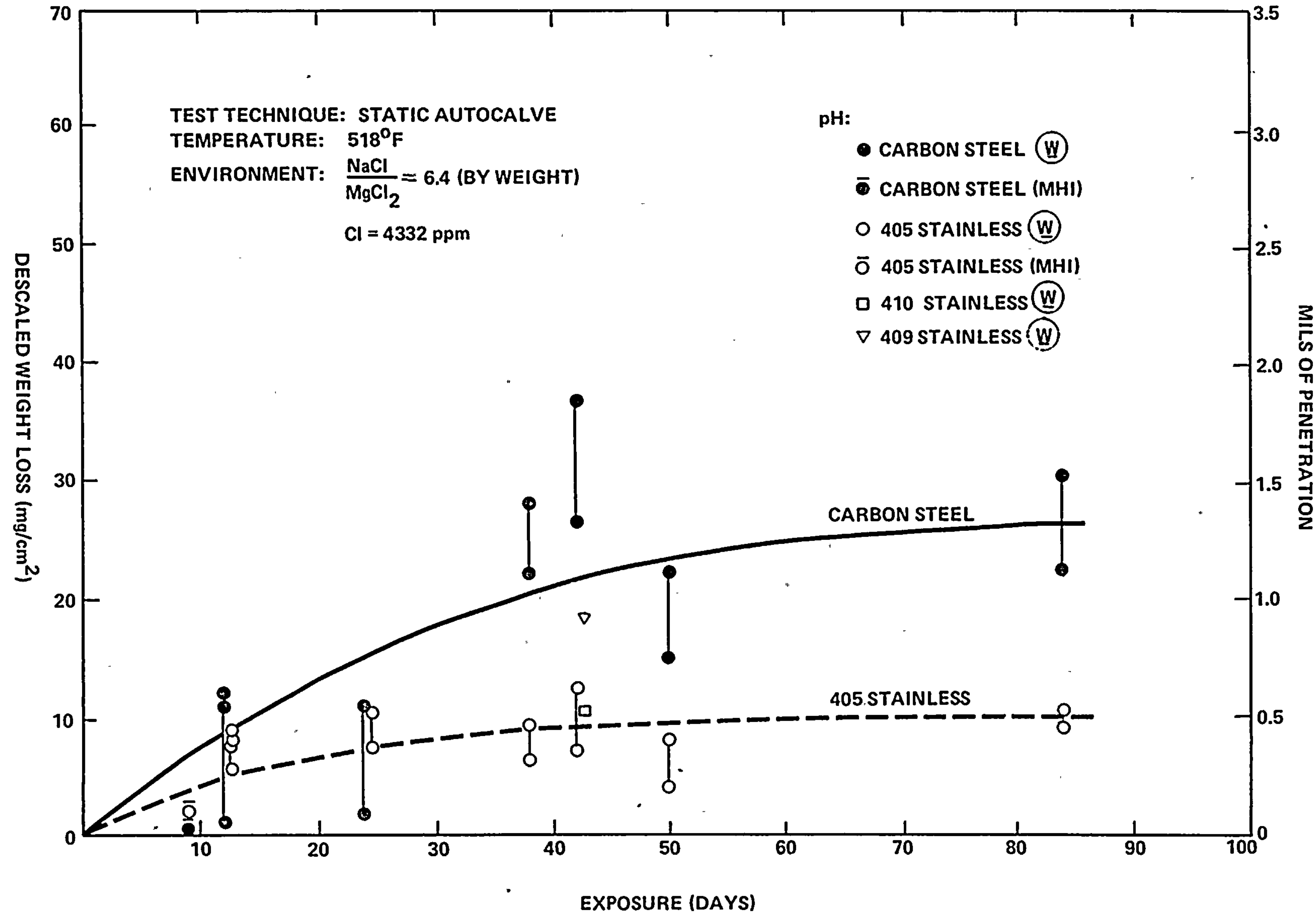


TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

DESCALED WEIGHT LOSS VS EXPOSURE IN SEA WATER/ $Fe\,Cl_2$/ $Ni\,Cl_2$

FIGURE A.22-1

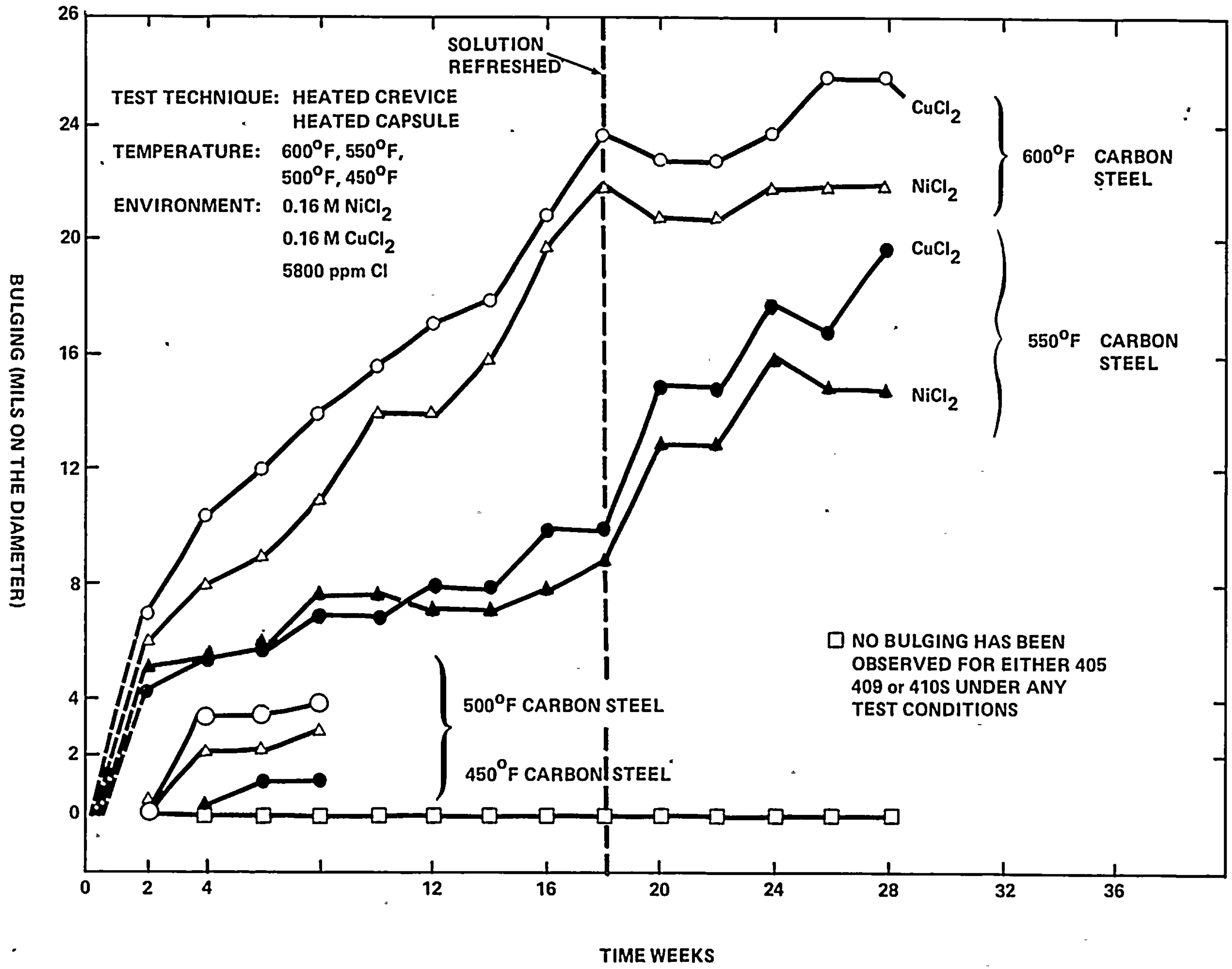
Revision 3
April 1978



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

DESCALED WEIGHT LOSS VS EXPOSURE
IN Na Cl/Mg Cl_2
FIGURE A. 22-2

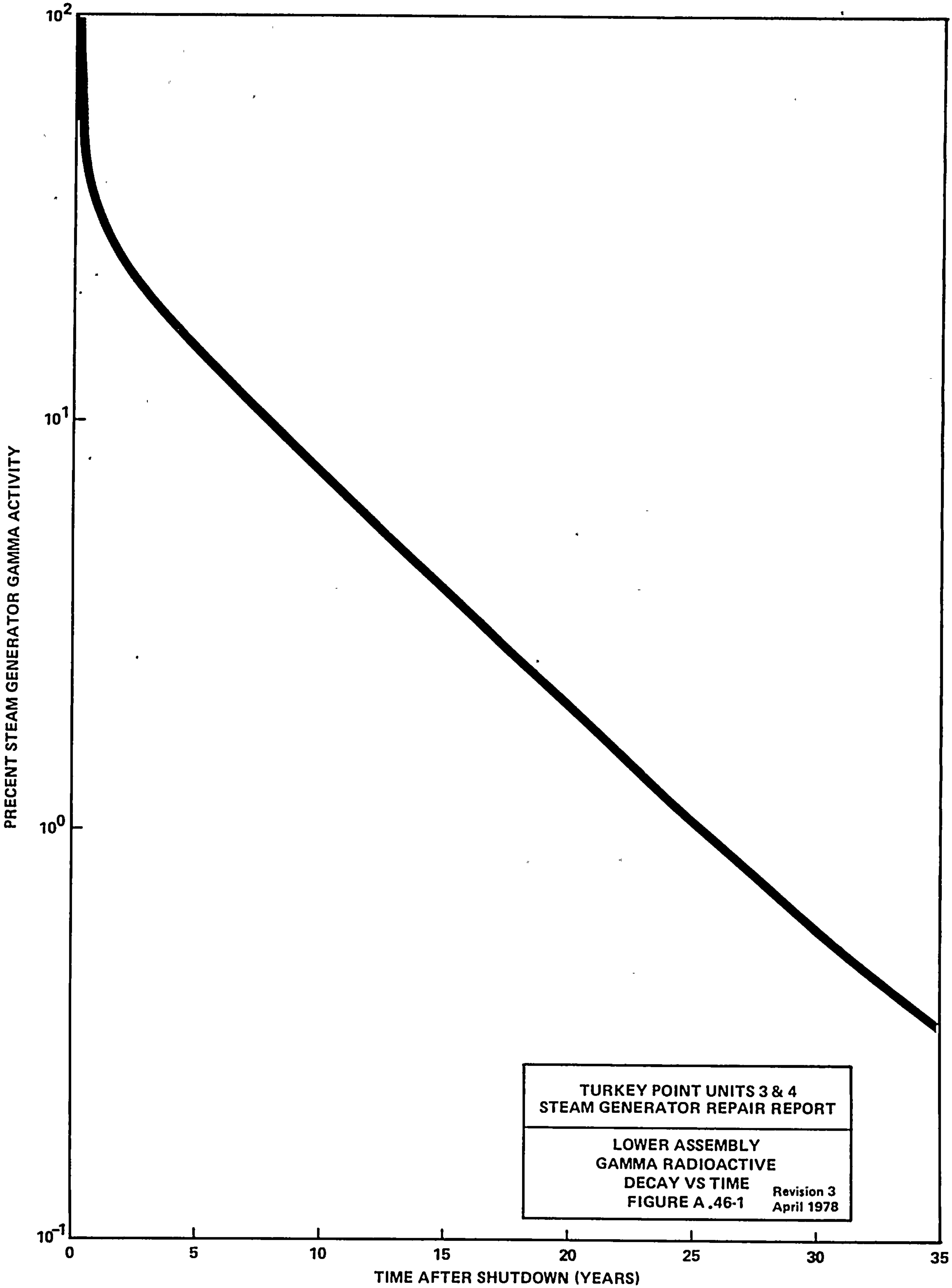
Revision 5
August 1978



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

BULGING VS EXPOSURE
IN $NiCl_2$ and $CuCl_2$
FIGURE A. 22-3

Revision 3
April 1978



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

LOWER ASSEMBLY
GAMMA RADIOACTIVE
DECAY VS TIME
FIGURE A.46-1

Revision 3
April 1978

1. SPENT FUEL POOL COOLING

(a) In the event of the postulated damage to the buried intake cooling water piping of the unit under repair (section 5.2.1.4) can the remaining CCW system provide cooling to both pools simultaneously? (b) Do you anticipate that a refueling of the operating unit may occur during the steam generator replacement of the other unit? (c) If so, what would be the maximum cooling water outlet temperature with only one CCW system operating? (d) If not, what is the expected maximum heat load on the one CCW system compared to its capacity and what will be the maximum cooling water outlet temperature under these conditions?

