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SAFETY EVALUATIC 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

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1.0 INTRODUCTION

By letter dated August 17, 1977, Virginia Electric & Power Company (the licensee) submitted a report titled "Steam Generator Repair Program, Surry Power Station Unit Nos. 1 and 2." This report was revised December 2, 1977, April 21, June 2, June 13, June 30, September 1, October 25, and November 10, 1978. We determined that the proposed program requires our review, approval and issuance of license amendments. Our evaluation of this program is presented in this report. A Notice of Proposed Issuance was published on October 27, 1977 (42 Fed. Reg. 56652). The steam generator repair program was reviewed by the ACRS Surry Sub-Committee on October 28, 1978.

VEPCO plans to replace all three steam generators in Unit 2 during the period from January through June 1979. Unit 1 is expected to be operating during this period of time. All of the Unit 1 steam generators are scheduled to be replaced in October 1979 through April 1980, after Unit 2 returns to power.

Questions raised by the Commonwealth of Virginia, Office of Attorney General, letter dated January 17, 1978 are discussed in Appendix A to this SER.

1.1 History of Steam Generator Operation

Surry Units 1 and 2 began commercial operation on December 22, 1972, and May 1, 1973, respectively. Like almost all units with U-tube design steam generators, they initially used a sodium phosphate secondary water chemistry treatment. This treatment was designed to remove precipitated and suspended solids by blowdown and was successful as a scale inhibitor. However, during early use many PWR U-tubed steam generators with Inconel-600 tubing started experiencing stress corrosion cracking. The cracking was attributed to free caustic which can be formed when the Na/PO_4 ratio exceeds 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 being adequately removed by the blowdown. The reaction products of these impurities and of the 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 become concentrated by evaporative processes and precipitated. This phosphate precipitation (hideout) caused localized wastage resulting in thinning of tube walls. The problem of stress corrosion cracking was corrected by maintaining the Na/PO_4 ratio below 2.6. However, this did not correct the phosphate hideout problem or the wastage of the Inconel-600 which increases as the sodium/phosphate ratio is lowered. Therefore, most PWRs with a U-tube design steam generator have discontinued the phosphate treatment and have now converted to an all volatile chemistry treatment (AVT). Surry 1 and 2 have been on AVT since about January 1975.

In 1975 circumferential indentation (denting) was observed in tubes of the steam generator at several PWR facilities including Surry 1 and 2. This denting was observed after 4 to 14 months of operation, following the conversion to AVT. 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 near the center of the tube bundle and at the peripheral locations where the plate is wedged to the wrapper and shell. The hard spots 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 plates 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 the water circulation holes. As a result of these forces on the tube support plate, several of the rectangular flow slots have also exhibited a phenomenon referred to as "hourglassing", i.e., the side walls of these flow slots have moved closer so that the centers of some of these slots have even closed.

On September 15, 1976, during normal operation, one U-tube in the inner-most row parallel to the rectangular flow slots in steam generator A at Surry Unit No. 2 rapidly developed a substantial reactor coolant to secondary leak (about 80 gpm). The tube causing the leak was removed for laboratory analysis. It was established that the leak resulted from an axial crack, approximately 4-1/4 inches in length at the apex of the U-bend. The crack was caused by intergranular stress corrosion initiated from the reactor coolant side. Since the initially parallel side walls of the flow slots in the top support plate had moved closer, the adjacent support plate material had also moved inward. This, in turn, forced an inward displacement of the legs of the U-bends at these locations. This inward movement of the legs of the U-bends increased the hoop 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 reactor coolant 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, tighter operating restrictions and more limited periods of operation between inspections have also been imposed on the more severely degraded units (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 Repair

All of the Surry Units 1 and 2 steam generators have undergone significant degradation. The wastage and denting phenomena above have led to tube wall thinning, support plate flow slot hourglassing, plate ligament cracking, tube denting, stress corrosion cracking, and several instances of reactor coolant to secondary leakage through cracked tubes. As of September 1978, tube plugging for various reasons has resulted in removing 21.4% of the steam generator tubes in Unit 1 and 21.5% of the tubes in Unit 2 from service.

