NUREG-0756

Safety Evaluation Report

related to steam generator repair at Turkey Point Plant Units 3 and 4

Docket Nos. 50-250 and 50-251

Florida Power and Light Company
Supercedes previous SER issued May 1979

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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1.0 INTRODUCTION

By letter dated September 20, 1977 Florida Power and Light Company (FPL) submitted a report entitled "Steam Generator Repair Report-Turkey Point Units 3 and 4."[1] This report was supplemented by revisions 1-7 dated December 20, 1977, March 7, April 25, June 20, August 4, and January 26, 1979 and March 28, 1980, respectively. The report describes a proposed program to repair the six steam generators on Units 3 and Units 4 by replacing the lower assembly, including the tube bundles, of each generator. We determined that the proposed program required our review, approval and issuance of license amendments. The program is the subject of a pending hearing before an Atomic Safety and Licensing Board. On August 3, 1979, an Order was issued granting intervenor status to Mr. Mark P. Oncavage. On May 14, 1979 we issued our Safety Evaluation and on June 29, 1979 we issued our Environmental Impact Appraisal for this program. Since then Revision 7 to the Repair Report has been issued. Our complete and updated safety evaluation of this program is presented in this report. Our updated evaluation of the environmental impact is presented in our Draft Environmental Impact Statement dated December 1980, NUREG-0743.

FPL plans to repair all six steam generators in Turkey Point 3 and 4. The Unit 4 steam generators have the most tubes plugged and therefore will be repaired first. The repair of Turkey Point 3 steam generators is expected to be started about one year later. Since power demands in the FPL system are greatest in the summer, and the repair is expected to take from six to nine months per unit, the repair should be started in the fall in order to be completed before the next summer peak demand. When FPL submitted the repair plan on September 20, 1977 the corporate plan was to be prepared to start the repair for Unit 4 in October 1978. The repair of the Unit 4 steam generators is now scheduled for the fall of 1981 and the Unit 3 steam generators, one year later. This schedule is predicated on the basis that the hearing on this matter has been completed and a favorable decision issued.

The steam generator repair program proposed by FPL for the Turkey Point Plant is similar to the one proposed by Virginia Electric Power Company (VEPCO)[2,3,4] for the Surry Station (plant). The two plants are similar. Each of the plants contain two Westinghouse three-loop PWR units that commenced commercial operation in 1972 and 1973. Both plants originally used a sodium phosphate secondary water chemistry treatment and both plants changed to all volatile chemistry treatment (AVT); Turkey Point in late 1974, Surry in early 1975. The repair program of the Surry units was approved on January 19, 1979. The Unit 2 repair was completed in May 1980 and Unit 1 repair commenced in September 1980.

1.1 History of Steam Generator Operation

Turkey Point Units 3 and 4 began commercial operation on December 14, 1972, and September 9, 1973, respectively. Like almost all units with U-tube design steam generators, they began operation using a sodium phosphate secondary water chemistry treatment. This treatment was designed to remove precipitated or suspended solids by blowdown and was successful as a scale inhibitor. However, during early use many PWR U-tube steam generators with Inconel 600 tubing experienced stress corrosion cracking. The cracking was attributed to free caustic which can be formed when the Na/PO_4 ratio exceeds the recommended limit of 2.6. In addition, some of the insoluble metallic phosphates, formed by the reaction of sodium phosphates with the dissolved solids in the feedwater, were not adequately removed by blowdown. The reaction products of these impurities and of corrosion products with the sodium phosphates tended to accumulate as sludge on the tubesheet and tube supports. In the sludge pile and associated crevices in the central region of the tube bundle where restricted water flow and high heat flux occur, the soluble sodium phosphates became concentrated by evaporative processes and precipitated. This phophate precipitation (hideout) at crevices in areas of the steam generator, noted above, caused localized wastage resulting in thinning of the tube wall. The problem of stress corrosion cracking was corrected by maintaining the Na/PO_4 ratio below 2.6. Although the recommended Na/PO_4 ratio was maintained, it did not correct the phosphate hideout problem or the wastage of the Inconel-600 which increases as the sodium/phosphate ratio is lowered. Largely to correct the wastage and caustic stress corrosion cracking encountered with the phosphate treatment for the secondary coolant have now converted to an all volatile chemistry (AVT). Both Turkey Point 3 and 4 were converted around August, 1974.

In 1975, radial deformation, or the so-called "denting", of steam generator tubes occured in several PWR facilities including Turkey Point 3 and 4, after 4 to 14 months operation, following the conversion from a sodium phosphate treatment to an AVT chemistry for the steam generator secondary coolant. Tube denting is most severe in rigid regions or so-called "hard spots" in the tube support plates. These hard spots are located in the tube lanes between the six rectangular flow slots in the support plates near the center of the tube bundle and around the peripherial locations of the support plate where the plate is wedged to the wrapper and shell. The hard spot areas do not contain the array of water circulation holes found elsewhere in the support plates.

The phenomenon of denting has been attributed to the accelerated corrosion of the carbon steel support plates in the annular spaces where the tubes intersect the support plate due to buildup, by processes analogous to phosphate hideout, of an acid environment in the crevices, containing chlorides. The resultant corrosion product (magnetite) from the carbon steel plate occupies approximately twice the volume of the material corroded. Thus, the continuing corrosion exerts sufficient compressive forces to diametrically deform the tube and crack the tube support plate ligaments between the tube holes and water circulation holes. As a result of the tube support plate deformation, the rectangular flow slots began to "hourglass"; i.e., the central portion of the parallel flow slot walls have moved closer so that some of the flow slots are closed or narrower in the center than at the ends.

On September 15, 1976, during normal operation, one U-tube in the innermost parallel to the rectangular flow slots in steam generator A at Surry Unit No. 2 rapidly developed a substantial primary to secondary leak (about 80 gpm). After removal of the damaged tube and subsequent laboratory analysis, it was established that the leak resulted from an axial crack, approximately 4-1/4 inches in length, in the U-bend apex due to intergranular stress corrosion cracking that initiated from the primary side. Since the initial parallel flow slot wall in the top support plate has moved closer, the support plate material around the tubes nearest this central portion of these flow slots has also moved inward, in turn forcing an inward displacement of the legs of the U-bends at these locations causing an increase in the loop strain and ovality of the tubes at the U-bend apex. It is this additional increase in strain at the apex of the U-bend which is believed to have initiated stress corrosion cracking of the Inconel 600 alloy tubing exposed to PWR reactor coolant. Similarly, leaks have developed in severely dented tubes by primary side stress corrosion as a result of the increase in strain.

Subsequent to the above leak we imposed augmented inservice inspection requirements on Surry Units 1 and 2, Turkey Point Units 3 and 4, San Onofre Unit 1 and Indian Point Unit 2. In addition, operating restrictions and limited periods of operation, typically six months, between inspections are also imposed on severely degraded units, i.e., Surry Units 1 and 2 and Turkey Point Units 3 and 4. The augmented inspection requirements include an assessment of the magnitude and progression of tube denting, and support plate deformation and cracking.

1.2 Reasons for Steam Generator Replacement

The six steam generators at Turkey Point Units 3 and 4 have all undergone a significant amount of degradation since they began operation. The wastage and denting phenomena, discussed earlier, have led to tube wall thinning, support plate flow slot hourglassing and plate ligament cracking, tube denting, stress corrosion cracking, and several instances of reactor coolant leakage through cracked tubes. As of October 1980, tube plugging for various reasons has resulted in removing 20.4% of the steam generator tubes in Unit 3 and 24% of the tubes in Unit 4 from service.

Due to the ongoing denting problem, the certainty that additional tube plugging can result in power derating, and the economic considerations of operating the two units at substantially reduced power, FPL submitted[1] a proposal for the repair of the degraded portions of the steam generators.

2.0 DESCRIPTION OF STEAM GENERATOR REPAIRS

2.1 Mechanical Design and Materials Changes

During 1975 several modifications were made to the steam generators to increase the circulation ratio. The modifications 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 modifying the feedring. These modifications will be retained or improved upon in the repaired steam generators under the proposed repair program. Also, additional modifications, as discussed below, will be incorporated.

A flow distribution baffle plate, located 18" above the tubesheet, will be used. The baffle plate in designed to assist and direct lateral flow across the tubesheet surface, minimize the number of tubes exposed to sludge, and cause the sludge to deposit near the center of the tube bundle at the blowdown intake.

An improved blowdown system is to be incorporated. The new system will use two 2-inch Schedule 40 Inconel internal blowdown pipes which will increase blowdown capacity. 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 repaired generators will have all the tubes expanded to the full depth of the tubesheet to eliminate the potential contaminant concentration sites.

The tube support plate material will be changed from carbon steel to SA-240 Type 405 ferritic stainless steel. The new baffle plates will also be constructed of SA-240 Type 405. This material is much more corrosion resistant in the chemistry expected during operation of the steam generator than in the currently used carbon steel. Corrosion of SA-240 will result in an oxide which is protective under conditions in which carbon steel corrodes rapidly, as demonstrated by laboratory tests.

The new tube support plates will have a quatrefoil design. The quatrefoil design, consisting of four flow lobes and four support lands, provides support to the tube while allowing water flow around it. The design has a lower pressure drop across the thickness of the plate than the existing drilled circulation hole design and results in higher average flow velocities along the tubes, which should prevent sludge deposition.

Also, the tubes will be recessed slightly into the tubesheet holes and then welded to the tubesheet cladding. This design reduces entry pressure losses and eliminates locations for possible crud buildup.

Since the circulation ratio will be greater in the repaired generators, modifications to the moisture separator equipment will be made to accommodate this increase, and minimize moisture and soluble corrodent species carryover into the turbines.

The new lower shell assemblies will have additional access ports that will improve the ability to inspect the tubesheet and flow distribution baffle, and will assist in sludge lancing. A 2-inch nozzle is being added to the upper shell to facilitate the wet layup of the steam generators during periods of inactivity. This nozzle can be used for addition of chemicals to maintain water quality. To lessen downtime and facilitate maintenance and inspection, a 3/8-inch primary shell drain is included in the channel head of the repaired generators to improve drainage of the channel head. Also 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.

Metallic ions from corrosion in the secondary side will be minimized by retubing of the condensers, feedwater heaters, and moisture separator/reheaters with more corrosion resistant materials.

An online condenser cleaning system has been installed to control calcium scale formation on tubes and attendant corrosion. In addition, a full flow condensate polishing demineralizer system is planned to be installed for maintaining feedwater quality.

2.2 Demineralizer System

On September 14, 1979 FPL announced[15] that a condensate polishing demineralizer system (demineralizer) would be installed in the Turkey Point Plant Unit Nos. 3 and 4. This system is planned for installation in the condensate/feedwater system at the discharge of the condensate pumps between the pumps and the No. 1 low pressure feedwater heater. The system's function is to purify the condensate by filtration and demineralization to assure high quality feedwater to the steam generators.

The condensate polishing demineralizer system is designed to process full condensate flow from the condensate pumps. A full-flow bypass system is provided to assure continuous uninterrupted condensate/feedwater system operation. The condensate polishers will be using the powdex resins which will be backwashed into the backwash receiving tank. It is estimated that each vessel will be backwashed at a frequency of approximately once every 21 days. Complete separation of resin and water will be accomplished by the spent resin handling systems. Under normal circumstances the backwash waste will contain a negligible amount of radioactivity in the water. The liquid content of the receiving tank can either be directed to the plant's radioactive liquid waste system or directly to the circulating water cooling system as appropriate. The applicant will be required to take batch samples and analyze the liquid waste prior to release to the circulating water cooling system according to the Plant Technical Specification (section 3.9.1.f). The licensee has indicated that the solid waste separated from backwash receiving tank will be screened for radioactivity content and handled in an applicable manner.