(e) What will be the delay time for getting the full core into the spent fuel pool from the unit undergoing steam generator replacement?

(f) Provide your response to these questions together with any other principal details that would clarify the coolability of the spent fuel in both units for the worst potential conditions under consideration.

(g) Provide the details of the analysis of a free falling crane boom on the auxiliary building roof over the spent fuel pool (section 5.2.1.2).

RESPONSE

a. As discussed in Subsection 5.2.1.4, damage to the intake cooling water (ICW) piping on the unit under repair will not impact the operating unit. The component cooling water (CCW) system of the operating unit has sufficient capacity to supply operational cooling needs (including its spent fuel pool) and to supply, through the Unit 3 and Unit 4 CCW intertie, the cooling needs of the spent fuel pool of the unit under repair.

b. Based upon the present schedule, a refueling of Unit 3 can be expected to occur during the Unit 4 repair outage. As delineated in item c. below, one ICW/CCW train of either unit has the capability of removing the maximum expected heat load of both units (including their spent fuel pools) 10 days after reactor shutdown. The same consideration applies should Unit 4 be refueled during the Unit 3 repair outage.

c. The heat load to be removed by one ICW/CCW train for the case of one unit under repair and one unit in refueling is similar to the case of one unit under repair and one unit under normal operation discussed in response to Question 5, Appendix A. The heat loads assumed in this case are as follows:

Heat Source	Heat Load (Btu/hr)	Flow to Remove Heat Load gpm
	Unit Under Repair	
Spent fuel pool conservatively assuming 4 - 2/3 cores are stored and accident occurs 10 days after shutdown for full core off-load.	$18.6X10^6$	2800
	Unit Under Refueling	
Spent fuel pool conservatively assuming 3 - 2/3 cores are stored with freshest batch decayed 1 yr.	$1.9X10^6$	230
Reactor vessel 10 days after shutdown for normal refueling	$16.7X10^6$	1995
Containment air coolers	$1.6X10^6$	560
Miscellaneous	$.1X10^6$	415
Total	$38.9X10^6$	6000

Assuming the most limiting ICW system piping failure, the single ICW/CCW system in the unaffected unit has the capability of removing $39X10^6$ Btu/hr with an ICW temperature of 95 F and a maximum CCW outlet temperature of 117 F. Therefore, a single ICW/CCW train is sufficient for the conservative scenario discussed above.