Due to the continuing denting related problems (requiring plant shutdown and occupational radiation exposure), the certainty that additional tube plugging can result in power derating, and the economic considerations for operating with substantially reduced heat transfer capacities on the two Units, the licensee has proposed to replace the degraded portions of these steam generators.

2.0 DESCRIPTION OF REPAIRED STEAM GENERATORS

2.1 Mechanical Design and Materials Changes

During 1975 several modifications were made to the existing 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. Also additional modifications, as discussed below, will be incorporated into the repaired steam generators.

A flow distribution baffle plate, located 18" above the tubesheet, will be used in the repaired generators. The baffle plate is designed to assist and direct the 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 in the repaired steam generators. The new system will have increased blowdown capacity through two 2-inch Schedule 40 Inconel internal blowdown pipes. The blowdown intake location is coordinated with the baffle plate design so that the maximum intake (flow) is located where the greatest amount of sludge is expected to deposit.

Unlike the existing design, all tubes in the repaired generators will be expanded to the full depth of the tube-sheet to eliminate the potential for contaminant concentration sites at these interfaces.

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 expected to be much more corrosion resistant than the carbon steel now in use. Furthermore, 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. Thus its corrosion product is not expected to exert the significant stresses observed with present design.

As another important design change, the tube support plates in the repaired steam generators will have "quatrefoil" design holes which will both support the tubes and provide for secondary water flow. In the quatrefoil design, the separate flow holes have been eliminated. In their place material has been removed from the tube holes in four places creating four flow lobes and leaving four support lands. These support the tube while allowing water flow around it. This design has a lower pressure drop across the thickness of the plate than the existing design and results in higher average flow velocities along the tube surfaces at these elevations. This should prevent most sludge depositions and, by eliminating a continuous narrow gap (tube support plate annulus), eliminate the denting phenomenon.

The tubes in the repaired generators will be recessed slightly into the tubesheet holes and then welded to the tube-sheet cladding. This design is expected to reduce entry pressure losses and eliminate locations for possible crud buildup on the reactor coolant side.

Since the secondary coolant 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.

To improve access for the inspection of the tubesheet and flow distribution baffle, and to assist in sludge lancing, the new lower shell assemblies will have additional access ports. Also, 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. The repaired steam generators will also have closure rings welded inside the channel head at the base of each reactor coolant nozzle so that closure plates (blind flanges) can be bolted in place during reactor coolant side maintenance.

2.2 Heat Treatment of Tubing

The Inconel-600 tubing used in the repaired steam generators will be thermally treated to produce a microstructure with improved resistance to stress corrosion cracking by 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 in sulphur-containing species.

2.3 ASME Code and 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 end preps, welding, and nondestructive examination will be in accordance with the applicable sections of the latest edition of the ASME Code. Also, the provisions of applicable Regulatory Guides will be met. The applicable Regulatory Guides are identified on page 9.C.5-1 of the licensee's report.

2.4 Removal and Reinstallation

The steam generator repair will consist of replacing the lower assembly of each steam generator including the shell and tube bundle. The steam separation equipment in the upper assembly will be refurbished and partially replaced. The old lower assemblies 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 Surry site. The new lower assemblies will arrive at the site by barge. They will be transferred to a wheeled transporter and hauled approximately 1.5 miles on the existing road along the intake canal to the containment building equipment hatch.

Prior to the repair work, the affected Unit will be shut down and defueled after seven days. The reactor vessel head will be replaced and all of its systems will be placed in condition for long term layup. The equipment hatch will be opened and access control will be established. A protective cover will then be placed over the reactor vessel and local decontamination work will begin. The biological shield wall for all three steam generators and a section of the pressurizer cubicle wall will be removed to provide access to the steam generators. A temporary ventilation and air filtration system as well as local barriers such as tents and ducting will be installed to minimize dust and the spread of contamination. Rails will be installed for transporting the lower steam generator assembly through the equipment hatch.