2.3 Heat Treatment of Tubing

The Inconel 600 tubing in the repaired steam generators will be thermally treated to produce a microstructure with improved resistance to stress corrosion cracking by PWR reactor coolant. In addition, the tubes in the innermost eight rows of the bundle will be stress relieved after bending to minimize residual stresses. Several benefits are expected to result from this reduction of residual stresses. These include improved resistance in stress corrosion cracking in NaOH and to intergranular attack in sulphur-containing species.

2.4 ASME Code Regulatory Guide Implementation

All new component parts of the repaired steam generators will be designed and fabricated to the 1974 edition of the ASME Boiler and Pressure Vessel Code, including all addenda through Winter, 1976. Additionally all piping weld and preps, welding, and nondestructive examination will be in accordance with the applicable sections of the lastest edition of the ASME Code. Also, applicable Regulatory Guides will be utilized as identified in the FPL Report[1] (Section 2.14).

2.5 Containment Integrity

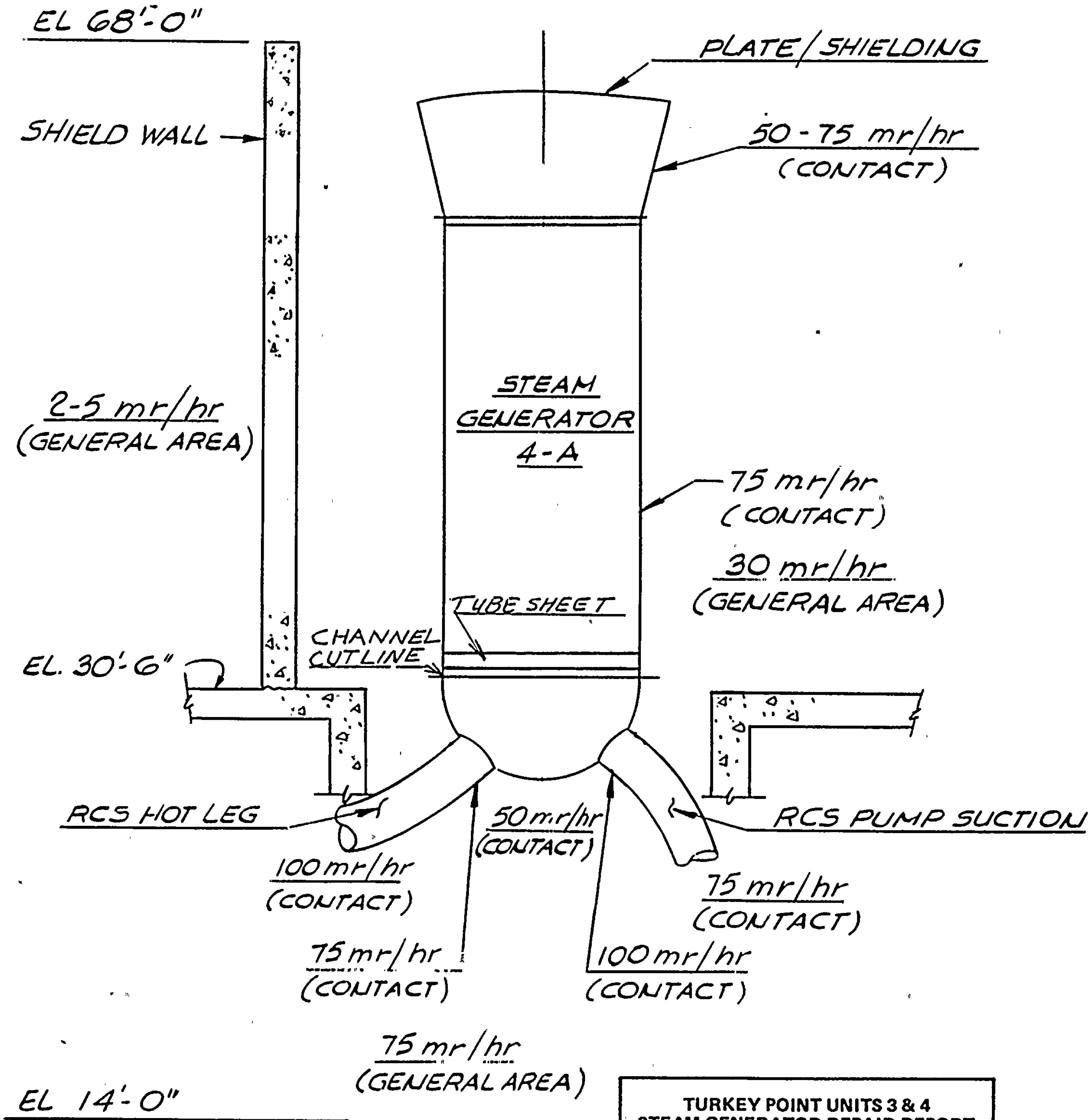
The requirements for containment integrity are specified in Section 15, Technical Specifications, of the Turkey Point Final Safety Analysis Report.[6] These requirements allow the containment to be breached if the reactor is in a cold shutdown condition or in the refueling shutdown condition, which is the case during the repair of the steam geneators. However, once the integrity of the containment has been violated, FPL must satisfy the requirements of Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," 10 CFR Part 50, before returning the reactor to power. Appendix J specifies the leakage testing requirements and acceptance criteria for the containment, containment penetrations, air lock door seals, components and isolation valves. The FPL leak testing program was reviewed and evaluated in Section 5.4 of The Staff SE, dated March 15, 1972.[7] We concluded that the FPL program was acceptable.

2.6 Removal and Reinstallation Operations

The repair will consist of replacing the lower assembly of each steam generator including the shell and the tube bundle and refurbishing and partially replacing the steam separation equipment in the upper assembly. The old lower assembly will be removed from the containment building through the existing equipment hatch and transported to a special storage facility that will be constructed on the Turkey Point site. The new lower assemblies arrived at the site by barge. They will be transferred to a wheeled transporter and hauled on the existing road to the containment building equipment hatch.

Prior to the repair work, the unit will be shut down and all systems will be placed in condition for long term layup. The reactor vessel head will be removed for refueling. All of the normal procedures for fuel cooling and fuel removal will be followed. The fuel will be removed from the reactor and placed in the spent fuel storage facility. The reactor vessel head will be replaced. The equipment hatch will be opened and access control will be established. The biological shield wall and a section of the operating floor concrete and structural steel will be removed to provide access to the steam generator. Guide rails will be installed for transporting the lower assembly through the equipment hatch.

After this preparatory work, the cutting of system piping can begin. This will include cutting and removal of sections of steam lines, feedwater lines, and miscellaneous smaller lines for the service air and water and the instrumentation system. Following the channel cut (Fig. 2.1), all steam generator supports will be disassembled and the steam generator lower assembly will be removed from its location and placed on the equipment floor where it will be prepared for storage and shipment. After the preparation is complete, the



TURKEY POINT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

DOSE RATES AROUND STEAM GENERATOR 4A

Figure 2.1

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assembly will be placed in a horizontal position on a transport mechanism. This mechanism will carry the assembly through the equipment hatch. A mobile crane will lift the lower assembly onto a transporter that will carry it to the steam generator storage facility on the site.

After removal and storage of all three steam generator lower assemblies, their replacements will be transported from the temporary storage location to the equipment hatch. The same machinery used to remove the lower assemblies will be used to install the new assemblies in their cubicles. The steam generator will be reinstalled and the upper assembly with its refurbished internals will be mounted on the lower assembly. After welding the two assemblies together, the piping will be replaced and the biological shield and internal structures will be reconstructed. While the preoperational and startup test program following these major repair activities are still being developed there will be cleaning, hydrostatic testing, baseline inservice inspections, and preoperational testing of instruments, components and systems. Then the reactor will be refueled and startup tests will be performed. The performance of the repaired steam generators will be tested for moisture carryover and verification of thermal and hydraulic characteristics.

2.7 Postinstallation Testing

A detailed preoperational testing program will be carried out by FPL prior to fuel loading to reestablish the integrity of the reactor coolant system and the main steam and feedwater system, to ensure that all systems are in operating condition and to provide baseline data for future performance evaluation. Hydrostatic pressure tests will be performed as well as the baseline inservice inspection of piping. The fuel manipulator crane will be reassembled and tested.

After the residual heat removal system has been tested and placed in service, fuel will be transferred to the reactor vessel. One third of the fuel assemblies placed in the vessel will be new fuel assemblies and the operation will not differ significantly from a normal refueling.

During the initial startup of the unit, tests will be performed to verify the thermal and hydraulic performance of the nuclear steam supply system.

FPL has not yet completed the preparation of detailed procedures for preoperational testing and startup of the unit after completion of the steam generator repairs. We will review the detailed procedures prior to fuel loading to verify that adequate testing will be performed to ensure safe startup of the unit after completion of these repairs.

2.8 Radiological Consideration

A major aspect of the repair effort is its radiological impact, including the occupational exposure accumulated during the repair effort. Battelle-Pacific Northwest Laboratories (PNL) has performed a generic radiological assessment[5] of steam generator replacement and disposal under contract to the NRC. The PNL estimates of occupational exposures (person-rem) were derived by taking into account expected dose rates and the person-hours needed for each maintenance activity.

Maintenance activities were developed by PNL as a composite of the work descriptions for the repair of the steam generators at Surry, Turkey Point and Palisades as determined by VEPCO, FPL and Consumer Power Company.

Person-hour estimates for each activity were developed by PNL based on prior experience with similar activities and on standard estimating techniques.

Dose rates were based on information from several operating PWRs including the Turkey Point Units. PNL usually selected dose rate values on the high end of the range of values measured at the several plants. The PNL estimates of occupational doses are intended to be conservative and represent upper bound values. The PNL estimates are presented as a range of values. The PNL lower value was estimated assuming credit for various techniques to reduce exposures, e.g., providing water shielding by maintaining high steam-generator water levels, and remote tooling and distance where applicable. FPL has committed to these types of techniques; consequently, it is appropriate to compare the PNL lower value with the FPL estimates.

The FPL occupational dose estimates include a detailed estimate of doses of 1084 person-rem per unit based on major job functions. These detailed estimates took into account the dose reduction measures proposed to maintain doses as low as is reasonably achievable (ALARA), including local decontamination, temporary lead shielding, pre-job training, and use of remote tools where practicable. FPL has estimated a range of doses for the steam generator repair program from 1730-2480 person-rem per unit with 2084 person-rem as the best estimate of dose. The range of doses presented represents FPL's best judgment with respect to the predicted worker doses considering uncertainties in prediction of job man-hours and radiation fields. For comparison purposes in this report, we are evaluating the PNL lower estimates versus FPL's detailed estimates.

The differences between FPL estimate of 2084 person-rem per unit and PNL lower generic estimate of 2400 person-rem per unit are within variations expected due to differing procedures and radiation exposure rates from site to site. The FPL dose estimates are based on a range of radiation field values from actual inplant surveys at Turkey Point. We therefore believe that the FPL detailed estimate of 2084 person-rem per unit is a realistic estimate for the repair of the steam generators in one Turkey Point unit. Consequently, in the remainder of this SE, we have used 2100 person-rem* per unit as the projected occupational dose for the steam generator repair work at Turkey Point. Based on our evaluation of FPL and PNL assumptions, as discussed in the following paragraphs, we have concluded that the FPL estimate is more representative of the actual doses. We have included the PNL estimate for comparison purposes.