Although there are no plans to schedule the repair and refueling outages of the units in such a manner that would result in the conservative heat loads assumed in this analysis (i.e., the incident occurring 10 days after both units shutdown), administrative procedures will prohibit the movement of steam generator lower assemblies over the ICW piping of the unit undergoing refueling for the first 10 days following shutdown if the reactor vessel head is removed.

d. Not applicable, based on the response to Part b.

e. In the response to Question 5, Appendix A, a delay time of 10 days was assumed to conservatively estimate the heat load on the spent fuel pool. However, it is expected that transfer of fuel from the reactor to the spent fuel pool will be completed approximately three weeks after unit shutdown.

f. The principal details pertaining to the coolability of the spent fuel pools is contained in parts a. through e., above, and in the response to Question 5, Appendix A.

g. Figure C.1-1 shows the boom motion geometry used for analyzing the boom impact on the auxiliary building roof over the spent fuel pool (Subsection 5.2.1.2). When the crane boom impacts the reinforced concrete roof parapet (Point B), a moment resisting plastic hinge is formed at the point of impact. The motion of the boom beyond the parapet is reduced by this moment resistance, thereby not impacting the roof. The parapet will maintain its structural integrity after impact.

The boom is assumed to be initially in a vertical position 81 degrees from the horizontal. This is the maximum angle to which the boom can be physically raised. The distance between the crane and structure may vary, resulting in a range of possible boom inclination angles (α) at impact. This in turn produces a range in the length of boom segments that may be over the roof at the time of impact. For angles between $\alpha = 9^{o}$ and $\alpha = 32.6^{o}$, the corresponding length of the boom segment that may be over the roof varies between approximately 10 feet and 2 feet. Therefore, the boom drop analysis is performed for this range of impact angles and the corresponding boom segment lengths. For α greater than 32.6^{o}, the boom tip will strike the wall with no segment of the boom extending beyond the plane of the auxiliary building wall.

For purposes of this analysis, a typical 70-foot boom section is assumed to impact the parapet. The boom cross section is comprised of four pipes in the shape of a box (two in each chord) and tied together by lattice bracing. The pipe chord is 5-1/2 inches in diameter with a wall thickness of 1/2 inch. The pipe material is T1 steel with a minimum yield strength of 110 ksi and minimum elongation of 18 percent.

The plastic capacity of the boom was determined as follows:

$M_p = 0.9F_y Z$ = Plastic moment capacity

$V_p = 0.5F_y A$ = Plastic shear capacity

Where: F_y = yield strength of pipe material

$$Z = 1/3\ d^3 \left[1-\left(1-\frac{2t}{d}\right)^3\right]$$ = plastic section modulus for two chord pipes where "d" is pipe diameter and "t" is pipe thickness.

A = cross sectional area of two pipes

The equation of motion of the free falling boom was derived by considering the boom as a slender rod rotating about a frictionless hinge. Upon impact with the auxiliary building parapet, plastic hinges are formed instantaneously in the boom chords at the point of impact. The motion of the outstanding boom segment is given by the following:

$$(\omega')^2 = \omega_\alpha^2 - \frac{3g}{x}\left\{\mathrm{Sin}\left[\left(\frac{\pi}{2}-\alpha\right)-\theta'\right] - \mathrm{Cos}\,\alpha\right\} - \frac{12\,M_p h\,\theta'}{m_T x^3} \quad \text{(Equation 1)}$$

Where: ω' = Angular velocity after impact

ω_α = Angular velocity prior to impact equal to $\frac{3g}{h}(1 - \mathrm{Cos}\,\alpha)$

g = Gravitational acceleration

α = Boom impact angle

h = Total boom length

x = Length of outstanding boom segment over roof

M_p = Plastic moment capacity of boom

m_T = Total boom mass

θ' = Angle of boom segment beyond Point B after impact with roof parapet

For the boom to touch the roof, $\theta' = \pi/2 - \alpha$ and by substitution into Equation 1 the angular velocity is given by the following:

$$(\omega')^2 = \omega_\alpha^2 + \frac{3g\mathrm{Cos}\,\alpha}{x} - \frac{12\,M_p h\left(\frac{\pi}{2}-\alpha\right)}{m_T x^3} \quad \text{(Equation 2)}$$

For all values α and x within the range previously mentioned, the quantity on the right hand side of Equation 2 is negative. This indicates that the boom segment will stop rotating prior to impacting the roof slab. Therefore, the structural integrity of the roof over the fuel pool area will be maintained during the postulated incident.

2. ELECTRICAL

Identify the cable circuit inside the containment connected to the steam generator and any other system associated with the repair of the steam generator, and provide the safety analysis to prove that no adverse interaction occurs to the operating unit.

RESPONSE

As discussed in Section 3.2, detailed engineering studies are in process to precisely define the components, pipes, cables, instruments, etc. within the containment affected by the repair activity. The results of these studies to date are contained in Sections 3.2.1 through 3.2.5 and, for instrumentation and electrical equipment, are summarized below.

a. Unit 3 and Unit 4 mechanical equipment to be temporarily relocated is listed in Section 3.2.1. The disconnection and reconnection of power cables to the equipment is discussed in Section 3.2.3. The associated feeder breakers for the equipment will be opened and tagged.

b. Unit 4 instrumentation to be temporarily removed is listed in Section 3.2.2; Unit 3 instrumentation to be removed is still under study. The removal and reinstallation of the associated cables is discussed in Section 3.2.3.

c. Unit 3 and Unit 4 circuits to be temporarily removed are listed in Table 3.2-1. The removal and reinstallation is discussed in Section 3.2.3. 7

As discussed in the response to Question 3, Appendix A, no postulated rigging incident inside the containment of the shutdown unit could adversely affect the safety-related functions of the operating unit.