After this preparatory work, the cutting of piping will begin. This will include cutting and removal of sections of steam lines, feedwater lines, reactor coolant inlet and outlet lines, and miscellaneous smaller lines for the blowdown system and the instrumentation system. The steam generator will then be cut at the transition cone and the upper portion of the shell will be removed, inverted and placed on the operating deck. Special covering devices will be used to seal the openings on the steam generators to minimize the spread of radioactive contamination. The steam generator supports will then be disassembled and the steam generator lower assembly will be lifted by the polar crane. This assembly will be lowered and 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. This process will be repeated for the other two steam generators.

After removal and storage of all three steam generator lower assemblies, their replacements will be transported from the barge dock or temporary storage location to the equipment hatch. During this time, the upper assemblies will be refurbished by installing new moisture separation equipment, feedrings and other internals. The same machinery used to remove the old lower assemblies will be used to install the new assemblies in their cubicles. The steam generator support system 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 pressurizer cubicle wall will be reconstructed. The shield and wall have no structural function.

Following these major repair activities there will be cleaning, hydrostatic testing, baseline inservice inspections, and pre-operational 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.

See paragraph 2.6 for a discussion of the measures to be taken to keep occupational exposures as low as reasonably achievable during the removal and installation of the steam generators.

2.5 Post Installation Testing

A detailed testing program will be carried out prior to reloading any fuel. This program is 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 the affected piping. The fuel manipulator crane will be reassembled and tested before reuse.

After the residual heat removal system has been tested and placed in service, fuel will be loaded into the reactor vessel. One third of the core will be new fuel assemblies. The balance will be irradiated fuel previously removed from the core. No new fuel handling procedures will be required for the core reload.

During the initial startup of the Unit, tests will be performed to verify the thermal and hydraulic performance of the nuclear steam supply system including a test of moisture carryover from the steam generators.

We have reviewed the licensee's criteria for the program for preoperational testing and startup after completion of the steam generator repairs and find them acceptable. Prior to fuel loading we will review the licensee's program to verify that adequate testing will be performed to ensure safe startup of the Unit after completion of these repairs.

2.6 Radiological Considerations

A major aspect of the repair effort is its radiological impact, including the occupational exposure accumulated during the repair effort and the radiological effluents released from the site. These considerations are discussed below.

Battelle-Pacific Northwest Laboratories (PNL) has performed a generic radiological assessment of steam generator replacement and disposal, which has been published in a separate NRC report, NUREG/CR-0199, "Radiological Assessment of Steam Generator Removal and Replacement". The PNL estimates of occupational exposures (man-rems) were derived by multiplying maintenance activity man-hours by exposure rates (R/hr) for these activities.

Maintenance activities were developed by PNL as a composite of the work descriptions for removal and replacement of the steam generators at Surry and Turkey Point as determined by VEPCO and FP&L.

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

Exposure rates were based on information from several sources including data from measurements made at several operating PWRs including the Surry Units. PNL usually selected exposure rate values on the high end of the range of values measured at the several plants. The PNL estimates of occupational exposures are intended to be conservative and represent upper bound values. The PNL estimates are presented as a range of values. The PNL upper value was estimated assuming no credit for dose saving techniques. The PNL lower value was estimated assuming credit for shielding by raising the steam-generator water level, remote tooling and distance where applicable. It is the PNL lower value which is used to compare with the licensee's estimates. The licensee's estimates are generally lower than PNL's because the licensee used actual plant data and took credit for temporary shielding (such as lead blankets) and local decontamination in addition to the measures taken by PNL. We have concluded that, based on the above factors, the licensee's estimates should be more representative of the actual doses.