The FPL estimates include 436 person-rem for miscellaneous activities such as supervision, security, quality assurance/quality control, and health physics. We have divided the estimate equally between the removal and reinstallation phase in this evaluation to permit comparison with the PNL estimates.

*The number 2084 does not indicate four significant figure accuracy, rather it represents a sum of several number of varying magnitudes. Therefore we use 2100 person-rem as the projected occupational dose.

2.8.1 Occupational Radiation Exposure

Separation, disassembly, removal and re-installation of the repaired steam generators must be done in radition fields. Federal regulations as specified in 10 CFR Section 20.1(c) state that licensees should make every reasonable effort to maintain radiation exposures as low as is reasonably achievable (ALARA). The FPL efforts to reduce occupational exposures to ALARA levels are addressed in this section.

The repair program activities can be broken down into four major categories: post-shutdown preparation, steam generator removal, installation of the repaired steam generators, and disposal of portions not reused in the repaired steam generators.

All of the activities associated with the repair activities and return to power have been incorporated into the dose estimates. These include health physics and quality assurance/quality control activities.

2.8.1.1 Post-Shutdown Preparation

The post-shutdown activities include defueling the reactor and storing the spent fuel in the storage pool. The defueling activities will be similar to a normal refueling except that the entire core will be unloaded and the reactor vessel head reinstalled. The time involved in defueling an entire core will be similar to the time involved in defueling, shuffling, and refueling one-third of a core. The radiation fields will be essentially the same as for a normal refueling, consequently, the expected occupational exposure should be similar to a normal refueling.

Following defueling and prior to starting removal of the first steam generator, temporary structures will be installed to facilitate the steam generator separation and removal activities. Water will be kept in the steam generator to provide shielding during the secondary side cut (removal of steam dome). These structures include contamination control envelopes around the steam generators, temporary ventilation systems, scaffolding and a temporary platform with guide rails at the equipment hatch to to facilitate removal of the steam generator lower assemblies.

The contamination control envelopes consist of galvanized steel enclosure structure which make use of existing concrete walls where practicable. Each enclosure structure (one for each steam generator) will have two windows, a ventilation inlet with roughing filter, a HEPA filter exhaust system, and double-door access area. The structure will be joined to the concrete walls and steam generators by angle irons and sealed with an air-setting sealing compound for relative air tightness.

A controlled entrance point for the steam generator project is planned for the 30'6" containment building level. If reactor coolant pumps and piping contribute significantly to the area dose rate, FPL will install temporary shielding. Drain lines and piping will be shielded or routed through infrequently occupied areas.

The preparation activities also include radiation surveys, local decontamination erection of scaffolding. Item to be removed include: piping and electrical components; floor slab and of structural steel; portions of the concrete shield walls to permit later removal of the steam generator lower assemblies, and small sections of containment internal structures.

In order to reduce occupational exposures, many of the activities will be performed with the steam generator secondary side partially with filled water to lower radiation fields. FPL has estimated a total dose of 283 person-rem per unit (excluding refueling) for these post-shutdown preparation activities. The major portion of this dose estimate is attributed to installation of temporary structures, local decontamination, and removal of insulation. FPL has not provided a detailed estimate for installation of temporary shielding. FPL states that the need for temporary shielding will be treated on an individual case basis. The need for shielding will be evaluated based on the dose savings for performing the job with shielding versus the dose incurred during installation and removal of the shielding.

PNL has estimated[5] an occupational dose of 730 man-rem for the post-shutdown preparation activities including 60 person-rem for defueling. The PNL estimate also assumes control of the steam generator secondary side water level to shield radiation emanating from the primary side corrosion products. PNL has included an estimate of 72 person-rem for radiation surveys, local decontamination, and installation of shielding. It is our opinion that some shielding and local decontamination will be necessary. As discussed above, although FPL has not provided an estimate for installation of shielding in the detailed dose estimate, the range of dose estimates FPL has provided does consider the effectiveness of temporary shielding and the time required for installation of the shielding, and is based on FPL knowledge of plant-specific design.

2.8.1.2 Steam Generator Removal

Removal activities include removal of the steam generator upper assemblies, lower assemblies, and major piping (main steam lines, feedwater lines, etc.). The lower assemblies will be removed from the reactor coolant system by parting the channel head of the steam generator. The highest exposure will most likely occur during cutting the channel head and removal of the old steam generator assembly. The cutting of the upper assembly is at the secondary side end of the generator and thus does not involves as much activity. Decontamination of the channel head by mechanical grit blasting and installation of inflatable plugs in the reactor coolant coolant piping is included. The channel head cut method will be performed in a contamination control galvanized steel enclosure with a ventilation system containing a HEPA filter to minimize the spread of airborne particulates. The cutting of the channel head will be done by a remote cutting device which follows a track around the steam generator. Cutting the channel head 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. The new lower assemblies will be welded into the same position as the original lower assemblies. A major part of the person-hours for the channel head cut will be spent in radiation fields which are an order of magnitude lower than those for the reactor coolant pipe cuts.

FPL plans to use remote cutting tools wherever possible to minimize the time personnel stay in radiation areas. It is planning to use mockups to familiarize personnel in the specifics of the cutting operations including space restraints, protective clothing, and special tasks required. The familiarization training should minimize time spent in radiation fields. The opening in the lower assembly, in addition to being sealed, may be shielded to reduce radiation streaming from the internal surfaces.

The 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.

The expected low contamination levels on the secondary side preclude the necessity of using contamination control envelopes at this location to control the spread of airborne activity. The steam generator wrapper and upper internals will be cut from outside the steam generator after the upper shell has been removed. The steam generator water level will be kept high to shield personnel making the cuts from radiation emanating from the lower shell internals.

All openings in the steam generator lower shell will be sealed with welded metal seals prior to removal of the steam generator lower assembly from the containment. The sealed assembly will be rigged for lifting, its supports will be disassembled, and it will then be removed from the containment.

All three existing generators will be removed before any of the new generator sections are brought into the containment. FPL has estimated a total occupational exposure of 1016 person-rem per unit for the removal activities. PNL has estimated[5] a dose of 780 person-rem for the removal phase.

2.8.1.3 Installation of Repaired Steam Generators

The installation phase involves bringing in and installing the new lower shell assemblies, transporting and reinstalling all the removed piping and associated transition pieces, installation/removal of hydroplugs, channel head welding and grinding, installation/removal of the inflatable plugs in the reactor coolant piping, reconstructing the concrete walls removed earlier, removing all temporary work structures, cleanup, and performing preoperational structural integrity tests.

Similar to the removal situation and for the same reasons, the major dose contribution to the installation activities is expected to be from reconnecting the lower assembly. To minimize radiation exposure, an automatic welding device will be used. PNL has estimated[5] a savings of 860 person-rem per generator (2600 person-rem per unit) based on using remote welding as compared to manual welding. This yields a total PNL estimated exposure of 620 person-rem per unit for the installation phase. FPL has estimated the exposure for this phase to be 644 person-rem per unit.

2.8.1.4 Disposal of Portions Not Reused

Disposal also affects the occupational exposures. This entails transportation to and placement in the storage facility. A description of this facility is contained in Section 2.8.2. Similarly there will be occupational exposure should the offsite shipment alternative be available. FPL has estimated that these alternatives would involve 50 person-rem each. We have evaluated these alternatives and find these numbers are reasonable.

2.8.1.5 ALARA Consideration

FPL's total estimate of 2084 person-rem per unit for the channel cut method takes into account the dose reduction measures described in Regulatory Guide 8.8, Rev. 3, "Information Relevant to Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," which include local decontamination, temporary lead shielding, pre-hob planning, pre-job training, and use of remote tools were practicable. In addition, FPL estimated a range of exposures from 1830-2480 person-rem/unit based on uncertainties regarding job person-hours, radiation fields, and the effectiveness of temporary shielding. PNL has estimated[5] a minimum total dose of 2400 person-rem per unit for a generic repair program.

FPL has committed to making every reasonable effort to keep radiation exposure ALARA in accordance with 10 CFR Section 20.1(c) and Regulatory Guide 8.8, Rev. 3 and is responsive to the Regulatory Staff Positions in Regulatory Guide 1.8, "Personnel Selection and Training", 8.2, "Guide for Administrative Practices in Radiation Monitoring," and 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." We conclude that the FPL efforts to maintain occupational doses ALARA during the repair efforts meet our positions in Regulatory Guide 8.8 and are therefore acceptable.

FPL will use some experienced personnel from the Surry Unit 2 steam generator removal and replacement. These individuals will provide added expertise to FPL for dealing with health physics problems associated with the repair. Communication of this knowledge gained during the Surry Unit 2 operation is a key ingredient in an effective ALARA program.

All craft personnel will be required to take training in radiological protection. The course will include instructions and demonstration in radiation protection principle, theory and practice, emergency planning and the FPL Radiological Protection Program. Personnel will be required to pass a comprehensive examination to have unescorted access in the radiation-controlled area.

Extensive training in other areas will be used throughout the repair. FPL has stated that scale models will be used to familiarize supervisory and key craft personnel with the repair effort. The models will be used to develop construction work plans to establish the most efficient work procedures. The models will also supplement work plans and allow supervisors and craft personnel to achieve the most efficient use of manpower which will reduce occupancy in radiation fields and, thus, reduce the total occupational dose. These models include a model of the entire containment which will be used in conjunction with radiation survey data to establish temporary shielding requirements. The model will also be useful in making person-rem assessments for particular work activities in radiation fields.

FPL will provide additional facilities for the repair effort including radiological protection training facility, additional health physics area (include separate offices for HP personnel) counting room, access control station, laboratory facilities, change room, and decontamination facilities.

Postoperational debriefings will be used for jobs incurring major radiation exposure. Feedback of dose experience will be examined with the aid of computer printouts of doses.

FPL is committed to provide a qualified radiological engineer assigned the responsibility and authority to function as ALARA coordinator in accordance with Regulatory Guide 8.8. This function will be the individual's primary function. The FPL corporate staff is developing an ALARA program and the HP manual incorporating the radiation protection program will be revised and the formal ALARA program implemented in early 1981. An ALARA engineer is budgeted for 1981 and will be integrated into the radiation protection organization.

Based on the above evaluation, we determine that the programs and procedures proposed by FPL in making the steam generator repairs demonstrate that it will meet the requirements of (1) 10 CFR Part 20 limits and as it relates to effort to maintain radiation exposure as low as reasonably achievable; (2) Regulatory Guide 8.8, as it relates to management policy and organization; personnel qualifications and training; design of facilities and equipment; radiation protection program, plans, and procedures; and the availability of supporting equipment, instrumentation, and facilities; (3) Regulatory Position C.1.f of Regulatory Guide 8.10, on modifications to reduce radiation exposures.

2.8.2 Disposal of Steam Generator Lower Assemblies

The steam generator lower assemblies will comprise the largest source of radioactive waste requiring disposal. Several options for the disposal of the lower assemblies were considered:

(1) Immediate intact shipment to a licensed burial facility;
(2) Delayed intact shipment to a licensed burial facility;
(3) Immediate cut-up and shipment to a licensed burial facility;
(4) Delayed cut-up and shipment to a licensed burial facility;
(5) Onsite storage until facility decommissioning.