3. ELECTRICAL

Describe the electrical load (temporary and permanent) in use for the steam generator repair and the electrical system modification, and identify the corresponding power source.

RESPONSE

The temporary electrical loads presently planned for the repair will include stress relieving equipment, welding equipment, miscellaneous power tools, supplemental HVAC for contamination control envelopes, and supplemental lighting, as required, inside and outside of the containment.

Supplemental HVAC for contamination control envelopes, stress relieving and welding equipment, miscellaneous power tools, and supplemental containment lighting will be fed from a temporary 1500-kVA transformer and a temporary distribution center located inside the containment during the repair. The temporary transformer will be fed by the normal power supply to either reactor coolant pump B or C. Power will be supplied to the temporary transformer via the normal 4.16-kV distribution system, the startup transformer and the 240-kV switchyard. (See FSAR Figures 8.2-2 and 3.) Except for the temporary transformer and load center, the on-site electrical distribution system will be configured the same as during a normal plant refueling shutdown; furthermore, the temporary electrical system modification will not degrade the existing level of safety in the unit under repair or the operating unit. Following the repair the electrical system will be returned to normal and the temporary transformer and load center will be removed from containment.

Supplemental lighting outside of containment will be fed from existing on-site construction power system which, in turn, is fed from the Florida City power line. This power system is independent of the on-site distribution system shown in FSAR Figures 8.2-2 and 3.

4. FIRE PROTECTION

The steam generator repair report does not address fire hazards or fire protection measures to be implemented during the repair program. This program may involve a large quantity of transient combustible materials in the form of clothing and sheeting materials for radiation protection and construction materials used for rigging or concrete forms. Ignition hazards will be present due to cutting, welding, and grinding operations.

As such, the potential for fires may be higher than for normal construction activities. At the present, the only fire suppression capability that exists in the containment building is provided by portable extinguishers. Although the effect of a fire may not lead to a major impact on plant safety with all fuel removed from the reactor, the consequences of fires could result in damage to systems and components important to safety for future plant operations. A fire could have an effect on outside contract personnel from both the products of combustion and the spread of radioactive contamination.

An analysis should be made of the quantities of combustibles involved with the steam generator repair program and the adequacy of fire protection measures provided to minimize the impact of potential fires.

The staff has not received a response from the licensee on the extent to which the fire protection program at Turkey Point meets the NRC guidelines on "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance." The licensee should address any items of non-conformance to these guidelines which will be implemented prior to the steam generator repair program as well as the basis for non-conformance to any items during the repair program.

Based upon the NRC guidance for fire protection, the following recommendations appear to be applicable for consideration for the repair program. The licensee is requested to address each of these recommendations and to indicate those which are or will be met and those which are not applicable along with the basis therefor.

1. All wood used inside the containment should be treated fire-retardant lumber.

2. Provisions should be made for the installation of a fire water standpipe system with hose stations capable of reaching containment areas.

3. Portable fire extinguishers should be provided throughout the containment working areas and in areas where cutting, welding, and grinding operations take place.

4. Battery-operated seal beam emergency lighting should be provided for general areas and access and engress routes.

5. Covered noncombustible containers should be used for disposing of used protective clothing.

6. Plastic sheeting and paper products used to control cleanliness or the spread of contamination should have suitable fire-retardant characteristics.

7. Additional fire protection equipment should be available when heavy loads or equipment are moved along the Unit 3 equipment hatch access way in the area of the diesel fuel oil storage tank.

8. Administrative procedures should be reviewed for the control of combustibles.

9. Pre-fire plans should be established to address all potential fire situations.

10. Periodic inspections should be made to address all fire hazards associated with construction activities.

RESPONSE

FPL compliance with the NRC guidelines on "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance" is described in the FPL letter to NRC dated June 5, 1978.

A fire inside containment can not cause offsite radioactive exposures of consequence because the fuel will be removed from the containment of the unit under repair, nor can it cause a reduction in the safe shutdown capability of the plant. Damage to the reactor coolant system or to other systems and components normally required to mitigate the consequences of an accident would be an economic consideration, not a nuclear safety consideration, during the repair. Additionally, safety considerations for future plant operations are not a concern because any damaged equipment would be inspected, tested and/or replaced, if necessary, prior to returning to power operation.

In the event a postulated fire inside containment, outside contract personnel will be evacuated from containment, thereby mitigating any potential effects from the products of combustion and the spread of radioactive contamination. The likelihood of a fire in the containment during repair is highly remote, since the following precautions will minimize the probability and impact of potential fires.

1. All wood used in safety related areas will be fire retardant to the extent practicable. Fire retardant wood will be used for scaffolding and work platforms inside the containment. Following the repair, a thorough inspection of the containment will be made to ensure that all wood is removed prior to unit startup.*

2. A fire water standpipe system is not planned for inside the containment because adequate fire prevention measures will be taken during the repair. These measures are discussed in the following paragraphs and are summarized below:

a. Contamination control envelopes for use during cutting operations will be constructed of fire retardant materials.

b. Combustible materials will be minimized in areas in which cutting and welding operations are being conducted.

c. Combustible materials inside the contamination control envelopes will be minimized.

d. A fire watch will be present during welding and cutting operations.

e. Adequate fire extinguishers are available throughout the containment and portable fire extinguishers will be readily accessible in the work areas when cutting and welding is performed.*

f. Fire hose of sufficient length to reach the most remote steam generator compartments as well as containment areas will be available and dedicated to fight a postulated fire inside the containment.