Because of the size and packaging involved, the only practical method for immediately shipping the assemblies intact would be by barge. FPL considers this a viable option and is actively pursuing the resolution of the problems involved such as barge offloading facilities, available disposal space, etc. We have evaluated this option[16] and find that if FPL follows applicable laws regarding such shipments, this option will provide reasonable assurance that the health and safety of the public will not be endangered and is acceptable.

Immediate cut-up and shipment is possible with transportation by truck or rail. The assemblies could be cut into suitable sized segments and packaged and transported. Cutting of the assemblies and subsequent handling would result in increased occupational exposures due to the activity on the surfaces exposed to reactor coolant. Some dose reduction could be achieved by decontamination of the reactor coolant surfaces. However, effective decontamination factors may not be achievable due to presence of a significant number of plugged tubes which would prevent decontamination chemicals from entering 20% or more of the tubes. Reduced exposures due to decontamination would be accompanied by a significant increase in decontamination solution liquid radioactive wastes. These wastes would have to be processed and solidified.

We conclude that immediate cut-up and offsite shipment will cause an unnecessary person-rem burden on the workers without providing a significant operational benefit to FPL and to the public as compared to onsite storage as discussed below.

FPL has proposed that the immediate intact offsite shipment is a preferred method of disposal; however, plans also include long-term onsite storage which would allow for decay of radioactivity to relatively low levels to minimize radiation exposures before processing for shipment. The lower assemblies would be stored in an engineered storage facility specifically constructed for this purpose. Such storage would provide for FPL responsibility and control of access and exposure to the assemblies until the radiation has decayed to levels that will allow easy disposal (e.g., Unit decommissioning). Based on decay of the expected radioactive corrosion products, it is estimated that storage for 30 years can reduce the radiation levels to less than 1% of those expected when the assemblies are removed from containment. The assemblies will be sealed with steel plates or plugs prior to removal from containment to eliminate airborne particulates from being released from internal surfaces. Internal decontamination will not be necessary because of the seals. Some surface contamination will be present on the outside of the assemblies. FPL has stated that the external surfaces will be decontaminated such that removable contamination levels will be less than 2200 dpm/100 square cm prior to removal from containment. We will require the welds to be coated to seal out moisture. Therefore, any release to the environment from transport of the assemblies to the onsite storage facility should be negligible.

We have reviewed the FPL storage building. Because the external contamination levels will be <2200 dpm/100 square cm, airborne releases from the external surfaces of the generators are not expected. FPL has proposed quarterly surveillance of the facility consisting of visual inspections and random swipes of the generators and area radiation surveys to assure that no airborne contaminants are being released from the facility. We have reviewed the FPL proposed surveillance program for the storage facility and find it acceptable.

There will be a limited amount of direct radiation which penetrates the storage building walls. Based on the maximum expected radioactive inventory of the steam generators and the shielding of the storage facility, FPL has estimated, using commonly accepted practices, an annual dose of less than one mrem to an individual at the site boundary. We have reviewed the bases for this estimate and consider the bases acceptable. We conclude that the expected radiation levels on contact with the outside of the facility walls are approximately the levels for unrestricted areas specified in 10 CFR Section 20.105. If upon completion of the storage phase FPL finds levels in excess of 10 CFR Section 20.105, FPL will be required to provide adequate control and posting pursuant to 10 CFR Section 20.203. We have reviewed the FPL proposed surveillance program for the storage facility and find it acceptable.

Our review of the five disposal options available is summarized in Table 2.1. This compares with the FPL estimates summarized in Table 2.2. Based on this summary, we have concluded that immediate offsite shipment would be an acceptable method of disposal provided the necessary State approvals are obtained. We have reviewed the FPL proposed method of storage, should the offsite shipment option not be available, and conclude that there is reasonable assurance that this storage will not endanger the health and safety of the public and is acceptable. In addition, we conclude that the measures to be taken to control and monitor this storage will keep occupational exposures and radioactive effluents as low as reasonably achievable.

2.9 Quality Assurance

FPL has made a commitment that the Quality Assurance Program for the repair of the steam generators will be in accordance with the Florida Power and Light "FPL Quality Assurance Topical Report" (FPLTQAR 1-76A*),[8] except as amplified in Section 3.6.1 of Rev. 3 of the FPL Steam Generator Repair Report. We find these amplifications to be acceptable clarifications of FPL commitments contained in FPLTQAR 1-76A. Work performed by Bechtel on the repair of the steam generators will comply with the "Bechtel Quality Assurance Program for Nuclear Power Plants" (BQ-TOP-1).[9]

The Quality Assurance Program for the design and fabrication of the steam generator replacement lower shell assemblies and other components will be in accordance with the Westinghouse Electric Corporation topical report (WCAP-8370 Rev. 9A).[10]

Each of the above reports has been reviewed by the NRC for compliance with Appendix B to 10 CFR Part 50 and has been found acceptable. We now have reviewed the aforementioned reports with specific consideration for the proposed steam generator repair. Based on our review we find that: (1) the repair activity is within the scope of the approved programs, and (2) adequate controls exist within the approved programs for the proposed work activities. Accordingly, we find the provisions established for the quality-related activities associated with the repair of the steam generators acceptable.

*Rev. 2A is now in effect.

Table 2.1

Steam Generator Disposal Alternatives[5]

Option	Approximate Person-Rem per Steam Generator	Approximate Airborne Release, Ci Per Generator
Immediate intact shipment	2.4[c]	Negligible[b]
Long-term[a] storage (including surveillance) with intact shipment	10	Negligible[b]
Long-term[a] storage with cut-up and shipment	16	0.005
Short-term storage with cut-up		
at 5 yr	230	0.026
at 15 yr	60	0.015
Immediate cut-up and shipment by rail/truck - no decontamination	580	0.042
Immediate cut-up and shipment by rail/truck - with chemical decontamination	270	0.010

[a] 30 to 40 years

[b] Since the steam generator will be sealed before it is removed from containment, no release of radioactive material is expected during the repair operation.

[c] Estimates for short-term storage followed by intact shipment would be only slightly larger than this, perhaps 5 person-rem.

Table 2.2

Cost of Alternate Disposal Methods (FPL)[1]

	Method	Cost, /Dollars	Exposure[a], Person-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
c.	Cut-up and disposal near term	$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

[a]Note that these doses are for six lower assemblies. The estimates in Table 2.1 are for one lower assembly.

3.0 EVALUATION

Several design changes, as discussed above, will be incorporated in the repaired steam generators. Our evaluation of these changes is given below.

3.1 Effects of Steam Generator Design Changes

The existing steam generators contain large amounts of sludge which has contributed to their previously discussed degradation. Since an AVT secondary water chemistry treatment will be used when the repaired steam generators begin operation, and residual phosphates will not be present in the system, any sludge which accumulates should not be of a chemical composition that could lead to degradation of the repaired generators. In addition to the elimination of phosphates, the retubing of the condenser, feedwater heaters, and moisture separator/reheaters, and the installation and use of condensate polishers will essentially eliminate sludge. Furthermore, even if sludge should form, we concur that a flow distribution baffle plate should minimize, or at least reduce, the number of tubes exposed to the sludge, and cause the sludge to deposit near the blowdown intake. Use of this baffle plate, in conjunction with the increased blowdown capacity, will reduce the amount of sludge that can accumulate in the generator.

Full-depth expansion of the tubes in the tubesheet is an improvement over the existing partially expanded arrangement and will minimize both crevice boiling and buildup of impurities in the tube to tubesheet crevice region.

A quatrefoil support plate design will be used in the repaired steam generators. In contrast, the existing steam generators use drilled hole support plates which have a very limited opening between the tube and tube support plate. The majority of flow in this drilled plate design is through separate circulation holes. The tube denting phenomenon, discussed earlier, has occurred when corrosion products (magnetite) have built up in the tube/tube support plate intersections (annuli) to the extent that the gap between the tube and support plate closes completely. The broached or quatrefoil design has no separate circulation holes. Substantial flow and much flow velocity will take place through the large open spaces in the quatrefoils around each tube. This results in a continuous flushing action, tending to wash out this tube/tube support plate area and thus prevent sludge deposits or scales.

The quatrefoil support plate design has led to some tube degradation, in the form of a type of erosion cavitation mechanism, in once-through steam generators. Although FPL has suggested that this will not be a problem in recirculating designs, we feel that the phenomenon is not understood well enough to assume that recirculating type designs will not see this type of degradation. Despite this reservation and for the reasons discussed above with regard to tube denting, we concur that the quatrefoil support plate design is an improvement over the existing drilled hole design and should be less prone to denting.

The repaired steam generators will use SA-240 Type 405 ferritic stainless steel for both the tube support plates and flow distribution baffle plate. The corrosion data provided indicate that, under the test condition, Type 405 stainless steel will be a greatly improved material for tube support plates over the carbon steel presently used. In the event that denting reactions be initiated, we would have some concern over the propensity of this material for stress corrosion cracking in a chloride environment. However, Westinghouse appears to have taken the proper precautions in stress relieving it to minimize the likelihood that stress corrosion will occur in the absence of denting.

The Inconel 600 tubing will be thermally treated, which should result in improvement in its resistance to stress corrosion cracking in the reactor coolant and secondary water, particularly in the U-bend regions. Further, in the eight innermost rows of tubes, the U-bends will be stress relieved after bending. We find this residual stress relieving process to be satisfactory and an improvement over existing practice.

Summary

Based on the information discussed and the evaluation made above, we conclude that the structural, mechanical, and materials aspects of the FPL proposed steam generator repair program are acceptable and there is reasonable assurance that the health and safety of the public will not be endangered. We further conclude that the new steam generator design has incorporated features to eliminate the potential for various forms of tube degradation observed to date.

3.2 Effects of Repair Activities

3.2.1 Protection of Safety-Related Equipment

FPL will take measures and establish controls to prevent construction accidents and protect safety-related structures, systems, and components from the hazards associated with steam generator transportation and repair activities. The general precautionary measures that will be taken by FPL include the following:

1. The reactor vessel will be completely defueled prior to the repair work.

2. The entire repair process will be preplanned to assure that it can be completed safely and efficiently.

3. The repair program will be carried out in accordance with the FPL corporate Quality Assurance Manual (FPS-NQA-100) and Section XI of the ASME Code. Bechtel Corporation has been retained by FPL as the Architect-Engineer for the repair program.

4. The containment boundary will not be disturbed except to open the equipment hatch.

5. The existing polar crane trolley will be replaced by a higher capacity temporary construction hoist. The temporary hoist will be inspected and tested prior to its use for construction lifts on the polar crane bridge and the removal of the steam generators.

Defueling of the reactor will begin shortly after shutdown and the normal procedures for defueling will be followed. The fuel will be stored in the spent fuel storage pool for the duration of the outage. The temperature of the pool is normally maintained at 95°F or less when the pool contains all of the fuel from the core and the spent fuel elements currently being stored. We independently estimated the cooling capability of the fuel pool cooling system in its evaluation of the increased storaged capacity of the pool. With our assumptions, including transferring the fuel 150 hours after shutdown, the calculation indicated that the fuel pool temperature would not exceed 139°F. The spent fuel cooling system consists of two redundant trains designed to seismic requirements. If a failure were to disable one of these trains, the remaining train could maintain the pool water temperature below 160°F. We find that these temperatures are acceptable.

In addition, if all of the system cooling of the fuel pool of the unit under repair were to be lost, the fuel pool could still be cooled by the operating unit cooling systems. The component cooling water (CCW) system of the operating unit has sufficient capacity to supply operational cooling needs, including its spent fuel pool, as well as the cooling needs of the spent fuel pool of the unit under repair, through the existing piping interties between Units 3 and 4. Moreover, based on our independent analysis, if the spent fuel pool cooling were to fail following a full core offload, the heatup rate would be such that boiling of the pool water would take 8-1/2 hours. This is sufficient time in which to make repairs or find an alternate source of makeup water for the spent fuel pool. Therefore, the present cooling capacity of the spent fuel pool and available makeup sources is adequate for the complete defueling of the reactor as planned for the steam generator repair activities.

In addition, specific potential hazards considered by FPL included the dropping of a steam generator lower assembly, a transporter accident, the toppling of a crane, the interaction of systems shared by both units and fires, each of which is discussed below.

In assessing potential hazards associated with the transportation of the steam generator lower assemblies; FPL considered failures of the transporter which consists of a semitrailer and a haul vehicle. FPL considered structural failure, overturning, and road failure. In considering overturning, the licensee found that it would require the loaded trailer bed to be inclined beyond a 31° angle from the horizontal. The planned side slopes of the haul route are far less than this 31° angle. Further, administrative limits will be placed on the turning radius and speed of the transporter to preclude overturning. The roadway along the haul route has been evaluated and appropriate sections will be upgraded in order to preclude roadway collapse or damage to the facilities that pass under it, such as electrical duct banks and intake cooling water lines.

FPL has considered the consequences of dropping a steam generator assembly (the heaviest load to be lifted during this repair program) either inside or outside the containment building. Since there will be no fuel in the containment building while heavy loads are being lifted, there will be no hazard associated with fuel assemblies. FPL has evaluated the consequences of a postulated drop of the 205*-ton steam generator lower assembly on buried facilites along the haul route. These include intake cooling water piping and electrical duct banks. Because of the existing cooling water interties between the two reactor units, the cooling systems would be realigned as necessary to provide cooling to a possibly damaged cooling system of one of the units. In the event of damage to the local control cables, alternate starting procedures for the affected

*Reduced to about 173 tons by virtue of the channel cut versus the coolant pipe cut.

pumps are available. With regard to dropping a steam generator assembly outside of the containment building, no other safety-related structures (such as the radioactive facility and the fuel storage building) are within the range of the devices used to lift the steam generators from the equipment hatch platform to the transporter. Based on our review of the FPL consideration above we have concluded that dropping a steam generator lower assembly will present no undue risk to safety-related structures.

FPL considered the toppling of a crane having a 70-foot boom. The potential consequences of such an accident were considered with respect to the safety-related structures, systems and components of the operating unit. The diesel-generator building and the auxiliary building were determined to be able to withstand the boom impact without penetration that would result in damage to equipment necessary for the safe shutdown of the operation unit or, in the case of the auxiliary building, the maintaining of the spent fuel pit cooling system. During the repair the fuel is removed from the affected containment building so that a toppling of the crane on this containment would not present a safety problem. Damage to the refueling water storage tanks and the primary water storage tanks, located along the proposed haul route, is precluded since the crane boom will be in the lowered position while traversing these roads. Based on our review of the FPL considerations, we have concluded that the falling of the crane boom on these safety-related structures would not prevent the safe shutdown of an operating unit and would not prevent adequate cooling of the fuel assemblies in the spent fuel pool.

3.2.2 Other Interactions with the Operating Unit

The normal and emergency electrical power distribution systems were reviewed to ensure that construction loads will not jeopardize the supply of electrical power to the operating unit. The results of that review are discussed below.

3.2.2.1 Offsite Power System

The offsite power supply system consists of two startup transformers to the Turkey Point Units 3 and 4. Each of the two units has a dedicated startup transformer which can automatically supply all AC power to both safety and nonsafety loads of each unit. Each startup transformer is capable of supplying the auxiliary loads for its associated nuclear unit and the safety loads for the other nuclear unit.

The temporary loads which are required for the repair of a steam generator will be fed from a temporary 1500-kVA transformer. After the reactor has been brought to cold shutdown the temporary transformer will be energized by the nonsafety 4.16-kV supply system through the startup transformer to the switchgear of the reactor coolant pump of the unit under repair, the onsite electrical distribution system will be configured the same as during a normal plant refueling shutdown. On this basis we conclude that the temporary electrical system modification will not degrade the onsite power system in the operating unit. A fault in this temporary load distribution addition will not cause a loss of power on the reactor coolant pumps in the operating reactor.

3.2.2.2 Emergency Power System

The onsite emergency diesel-generator system for Turkey Point Units 3 and 4 consists of two diesel-generators. The diesel-generators start on either a safety injection signal or on the loss of voltage on a 4160V bus(es) of either unit.

Upon loss of voltage, the following automatic sequence is initiated:

(1) Diesel-generators are started;
(2) "Preferred supply" breakers of the 4160V buses are tripped;
(3) Diesel-generator supply breakers close.

In case of a safety injection signal on the operating unit in coincidence with the loss of power, step 3 above is followed by the sequential starting of all engineered safeguard equipment for the operating unit.

In case of a safety injection signal in the operating unit, without loss of power, both diesel-generators are started and maintained in an idling mode.

During the repair of the steam generators, the engineered safety features (ESF) equipment of the unit under repair will be disabled after the reactor has been defueled by the associated feeder breakers being locked open and tagged. In addition to the lockout of power to the ESF loads connected to the buses of the unit under repair, all those buses that can carry any initiation signal to the shared diesel-generators and which could potentially cause them to become dedicated to the unit under repair (and its loads) will disabled by disconnection. This step is necessary in order to prevent any possibility that the shared diesel-generators, and its loads, may become dedicated to the unit under repair. We find these provisions, proposed by FPL to ensure dedication of onsite emergency power to the operating unit, acceptable. Upon completion of the steam generator replacement work in each unit, the circuitry is to be tested for proper performance prior to the resumption of power operation for that particular nuclear unit.

With regard to the power requirements for the spent fuel pool cooling, we have determined that emergency power, assuming a total loss of offsite power, is not required to be available in less than 8 hours for any safety functions. FPL has confirmed that power could and would be restored within 8 hours by manual operator actions to the fuel pool of the unit undergoing repair. We require that FPL prepare procedures to ensure this latter capability.

The diesel fuel-oil storage system for the two diesel-generators at Turkey Point Units 3 and 4 consists of two day tanks within the diesel generator buildings and one main storage tank outside the building. Inasmuch as the day tanks have a combined capacity of only 8000 gallons, the main storage tank must be operational in order to meet the Technical Specifications for the plant, which require that there be an availability of 40,000 gallons of diesel fuel-oil.

During the construction phase of the steam generator repair program, the containment ramp will be removed and replaced by a temporary loading platform. Inasmuch as the containment ramp of Unit No. 3 is a part of the oil retention dike around the main storage tank, the removal of the ramp eliminates the fire protection feature of the dike. In view of this, some remedy is needed in order to restore the main fuel-oil storage facility to its fully available condition.. One alternative is to replace the missing portion of the dike with a temporary structure, and the other alternative is to drain the diesel fuel-oil from the main storage tank and place the fuel-oil in a temporary location elsewhere on the site. If the fuel-oil is placed in a temporary location, the supply must be verified to be operational prior to disabling the permanent system.

Either of these alternatives is acceptable in concept. However, the final choice by FPL must be designed to assure that the Technical Specifications of the plant are satisfied and that the choice meets the minimum NRC standards and requirements associated with the operating license. This will include appropriate application of quality assurance and seismic site requirements to any temporary structures, piping and components; of cleanliness requirements on the fuel-oil; and of other existing functional and operation requirements of this fuel oil supply. We require that these details be addressed and adequately demonstrated by FPL prior to initiating the construction changes affecting the fuel-oil retention dike surrounding the main diesel fuel-oil storage tank.

Summary

The spent fuel pool emergency power requirements are acceptable on the basis that FPL submits acceptable procedures to ensure that power can be restored to the spent fuel pool of the unit under repair within 8 hours. The diesel oil fuel storage supply is acceptable, however, the details of the FPL plan to assure the diesel fuel supply during the repair of Unit 3 must be addressed and adequately demonstrated prior to initiating the construction changes affecting the fuel oil supply. On the basis of our above review and the satisfactory resolution of the conditions cited, we conclude that the provisions by FPL to ensure dedication of onsite emergency power to the operating unit are acceptable.

3.2.3 Fire Protection

An evaluation of the fire protection program for the Turkey Point Plant Units 3 and 4 containment buildings was included in the NRC Safety Evaluation dated March 21, 1979[11] and supplemental information dated April 4, July 9, and October 9, 1980.[13] This information is supplemented by the FPL report "Steam Generator Repair Report for the Turkey Point Power Station, Units 3 and 4,"[1] which addressed the specific fire hazards associated with the steam generator repair outage. In this regard it should be noted that a fire inside containment cannot cause offsite radioactivity exposures of consequence because the fuel will be removed from the containment of the unit under repair nor can it impair the safe shutdown capability of the plant. Nevertheless, the following is a summary of the fire protection measures to be taken during the repair operations.

The use of combustibles in the containment will be minimized to the extent practicable. Fire retardant scaffolding and materials will be used. Good housekeeping will assure that wood crates and other combustible trash are removed from the containment in a timely manner.

However, additional amounts of combustible materials will necessarily be introduced into containment including protective clothing, cleaning fluid, charcoal filters, and plastic sheeting, but the use of these will be minimized in those areas in which cutting and welding is being done.

The fire protection for the containment consists of fire extinguishers throughout the containment, and portable fire extinguishers will be accessible in the work areas when cutting and welding is performed. A portable foam system suitable for use inside containment on liquid hydrocarbon fires will be onsite and promptly available upon demand throughout the repair. The existing containment lighting system and emergency lighting are available.

Even though FPL will not provide a permanently installed fire water standpipe system in each containment before the initiation of the steam generator repair program, a fire hose of sufficient length to reach the most remote steam generator compartments will be available and dedicated to fight fire inside containment. A fire watch will be continually present during all welding and cutting operations.

Administrative controls related to fire protection are presently in effect at the plant and are applicable during the steam generator repair outage. Additional fire protection personnel will be assigned to the replacement activities in the containment. All administrative site procedures will be reviewed for the control of combustibles and these procedures will identify all known potential fire hazards. A fire plan for the repair activities will be formulated and coordinated with the station fire plan.

Based on our review of the fire protection measures to be taken to protect safety-related structures, systems, and components, we have concluded that there is reasonable assurance that the proposed construction activities can be conducted without significantly increasing the potential for damage to safety-related systems.

3.2.4 Water Chemistry

We have reviewed the demineralizer system described in Section 2.2. Based on our review of the information submitted by the licensee, we conclude that the proposed design modifications to include the condensate polishing demineralizer system in the condensate/feedwater system is for improving the quality of feedwater to the steam generators and it is independent of the steam generator repair program. We find the proposed modification to be acceptable and it does not change our conclusions of the evaluation of radioactive waste management systems provided in the Safety Evaluation Report[7] for Turkey Point Plant Units 3 and 4 dated March 1972.