3. In addition to the fire extinguishers already in the containment, portable fire extinguishers will be readily accessible in the work area as necessary, in accordance with existing FPL welding and cutting procedures.*

4. During the repair, the existing containment lighting system which includes dc emergency lighting will be operable.

5. Procedures for handling used protective clothing during a normal refueling shutdown will be employed during the steam generator repair. 7

6. Normal refueling shutdown practices for plastic sheeting and paper products will be employed during the steam generator repair. Also, controls will be established for the removal of combustibles as stated in item 8.*

7. A portable foam system, suitable for use on hydrocarbon liquid fires, will be available on-site during the repair.

8. All administrative site procedures will be reviewed for the control of combustibles. Controls will be established for the removal of all combustible waste, debris, scrap, oil spills, or other combustibles resulting from the work activity (unless suitably contained) in the area following completion of the activity or once per 24 hours, whichever is sooner.*

9. Fire fighting strategies to be in effect during the repair outage will be discussed during fire brigade training sessions.*

10. The current practice of performing routine inspections for fire hazards will continue during the repair outage.*

*Additional information complementing these responses has been provided to the NRC in FPL's response to NRC Guidelines on "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance", by FPL letter to NRC dated June 5, 1978.

1. Provide a detailed description of the onsite storage facility for the replaced steam generator lower assemblies. Include a description of all systems designed to minimize the spread of radioactivity (e.g., floors, sumps, air filtration units).

RESPONSE

Preliminary engineering design consists of a 42 ft by 130 ft steam generator storage facility designed as a reinforced concrete structure, with a watertight concrete roof.

Since the steam generators will be sealed in addition to being in a facility having a watertight concrete roof and reinforced concrete walls, there are no potential means to transport the surface contamination from the lower assembly external surfaces. Therefore a floor, sumps and/or air filtration units are not required. However, a 6 inch reinforced concrete floor will be provided.

The steam generator storage facility will be located in the laydown area 7
approximately 150 feet south of the ash disposal pits and 290 feet east of the Radwaste Building, or approximate area, per Figure D.1-1. The facility will be founded on engineered fill at finished grade elevation 17'-6" MLW.

The structure will consist of 2'-0" thick reinforced exterior concrete shield walls. A 2'-0" thick reinforced interior concrete wall will be provided through the center of the structure for the full length of the facility. The interior wall will provide roof support and separation between the Unit 3 and Unit 4 steam generator lower assemblies. All walls will be founded on continuous trip footings with bases at approximately elevation +15'-6" MLW, 2 feet below the finished grade elevation. A maze shielded entrance with a door through the exterior wall will be provided to allow for periodic surveillance of lower assembly seal integrity. Each lower assembly weighing approximately 173 tons with two steel support saddles will be sotred in the facility on reinforced concrete bearing pads. Top of floor and pad elevation will be +18'-0" MLW.

The facility roof will be designed to be watertight. The major roof components will consist of precast reinforced concrete roof panels with concrete topping for a thickness of approximately 1 foot. The elevation of top roof will be approximately +39'-6" MLW. In each case, the lower assemblies will be lowered into the facility by cranes with subsequent installation of the roof.

The design of the wall thickness was determined using a point-kernal computer code which used semi-empirical methods developed by Rockwell[1] for calculating the direct gama dose rates from a homogeneous volumetric cylindrical source through slab shields.

The values of the source terms for the analysis were based on the results of a field survey* of a steam generator in a drained condition one month after shutdown. For conservatism it was assumed that all short-lived isotopes had decayed away and the sole contributor to the measured dose rate was cobalt-60, which has the highest average gamma ray energies and is therefore the most difficult to shield for a given curie level. The results of the conservative analyses indicate that 24-inch concrete walls are required to meet the dose criteria of 2.5 mr/hr at the exterior wall surfaces. However, the dose at the exterior wall surface is expected to be at or below 0.25 mr/hr.