However, to assure that the water chemistry control and monitoring program is appropriate, we require that prior to plant re-startup, FPL should submit for evaluation by the NRC a steam generator secondary water chemistry control and monitoring program which will address the following:

1. Identification of a sampling schedule for the critical parameters and of control points for these parameters for each mode of operation: normal operation, hot startup, cold startup, hot shutdown, cold wet layup;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions* for off-control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data and (b) the sequence and timing of administrative events required to initiate corrective action.

FPL should verify that the steam generator secondary water chemistry control program incorporates technical recommendations of the NSSS vendor. Any significant deviations from NSSS vendor recommendations should be noted and justified technically.

3.2.5 Decontamination of the Channel Head

On November 4, 1980, FPL responded to our letter dated September 29, 1980 regarding the decontamination process to be used for the channel head area of the steam generator. The FPL response indicated that the evaluation of the alternate methods of decontamintion was not completed and some generic statements regarding the process were made. On November 21, 1980, this answer was supplemented by FPL with a statement that the preferred method was one which involves high-pressure spraying of grit and water slurry to mechanically remove the film of corrosion products and would be similar to the system used at San Onofre Unit 1.

FPL has indicated that, to achieve the objectives of reducing occupational exposure to radiation to as low as reasonably achievable, decontamination of the channel head region of the steam generators will likely be performed. The interior surfaces of the channel head and divider plate in each steam generator will be decontaminated prior to the final parting cut that will separate the tube sheet/tube bundle and lower shell from the channel head, should the decontamination effort show potential person-rem savings over the cutting operation without decontamination. Following removal of the lower shell, further decontamination will be made on the channel head/coolant pipe interior surfaces. The method of decontamination will involve high-pressure spraying of a grit (either magnetite or alumina) and water slurry to remove radioactive corrosion product film from the surfaces. The decontamination system would be similar to that used recently by the San Onofre Nuclear Generating Station Unit No. 1.

We have previously reviewed the decontamination program of San Onofre for steam generator repair and found it acceptable. We determined that no adverse effect would result when magnetite is replaced by alumina in the grit slurry injection method of decontamination. On this basis, we determined that the

*Branch Technical Position MTEB 5-3 describes the acceptable means for monitoring secondary side water chemistry in PWR steam generators, including corrective actions for off-control point chemistry conditions. However, the staff is amenable to alternatives, particularly to Branch Technical Position B.3.b(9) of MTEB 5-3 (96-hour time limit to repair or plug confirmed condenser tube leaks).

San Onofre decontamination method using either magnetite or alumina may be used at the Turkey Point Plant Unit Nos. 3 and 4 during the steam generator repair work provided that the following conditions are met. These conditions have been imposed on San Onofre Nuclear Generating Station as a result of the planning and operating experience obtained from the decontamination at the San Onofre plant.

1. The pressure in the inflatable plug seal in the RCS pipe nozzles should be monitored. Upon loss of seal pressure, injection of the grit slurry should be stopped immediately and the seal plug replaced.

2. Written procedures should be provided to include accountability controls of all tools, equipment, materials, and supplies that are to be used in the channel heads to prevent inadvertent entry of such items into the reactor primary coolant system. These controls should be in effect whenever the inflatable plug seals and their associated cover plates are not in place in the nozzles of the reactor coolant system piping.

3. Written procedures should be provided to restrict materials to be used in the channel head area to prevent the presence of materials having potential adverse effects on the reactor coolant system components (for example, chloride-bearing materials).

4. Written procedures should be provided to include instructions to require that the channel head area, including the nozzles, be inspected and confirmed to be free of all loose materials, equipment, and tools prior to removing the cover plate from the inflatable plug seal.

5. Prior to closing up the reactor coolant system and starting the RCS pump, any loose debris, including the abrasive grits, in the channel head, RCS hot leg, and cold leg should be cleaned up.

6. Prior to resumption of power operation, FPL should submit for NRC review and acceptance a report which will include an analysis of the possible effects of any foreign material which has entered the primary coolant system and has not been retrieved. The report should include all work on the decontamination and steam generator repair.

Based on the above evaluation, we determined that the proposed decontamination will meet (1) the requirements of General Design Criterion 14 of Appendix A to 10 CFR Part 50, as it relates to the effects of decontamination on primary system boundary material; (2) 10 CFR 20.1(c) as it relates to efforts to maintain radiation exposure as low as reasonably achievable; (3) Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to reducing crud accumulations by fluid flushing; and (4) Regulatory Position C.1.f of Regulatory Guide 8.10, on modifications to reduce radiation exposures. On this basis, and subject to the licensee's meeting the six conditions above, we conclude that the steam generator decontamination to be performed at Turkey Point Plant Unit Nos. 3 and 4 is acceptable.

3.2.6 Sealing of the Used Steam Generator Lower Assemblies

On October 31, 1980 FPL responded to our request of October 21, 1980 for information regarding the "sealing" of the used steam generator lower assemblies before storage onsite and/or shipment offsite. On November 21, 1980 additional information was submitted on the same subject.

We have reviewed the information submitted and find that the "drying" of the inside of the generators is adequate to prevent leakage of radioactive material to the outside due to internal corrosion. This is based on the fact that the generators will be drained and that the oxygen depletion study shows that the secondary side water would not be sufficient to support through-wall corrosion. In addition, to assure that no additional oxygen can get into the generators, a heavy body varnish, such as glyptol, should be applied to all seal welds and to a two-inch area on either side to assure adequate coverage.

The November 21, 1980 FPL submittal also provided additional details on the weld closures. The submittal stated that FPL plans to seal the steam generator lower assemblies using 1-1/2-inch-thick fillet welds. The submittal also stated that although the weld design and specific procedures have not yet been finalized, the following statements may be made about the weld:

The steam generator channel head weld material will be E7018 ACC welding electrodes. The welding will be done in accordance with ASME Section IX procedures and standards. The weld will be a 1-1/2-inch-thick fillet weld, one inch thick at the throat. The ultimate strength of the weld will be 72,000 psi, and the design allowable strength will be 21,000 psi. Little weld preparation is expected to be necessary; we currently anticipate only some minor polishing of rust or other surface imperfections (if any) to be necessary.

A weld with the above characteristics would not be breached in a 12-foot drop in certain drop modes while it might be breached in certain others. In order to obtain assurance under all modes that the channel head weld would not be breached, the proposed 1-1/2-inch-thick fillet weld could be increased to a considerably larger fillet weld. In order to make this larger fillet weld, the number of man hours required would increase by approximately an order of magnitude, thus significantly increasing the occupational exposure required, as well as the cost. FPL does not believe it is consistent with the ALARA concept to incur a substantial and certain increase in occupational exposure in order to prevent releases which would only occur in the case of a very low probability accident, and which are calculated to be low.

We have reviewed the above information and find it an acceptable way to weld the steam generator assemblies. Since there is a small possibility of a breach in the weld if dropped during a 12-foot lift, we will require that FPL provide the NRC with lift procedures, including applicable QA, and a description of the lift equipment to be used. This will provide additional assurance that the risk of exposure is acceptably small.

3.3 Transient and Accident Analyses

3.3.1 Discussion

This section discusses the effect the repaired steam generators have on the transient and accident analyses. As can be seen from Tables 3.3-1 and 3.3-2, FPL has stated that the majority of the relevant design parameters and plant operating parameters will not be changed from those for the present steam generators during steady state. Therefore, systems responses to transient conditions with the repaired steam generators are expected to be essentially the same as for the original steam generators prior to tube plugging. The impact on the transient and accident analyses, therefore, is not significant

Table 3.3-1

STEAM GENERATOR DESIGN DATA (PER STEAM GENERATOR)[1]

	Original	Refurbished
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, degrees 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, degrees Fahrenheit	516.0	N.C.
Steam Pressure, psig	770	N.C.
Feedwater Temperature at 100% load, degrees Fahrenheit	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 (~ -1.4%)
Tube length (largest U-bend), in.	397.5	N.C.
Total Heat Transfer Surface Area, ft^2	44,430	43,467 (~ -2.2%)
Reactor Coolant Water, Volume, ft^3	945	935 (~ -1.1%)
Reactor Coolant Flow, lb/hr	33.83×10^6	N.C.
Secondary Side Volume, ft^3	4580	4596 (~ 0.3%)
†Secondary Side Mass No Load, lbs	134,000	N.C.
†Secondary Side Mass 100% Power, lbs	76,300	80,300 (~ +5.2%)
Center of Gravity (from the support pads), ft/in.	25/4	N.C.

*No change
†Values are rounded off

Table 3.3-2

COMPARISON OF PARAMETERS FOR ORIGINAL AND REPAIRED STEAM GENERATORS[1]

Primary Pressure Drop	Decreased by 0.7 psi
Fouling Factor	Unchanged
*Nominal Flow Area	Decreased by ∿1.5%
Equivalent Tube Length	Unchanged
Total Heat Transfer Surface Area	Decreased by ∿2.2%
Heat Transfer Coefficient	Increased by ∿2.5%
Nominal Power/SG	Unchanged
Nominal Hot Leg Temperature	Unchanged
Nominal Cold Leg Temparature	Unchanged

*This decrease in flow area is due to the reduction in number of steam generator tubes. Credit has not been taken for the compensating increase in flow area due to the improved manufacturing tolerance on the tube wall thickness.

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and FPL analyses presented in Final Safety Analysis Report[6] (FSAR) remain valid for the repaired steam generators.

In the following sections we have discussed possible changes in the events previously analyzed in the FSAR. The following plant conditions were used in those analyses:

Thermal design flow, gpm/loop	89,500
S. G. tube plugging, %	0
*Power level, Mwt (100%)	2200
*T_{avg} at 100% power, °F	574.2
ΔT at 100% power, °F	55.9
Steady state DNBR	1.63
$F^N_{\Delta H}$	1.75
F_Q maximum	2.55

*The analyses conservatively used 102% power (2244) and T_{avg} +4° (578.2)

It should be noted that for this evaluation the FSAR constitutes the reference cycle. Therefore, if the values of any core physics or plant operating parameters for the reload cycle following the steam generator repair are not bounded by those used in the FSAR, a reevaluation of the affected event(s) will be required prior to operation. Any such reanalyses submitted to the staff should be in accordance with Regulatory Guide 1.70, Revision 3.[12]

It should also be pointed out that the current Emergency Core Cooling System (ECCS) analysis of record for the plant using an approved model is only for the current condition of the original steam generators, i.e., with plugged tubes. If credit for the unplugged configuration of the repaired steam generators is to be taken, a new ECCS analysis using the approved model will be required. A reload report will be submitted for our review and approval prior to startup of the repaired unit if the fuel loading is different than previously reviewed.

3.3.2 Non-LOCA Accidents and Transients

In our evaluation, only the potential effects of the repaired steam generators on the FSAR analyses have been considered. All other parameters are assumed to have their FSAR values. As will be seen, most events are not affected by the slight changes which have been made to a few of the relevant parameters.

For some events, such as Rod Cluster Control Assembly (RCCA) withdrawal and RCCA ejection, there will be no effect due to the repair of the steam generators. The nuclear and thermal time constants of the fuel are much smaller than the fluid mixing and transport time. These events are terminated in less than a loop transport time and, therefore, are unaffected by the steam generators. For the RCCA drop accident and the malpositioning of part length rods (note that removal of these part length rods has been approved by the NRC[7]), the neutron flux redistribution is the limiting consideration. Since this is not dependent on the steam generator performance, these analyses are not affected.

For the loss of reactor flow events, the reactor is rapidly tripped on low frequency, low voltage or low coolant flow. Changes in coolant temperature due to secondary parameter changes would not be detected in the core during the time frame of interest for these events. These analyses are, therefore, also unaffected.

For a chemical and volume control system malfunction, the boron dilution rate depends on the charging pump characteristics and the reactor coolant volume. The small reduction in reactor coolant volume (~1%) from the FSAR value will not significantly change the time available for operator action. Therefore, this minor design change will have a negligible effect on the analysis of this event.

The turbine generator design analysis is not affected by the repair of the steam generators since steam and feedwater conditions are unchanged.

The steam generator repair may affect those events for which the transient reactor coolant conditions result from an interaction of the reactor coolant with the secondary system. These remaining events, which are generally concerned with coolant heatup or cooldown through the secondary side, are discussed in the following sections. For the repaired steam generators, the increase in the heat transfer coefficient (U) offsets the decrease in heat transfer area (A) so that the resulting heat transfer (UA) remains essentially unchanged.

3.3.2.1 Excessive Load Increase

This event involves a rapid increase in steam generator steam flow which causes a power mismatch between the reactor core power and the steam generator load demand. This results in a decrease in reactor coolant temperature and increase in core power. The FSAR analysis shows that a 10% increase in steam flow from full power can be accommodated without reactor trip. The repaired steam generators, which have a higher (~5%) full power fluid inventory, could cause the transient to progress slower. However, the same final steady-state condition will be reached.

3.3.2.2 Startup of an Inactive Reactor Coolant Loop

This event involves the injection of colder water into the core and a significant increase in core flow. This results in a rapid increase in core power. The FSAR analysis assumed that the water in the inactive loop was at the saturation temperature of the secondary side. This is independent of the heat transfer characteristics of the steam generator and will, therefore, be unchanged. The reduction in reactor coolant volume would cause a negligible reduction in the duration of the cold water slug. The delay time for the slug to reach the core will remain unchanged. Therefore, the FSAR analysis of this event would not be significantly affected by the repaired steam generators.

3.3.2.3 Excessive Heat Removal Due to Feedwater System Malfunction

This event involves the addition of excessive feedwater to the steam generator or the inadvertant opening of the feedwater bypass valve. This results in a decrease in reactor coolant temperature and an increase in core power due to moderator feedback. At full power, the FSAR analysis shows that a new steady state condition is reached without reactor trip. Since the repaired steam generators will have a higher full power secondary side mass inventory, the cooldown rate would be slower. However, the same endpoint condition will be reached.

3.3.2.4 Loss of External Electrical Load

A loss of external electrical load event such as a turbine trip results in an increase in reactor coolant temperature and pressure and a decrease in core power. The complete loss of load from 102% power analyzed in the FSAR assumed that there was not a direct reactor trip due to the turbine trip. The increase in full-power inventory of the repaired steam generators would provide additional heat capacity and reduce the heatup rate. Therefore, there are no adverse effects on this event due to the repair of the steam generators.

3.3.2.5 Loss of Normal Feedwater

The loss of normal feedwater results in a loss of capability of the secondary system to remove the heat generated in the core. Since the repaired steam generators will have a higher full-power secondary side mass inventory, additional steam generator heat removal capacity is available. Also, since the dimensions of the steam generators have not changed, the FSAR conclusions that the tubesheet in the steam generators receiving auxiliary feedwater will remain covered and adequate heat transfer capability will be maintained remain valid. Therefore, there are no adverse effects on this event due to the repaired steam generators.

3.3.2.6 Loss of All AC Power to the Station Auxiliaries

The loss of AC power with turbine trip and reactor trip results in a reactor coolant flow coastdown to natural circulation flow rates and an increase in secondary pressure. In the repaired steam generators the tubes will be recessed slightly into the tubesheet holes, thus reducing pressure drop at the entrance to the tubes which will enhance flow. Therefore, the FSAR analysis of this event is conservative for the repaired steam generators.

3.3.2.7 Rupture of a Main Steam Pipe

A steamline break results in a rapid depressurization of the steam generator, a decrease in reactor coolant temperature, and an increase in core reactivity. The FSAR analysis was performed for end of cycle, hot shutdown conditions. This event is unaffected by the repair of the steam generators because the no load fluid inventory of the steam generators which was used in the FSAR is still bounding, and the flow area of the main steam line, the reactivity coefficients and the emergency shutdown system are unchanged.

3.3.3 Loss-of-Coolant Accident (LOCA)

The minor design and operational differences of the repaired steam generator, such as number of tubes, full-power fluid inventory, and pressure drop across the steam generator, are not expected to significantly affect the LOCA analysis. The reduction in flow area and reactor coolant volume due to the lesser number of tubes is approximately equivalent to 1.4% of the tubes in the original steam generator being plugged.

The FSAR ECCS analysis is based on a model which the staff no longer finds acceptable. Therefore, the analysis cannot be used to satisfy the requirements of 10 CFR 50.46. As mentioned above, the ECCS analysis of record, based on the currently approved model, has been performed assuming a significant number of steam generator tubes plugged.

We consider the ECCS analysis of record to be conservative for plant operation with the repaired steam generators. If credit for the unplugged configuration of the steam generators is to be taken, a new LOCA analysis performed with the currently approved model must be submitted.

The repaired steam generators do not have a significant effect on the small-break LOCA. Therefore, the current small-break LOCA analyses are acceptable for the plant with the repaired steam generators.

3.3.4 Steam Generator Tube Rupture

The improved manufacturing tolerance on the tube wall thickness will result in a slight increase in the tube inner diameter. This increase in diameter (0.005 inch) will have a negligible effect on the tube rupture analysis. Therefore, the consequences of this event, as reported in the FSAR, will be unchanged by the steam generator repair.

3.3.5 Summary

The changes in design and plant operational parameters listed in Tables 3.3-1 and 3.3-2 have been evaluated to determine their effect on the safety analyses. We have concluded that the repaired steam generators will not have any significant adverse effect on the transient and accident analyses and, therefore, that the analyses and conclusions presented in the FSAR (except for LOCA) remain valid for the same core physics and plant operating parameters. For the LOCA, new analyses will be submitted as discussed in Section 3.3.1.

3.4 Radiological Consequences of Postulated Accidents

3.4.1 Accidents During Operation with Repaired Steam Generators

The repaired steam generators will not significantly affect the dose consequences of accidents involving the secondary system. The accidents involving significant dose consequences are the main steam line failure, steam generator tube failure and control rod ejection. The only design change that affects the accident dose consequences is a 5% increase in the volume of the secondary side of the steam generator. The reactor coolant system parameters which affect these accidents will not be changed significantly by the repaired steam generators. These parameters include reactor coolant leakage to the secondary system and the reactor cooldown period. The contribution to offsite doses from the secondary system is minor in all three accidents because of low activity levels in the seconary system. The major dose contribution is from reactor coolant leakage into the secondary system during the accidents.

In both the steam generator tube failure and control rod ejection accidents, the increased volume of the secondary system provides for more dilution of the activity which leaks from the reactor coolant site. Because the reactor coolant system parameters have not changed, the total reactor coolant side release time and volume will not change. Therefore, the increased secondary volume should result in a negligible change in doses.

The reactor coolant system parameters which affect the main steam line failure accident also remain unchanged. Assuming the same concentration of radionuclides (pre-existing inleakage of reactor coolant), the increased mass of the secondary side will result in a slight increase in offsite doses. The contribution to the doses from additional reactor coolant inleakage during the accident itself would be unchanged. Because the secondary volume increases by 5% and most of the dose is a result of "fresh" reactor coolant inleakage, the total offsite dose will increase by much less than 5%. This slight increase in total offsite dose will not reult in estimated consequences in excess of the 10 CFR Part 100 guidelines, and the conclusions concerning these accidents reached in the March 15, 1972 Safety Evaluation for the Turkey Point Plant[7] are not changed due to the repair of the steam generators.

3.4.2 Accidents During the Repair Effort

FPL has analyzed the potential consequences of postulated accidents associated with the repair effort. FPL has analyzed the potential for steam generator crane rigging accidents which may affect the refueling water storage tank and primary water storage tank and concluded that rigging operations will be conducted in areas sufficiently removed from these tanks to preclude damage to these structures.

FPL has analyzed the potential consequences of postulated accidents associated with the repair effort. FPL has analyzed the potential for steam generator crane rigging accidents which may affect the refueling water storage tank and primary water storage tank and concluded that rigging operations will be conducted in areas sufficiently removed from these tanks to preclude damage to these structures. FPL has also evaluated the potential for a steam generator being dislodged from the rigging and striking the radwaste or fuel handling building. FPL has concluded that both buildings are capable of withstanding all postulated impacts with no breach of integrity. We have evaluated the FPL report[1] and concur with the above conclusions. Therefore, we conclude that there will be no radioactive release to the environment from these construction-related accidents.

The staff has evaluated the potential consequences of lifting accidents with the lower assembly both inside and outside of containment and accidents during the initial decontamination of the channel head prior to cutting.

FPL will be installing two safety features for the repair effort: a special 18000 CFM vent system which exhausts through a HEPA filter and a covering for the equipment hatch which will reduce its opening to less than 18 square feet. With the existing purge system shut down (it vents through a roughing filter) but with the special vent system operating and the covering in place, the air flow within the containment building will be filtered through the HEPA filter before it reaches the environment. FPL procedures will provide for attaining this configuration quickly in case of a radiological accident. A few days after shutdown, based on experimental measurements of the composition and amount of the crud and extrapolation to the nine years of operating time for Unit 3 at its repair date, a steam generator is calculated to contain 380 curies of activity.

FPL has calculated that 45 curies of activity is to be removed from the channel head during the decontamination process. Although this process will take place over a few shifts and therefore only a fraction of the expected amount to be removed will be in the system at one time, total release of 45 curies was assumed in the staff evaluation of the decontamination accident. The limiting potential receptor from consideration of both breathing rate and dose factors was assumed; that is, the teenager's lung. The teenager was assumed to remain at the exclusion area boundary at the side of the plant closer to the population; that is, the north sector. The accident atmospheric diffusion factor for this sector is 1.4×10^{-4} sec/m^3. Dose conversion factors for the experimentally determined distribution of nuclides in the crud were obtained from Regulatory Guide 1.109. The release was assumed to take place through the HEPA filter with only a 99% efficiency. The conservatively evaluated consequence for the decontamination accident is, then, 8 mrem.

After initial decontamination of the channel head, the steam generator will be cut below the tube sheet. This cutting operation will be performed within an enclosure with a HEPA filter exhaust system. FPL has estimated that 0.035 curies of activity would be released through the HEPA filter of the enclosure due to total vaporization of the material in the vicinity of the cut. The output of the HEPA filter is ducted directly to the purge system. Therefore, if the filter, which is located at the special enclosure, were damaged, all the activity from the cutting operation would be emitted from the plant. The staff has conservatively assumed that 3.5 curies of activity is released.

The impact at the exclusion area boundary, using the same conservative potential receptor, meteorology, and dose conversion factors as for the decontamination accident, would be 63 mrem. No credit has been taken for the ability to stop cutting. The cut at the top of the assembly to remove the moisture separators is within the secondary side and involves little activity.