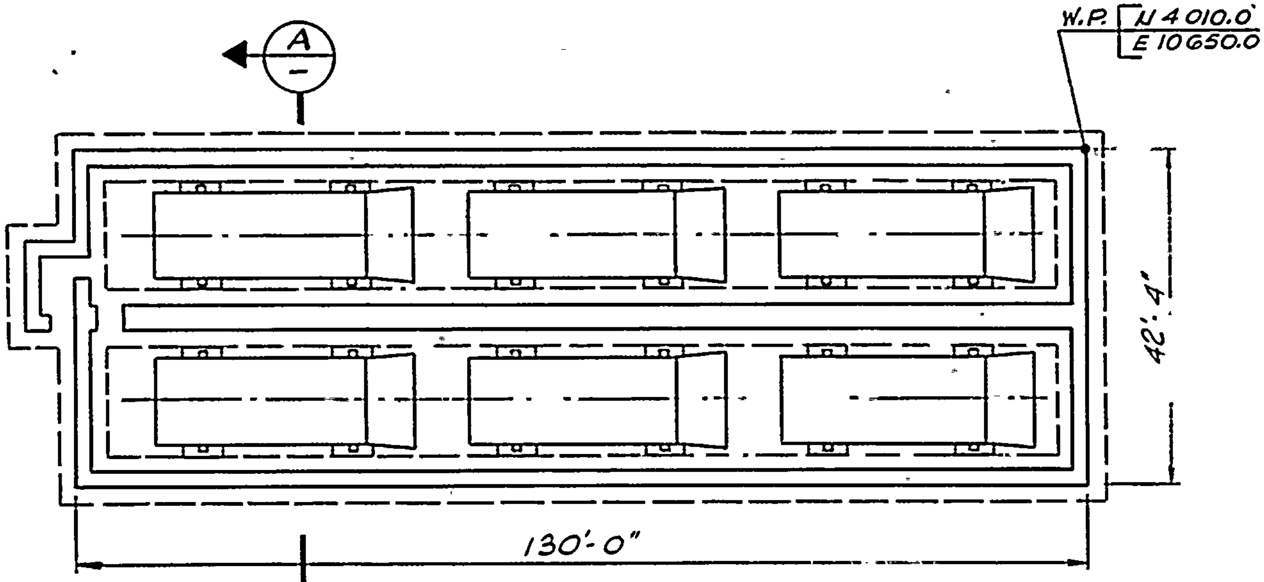
The skyshine analysis was performed with an industry-recorgnized computer code G^3 [2] based upon the same field survey* of a steam generator, assuming the average energy of Co^{60} as that of the source. The source strength of the isotropic point source was determined by calculating a normalization constant equal to the total photon leakage from the steam generator. The skyshine contribution, without taking credit for a shielding roof, will not increase the dose rate outside the compound over 0.25 mr/hr.

The resulting dose equivalent to an individual at the north site boundary location for a full year was calculated assuming 2.5 mr/hr at the outside surfaces of the storage compound, plus the skyshine contribution assuming no roof on the storage facility. The calculated dose was 5.2×10^{-3} mrem which is considered an insignificant contribution of the offsite dose (see SGRR Section 3.4.6). The presently proposed facility location (see Figure D.1-1) was assumed for the aforementioned analysis.

*See SGRR Figure 3.3-7

(1) T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., New York (1956).

(2) R. E. Malefant, G^3: A General Purpose Gamma-Ray Scattering Program, Los Alamos Scientific Laboratory, LA 5176 (June 1973).



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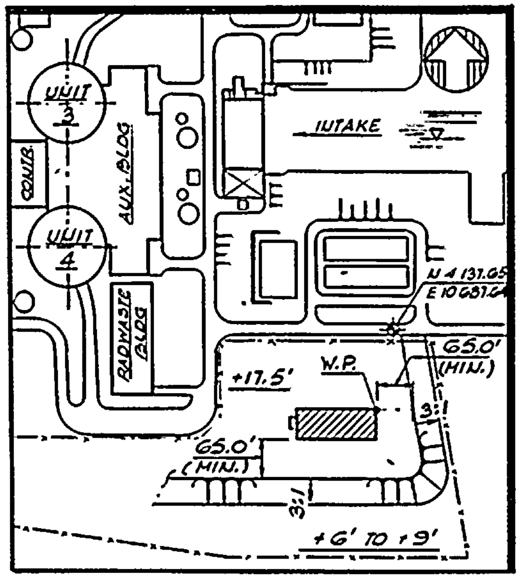
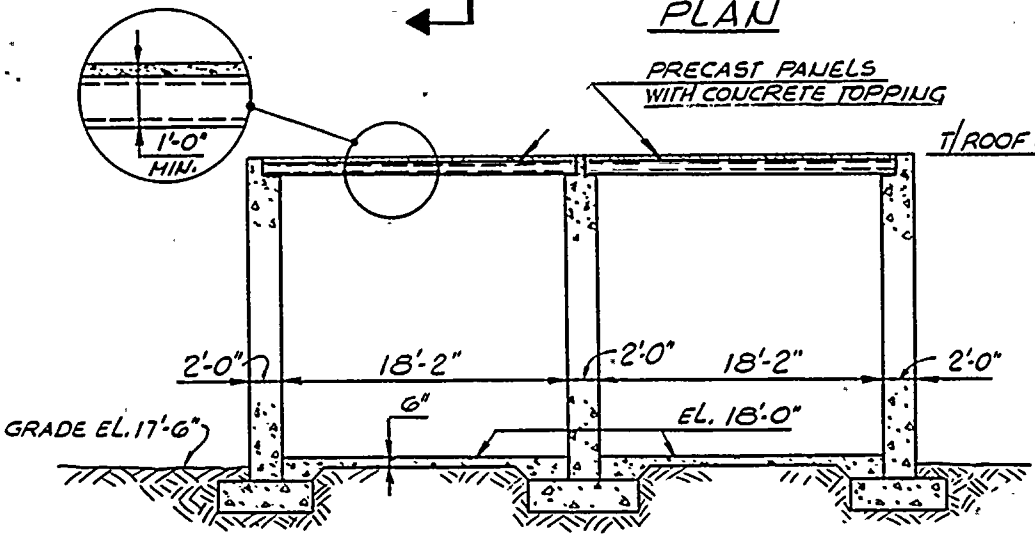