After cutting, the lower assembly will be lifted to the operating floor where plates will be welded over the cut ends. The lift itself will be made with the primary side covered only by a herculite cover; lack of room in the vicinity of the steam generator precludes bringing in a large metal plate prior to the lift. FPL has procedures which are designed to assure that the crane and rigging to be used in the lift are tested and are adequately conservative in design basis load. Although the likelihood of a drop accident is extremely small, the staff has evaluated the potential offsite consequences of such an accident using conservative assumptions of the amount of activity that could be released. Twenty percent of the content of the steam generator was assumed to be loose enough to be released by the mechanical shock of the drop. This value is in accord with the assumptions made in other evaluations of similar conditions. Even if this large amount of activity were to be released from the steam generator and any primary piping upon which it might be dropped, it is unlikely that it could be carried to the special vent. The particles that might be removed because of the mechanical shock have a range of sizes with different propensities to remain airborne or to be resuspended and the reduced airflow will minimize the turbulence that might transport or resuspend the radioactive crud. For the same potential receptor, meteorology and dose conversion factors as for the decontamination accident, the conservatively evaluated consequence of the drop accident is 14 mrem.

The steam generator lower assembly, after the end plates are welded in place, will be lowered to a carrier from the level of the equipment hatch. Further, if the steam generators are prepared for shipment offsite, a lift is required for loading. FPL has concluded that the welded plate over the primary side might be breached if the steam generator were dropped. Again in this case, as for the drop inside containment, the likelihood of this accident is extremely small. Since it is even more unlikely that the weld might completely fail and because the drop distance is smaller, it was assumed that only half of the "loose" crud was released from the lower assembly. The lift was assumed to take place near the laydown area, about 300 meters from the receptor location on the shoreline. The χ/Q for this distance for stable conditions is 3.5×10^{-3}. Using 33.5 curies as the amount of activity released and assuming the same potential receptor and dose conversion factors as for the decontamination accident, the radiological consequence of the drop accident outside containment is 15 rem. While this value is highly conservative for several reasons, it emphasizes the importance of a high quality weld and adequate rigging to prevent a drop.

We conclude that the consequences of releases from all accidents except the drop outside containment are well within the limits of 10 CFR Part 20 for normal operation and are acceptable. With special attention to procedures for rigging and lifting, the risk of release of radioactive material during a drop accident may be made acceptably small (see Section 3.2.6).

3.5 Security

FPL has an approved Modified Security Plan[14] which will be implemented during the repair program to assure that the security program in effect at the Turkey Point Plant is not degraded as a result of steam generator repair program activities. We have reviewed the FPL program in light of these measures and have concluded that the program will not be degraded.

3.6 Special License Conditions

During the repair program the following temporary license conditions will be imposed:

(1) All fuel shall be removed from the reactor pressure vessel of the unit under repair and stored in the spent fuel pool.

(2) The health physics program and procedures which have been established for the steam generator repair program shall be implemented.

(3) Progress reports shall be provided at 60-day intervals from the start of the repair program and due 30 days after close of the interval with a final report provided within 60 days after completion of the repair. These reports will include:

(i) A summary of the occupation exposure expended to date using the format and detail of Table 3.3-2 of the "Steam Generator Repair Report" as supplemented.

(ii) An evaluation of the effectiveness of dose reduction techniques as specified in Section 3.3.5 of the "Steam Generator Repair Report" as supplemented in reducing occupational exposures.

(iii) An estimate of radioactivity released in both liquid and gaseous effluents.

(iv) An estimate of the solid radioactive waste generated during the repair effort including volume and radioactive content.

(4) Procedures shall be prepared to assure that power can be restored by manual operator actions to the fuel pool of the unit undergoing repair within eight hours (3.2.2.A).

(5) The remedy chosen by FPL to provide the availability of the diesel fuel supply while the oil-retention dike is removed from the main diesel safety tank shall be addressed and adequately demonstrated by FPL prior to initiating the construction changes affecting the dike (3.2.2.B).

(6) Sixty days prior to fuel loading, the program for preoperational testing and startup shall be submitted for NRC review (2.7).

(7) Sixty days prior to fuel loading, FPL should submit for evaluation by the NRC a steam generator secondary water chemistry control and monitoring program (3.2.4) which will address the following:

(a) Identification of a sampling schedule for the critical parameters and of control points for these parameters for each mode of operation: normal operation, hot startup, cold startup, hot shutdown, cold wet layup;

(b) Identification of the procedures used to measure the values of the critical parameters;

(c) Identification of process sampling points;

(d) Procedure for the recording and management of data;

(e) Procedures defining corrective actions* for off-control point chemistry conditions; and

(f) A procedure identifying (a) the authority responsible for the interpretation of the data and (b) the sequence and timing of adminstrative events required to initiate corrective action.

FPL should verify that the steam generator secondary water chemistry control program incorporates technical recommendations of the NSSS vendor. Any significant deviations from NSSS vendor recommendations should be noted and justified technically.

(8) Sixty days prior to the decontamination of the channel head, FPL should meet the following conditions (3.2.5):

(a) A system should be set up so that the pressure in the inflatable plug seal in the RCS pipe nozzles should be monitored. Upon loss of seal pressure, injection of the grit slurry should be stopped immediately and the seal plug replaced.

(b) Written procedures should be provided to include accountability controls of all tools, equipment, materials, and supplies that are to be used in the channel heads to prevent inadvertent entry of such items into the reactor primary coolant system. These controls should be in effect whenever the inflatable plug seals and their associated cover plates are not in place in the nozzles of the reactor coolant system piping.

(c) Written procedures should be provided to restrict materials to be used in the channel head area to prevent the presence of materials having potential adverse effects on the reactor coolant system components (for example, chloride-bearing materials).

*Branch Technical Position MTEB 5-3 describes the acceptable means for monitoring secondary side water chemistry in PWR steam generators, including corrective actions for off-control point chemistry conditions. However, the staff is amenable to alternatives, particularly to Branch Technical Position B.3.b(9) of MTEB 5-3 (96-hour time limit to repair or plug confirmed condenser tube leaks).

(d) Written procedures should be provided to include instructions to require that the channel head area, including the nozzles, be inspected and confirmed to be free of all loose materials, equipment, and tools prior to removing the cover plate from the inflatable plug seal.

(e) Prior to closing up the reactor coolant system and starting the RCS pumps, any loose debris, including the abrasive grits, in the channel head, RCS hot leg, and cold leg should be cleaned up.

(f) Prior to resumption of power operation, the licensee should submit for NRC review and acceptance a report which will include an analysis of the possible effects of any foreign material which has entered the primary coolant system and has not been retrieved. The report should include all work on the decontamination and steam generator repair.

(9) Sixty days prior to the movement of the used steam generator lower assemblies from the containment, the procedures for the move, associated QA requirements, and a description of the equipment to be used shall be provided to the NRC (3.2.6).

(10) Before storage or shipment of the used steam generator lower assemblies, the seal welds must be coated with a heavy body varnish such as glyptol (3.2.6).

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the conduct of the proposed action, and (2) such activities will be conducted in compliance with the Commisssion's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date:

5.0 REFERENCES

1. "Steam Generator Repair Report - Turkey Point Units 3 and 4," Florida Power and Light Company, September 20, 1977, as supplemented on December 20, 1977, March 7, April 25, June 20, August 4, and December 15, 1978, January 26, 1979, and March 20, 1980.

2. Steam Generator Repair Program, Surry Power Station Unit Nos. 1 and 2, Virginia Electric and Power Co., August 17, 1977, and revisions dated December 2, 1977, June 2, June 13, June 30, September 1, October 25, and November 10, 1978.

3. Safety Evaluation Report by the Office of Nuclear Reactor Regulation, License Nos. DPR-32 and DPR-37, Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Docket Nos. 50-280 and 50-281, U.S. Nuclear Regulation Commission, December 15, 1978.

4. Environmental Impact Appraisal by the Office of Nuclear Reactor Regulation, License Nos. DPR-32 and DPR-37, Virginia Electric and Power Co., Surry Power Stations, Units 1 and 2, Docket Nos. 50-280, 50-281, Nuclear Regulatory Commission, January 20, 1979. A Draft Environmental Statement (NUREG-0663) was issued in March 1980 per Commission Order dated March 4, 1980, and a Final Environmental Statement (NUREG-0692)*was issued in July 1980.

5. Hoenes, G. R., Waite, D. A. and McCormack, "Radiological Assessment of Steam Generator Removal and Replacement," NUREG/CR-0199, PNL-2656 Battelle Pacific Northwest Laboratories for U.S. Nuclear Regulatory Commission, September 1978. Revised as NUREG/CR-1595, December 1980.*

6. Final Safety Analysis Report, Turkey Point Plant, Unit Nos. 3 and 4, Florida Power and Light Company, May 12, 1966 (as supplemented).

7. Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission In the Matter of Florida Power and Light Co., Turkey Point Plant Unit Nos. 3 and 4, Dade County, Florida, Docket Nos. 50-250 and 50-251, March 15, 1972.

8. "FPL Quality Assurance Topical Report," FPLTQAR 1-76A, Revision 2, Florida Power and Light Company, September 8, 1977.

9. "Bechtel Quality Assurance Program for Nuclear Plants," Topical Report BQ-TOP-1, Revision 2A, October 15, 1979.

10. "Quality Assurance Plan - Westinghouse Water Reactor Division," Topical Report SCAP-8370, Revision 8A," Approved by letter from NRC (Heltemes) to Westinghouse (Eicheldinger), September 10, 1977.

11. Letter from NRC (Schwencer) to FPL (Uhrig) transmitting Amendment Nos. 45 and 37 to License Nos. DPR-31 and DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4 with attached NRC Safety Evaluation for the FPL Fire Protection Program, March 21, 1979.

12. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Office of Standards Development, U.S. Nuclear Regulatory Commission, Regulatory Guide 1.70, Revision 3, November 1978.

13. Letters from NRC (Schwencer and Varga) to Florida Power and Light Co. transmitting our evaluation of design details and a summary of staff positions, dated April 4, July 9, and October 9, 1980.

14. Letter from NRC (Schwencer) to Florida Power and Light Co. (Uhrig) transmitting Amendment Nos. 44 and 36 to License Nos. DPR-31 and DPR-41, approving the Modified Amended Security Plan, dated February 27, 1979.

15. Licensee's Statement Concerning Intervenor's August 30, 1979 contentions, In the Matter of Florida Power and Light Co., Turkey Point Plant Unit Nos. 3 and 4, Docket Nos. 50-250, 50-251, September 14, 1979 (p. 6).

16. Draft Environmental Statement related to Steam Generator Repair at the Turkey Point Plant, Units Nos. 3 and 4, Docket Nos. 50-250 and 50-251, Florida Power and Light Company, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, NUREG-0743, December 1980.**

17. Letter from NRC (Schwencer) to Florida Power and Light Company (Uhrig) transmitting Amendment Nos. 37 and 30 to License Nos. DPR-31 and DPR-41, approving the removal of the part-length rods, dated September 21, 1978.

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

**Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER *(Assigned by DDC)* NUREG-0756
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16. ABSTRACT *(200 words or less)*

A Safety Evaluation Report for the Turkey Point Plant, Units 3 and 4, has been prepared by the Office of Nuclear Reactor Regulation. This report considers the safety aspects of the proposed steam generator repair at Turkey Point Plant, Units 3 and 4. The report focuses on the occupational radiation exposure associated with the proposed repair program. It concludes that there is reasonable assurance that the health and safety of the public will not be endangered by the conduct of the proposed action and such activities will be conducted in compliance with the commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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