KEY PLAN



SECTION A

NOTE

THESE DIMENSIONS ARE PRELIMINARY AND MAY VARY WITH FINAL DESIGN. FIELD SURVEY INDICATES THAT EXISTING LAYDOWN AREA ELEVATIONS VARY FROM +6.22' TO +9.27' MLW

TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

PROPOSED
STEAM GENERATOR STORAGE COMPOUND
FIGURE D.1-1

Revision 7
March 1980

2. Provide a description of the proposed surveillance requirements for the steam generator lower assemblies. Describe the methods to be used to monitor airborne and liquid releases to the environment from the storage facility. Describe the monitoring program to be used to assure the integrity of the seals on the steam generator openings and that the external surface contamination remains below acceptable levels. Include the proposed surveillance frequency for the above monitoring programs.

RESPONSE

As stated in SGRR Section 3.4.6, the proposed surveillance requirements for the steam generator lower assemblies will consist of a quarterly surveillance program which includes visual inspection of the external surfaces of the lower assemblies, random swipes of the welds sealing the openings in the lower assemblies, and area radiation surveys inside of the steam generator storage facility. This program will provide the necessary assurance that steam generator seal weld integrity is maintained and that surface contamination levels remain below acceptable levels.

The steam generator compound will have reinforced concrete walls and a watertight concrete roof. This coupled with the surveillance program discussed above provides appropriate assurance that there will be no airborne or liquid releases to the environment from the storage facility.

3. Provide a description of the methodology of determining surface contamination levels on the replaced steam generators. Provide the criteria used for determining if the external surfaces of the steam generators must be decontaminated. If the steam generators are to be stored in an open facility (e.g., open roof or earthen floor), it is the staff's position that external loose and fixed surface contamination should be reduced to a minimum. A level acceptable to the staff is the contamination level given in Table 1 of Regulatory Guide 1.86. Provide assurance that (1) the external surfaces of the steam generators will be decontaminated to acceptable levels, (2) the surface activity will be adequately contained or (3) justification for allowing surface contamination levels in excess of Regulatory Guide 1.86 for open storage of the generators.

RESPONSE

Standard swipe techniques will be used according to 49 CFR 173.397 to measure removable contamination on the external surfaces of the steam generators. If the removable contamination level on the swipe is ≤2200 dpm/100 cm^2 (β-γ), decontamination will not be required. If the contamination levels are >2200 dpm/100 cm^2 (β-γ) on the swipe, the external surfaces will be decontaminated until the measured activity on the swipe is ≤2200 dpm/100 cm^2 (β-γ).

As per our response to Question #1, Appendix D, the steam generator storage facility will be provided with a watertight concrete roof; hence the methods and criteria of 49 CFR 173.397 are sufficient. It should be further noted that this closed storage facility will be located within the security-controlled area with access controlled by Health Physics personnel.

4. If you elect to store the steam generators in a facility which is open to the environment (e.g., no roof), provide an estimate of the occupational exposures necessary to decontaminate the external surfaces to the levels specified in the previous question. In addition, provide estimates of liquid and gaseous effluents and volumes of solid waste generated during the decontamination. Provide justification that the decontamination and storage in an unenclosed storage facility will keep radioactive effluents and occupational radiation exposures "as low as reasonably achievable" when compared to storage in an enclosed storage facility.

RESPONSE

As per response to Question #1, Appendix D, the steam generator storage facility will have a watertight concrete roof, which will protect the steam generators from the environment.

5. Section 5.2.2 of the Turkey Point "Steam Generator Repair Report" gives the estimated radioactive releases from the repair effort. Do these releases include expected releases from fuel movement activities? If not, provide the additional releases to the environment resulting from fuel handling operations.

RESPONSE

The estimated radioactive releases provided in Section 5.2.2 of the Steam Generator Repair Report do not include releases associated with fuel movement activities.

The amount of fuel handling associated with a full core offload and full core reload is not expected to be significantly different than the amount of fuel handling associated with a normal refueling outage which includes core shuffle, spent fuel offload, new assembly reload, normally required insert replacement, etc.

Therefore airborne activity levels during the full core offload period are expected to be similar to those associated with a normal refueling. However, since the full core reload will occur at the end of the steam generator repair outage, the airborne activity levels will be lower due to the decay which occurred during the outage period.

Hence, potential releases to the environment associated solely with fuel movement activities during the repair effort are not expected to be different than those associated with normal refueling.

6. It is the staff's position that significant amounts of Fe-55 and Ni-63 may be released during the steam generator repair effort. It is our position that you (1) commit to perform a monthly composite sample of all liquid effluents for Fe-55 and Ni-63 or (2) provide adequate justification for not monitoring for these isotopes.

RESPONSE

Fe-55 and Ni-63 are both low energy emitters which require complex techniques to determine isotopic composition. The gross beta activity is reported in our semi-annual report. The bio-accumulation factor (NRC Reg. Guide 1.109) for these isotopes is not well established. Consequently we do not feel that performing a monthly composite sample of all liquid effluents for Fe-55 and Ni-63 is cost justifiable. However, at least one representative laundry sample will be checked for Fe-55 and Ni-63 during the steam generator repair effort (obtained during the period when steam generator channel head cuts are made) to determine the magnitude of releases.