PNL also provides upper bound estimates of radioactive effluents which could be released as a result of the replacement effort. The estimates given in this report are on a per unit basis (i.e., repair of 3 steam generators) unless otherwise noted.

2.6.1 Occupational Radiation Exposure

Removal and installation of the repaired steam generators, separation and disassembly, must be done in significant radiation fields. Federal regulations, as specified in 10 CFR Part 20.1(c), state that licensees should make "every reasonable effort to maintain radiation exposures...as low as is reasonably achievable" (ALARA). The licensee's 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 removal, replacement and return to power have been incorporated into the dose estimates. These include health physics and quality assurance/quality control activities.

2.6.1.1 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. Since the actual fuel transfer time is only a fraction of the refueling operation compared to preparation and buttonup activities, the total defueling time for a full core is not expected to be significantly greater than a normal refueling of 1/3 of a core. The radiation field will be the same as during 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, the reactor coolant system will be partially drained. Temporary structures will then be installed to facilitate the steam generator separation and removal activities. These structures include a reactor vessel cavity cover which will provide a contiguous work area on the operating floor, contamination control envelopes around the drained reactor coolant piping at the separation points, temporary ventilation systems, scaffolding lighting and temporary shielding.

The preparation activities also include radiation surveys and local decontamination. Portions of the biological shield wall will then be removed, prior to cutting the reactor coolant piping to permit later removal of the steam generator lower assemblies.

The polar crane will be inspected and tested and the steam generator transport systems inside (and outside of) containment will be constructed.

In order to reduce occupational exposures many of the activities will be performed with the steam generator secondary side partially filled with water to lower radiation fields. The licensee has estimated a total dose of 599 man-rem per Unit for these post-shutdown preparation activities. The major portion of this dose estimate is attributed to dose reduction efforts such as installation of temporary shielding, containment cleanup and local decontamination.

PNL (NUREG/CR-0199) has estimated an occupational dose of 450 man-rem for the post-shutdown preparation activities. The PNL estimate assumes control of the steam generator secondary side water level to shield radiation emanating from the primary side corrosion products. The licensee's estimate for this phase is higher than the PNL estimate because the licensee has estimated that approximately 12,600 man hours will be expended on dose reduction techniques such as installation of temporary shielding and local cleanup and decontamination which will result in 405 man-rems of exposure. PNL has estimated 720 man-hours for installation of shielding and local decontamination resulting in 48 man-rem.

The PNL man-hour estimate is lower than VEPCO's because of the difficulty in providing a generic estimate of an activity which is plant specific in nature. The licensee's estimate is based on it's knowledge of plant specific design and should be more representative of that actually spent. Although VEPCO's estimate is higher, the extra exposure spent for shielding and decontamination will be recovered in dose savings in the removal and installation phases. PNL has not taken credit for dose savings from temporary shielding and local decontamination in subsequent repair activities.

2.6.1.2 Steam Generator Removal

Removal activities include removal of the thermal insulation around the steam generators and pipe separation areas and around the reactor coolant and secondary system piping. Main steam lines, feedwater, reactor coolant inlet and outlet and miscellaneous pipe segments must all be removed to provide clearances in the steam generator area. The highest exposures will most likely occur during cutting of the reactor coolant piping because of the manhours required in the radiation area to complete the cutting. These cuts will be performed in a contamination control envelope with a ventilation system containing a HEPA filter to minimize the spread of airborne particulates. A plasma arc cutting device will make these cuts to minimize the total personnel stay time in the radiation fields near this piping. In addition, shielding of adjacent high radiation sources such as the reactor coolant pumps and valves will be used to reduce the radiation fields where personnel must be present. Mockups will be used to familarize skilled personnel in the specifics of the cutting operations including space restraints, protective clothing, and special tasks required. The familiarization training should minimize the time required to perform the operations and thus, minimize time spent in radiation fields. The cut reactor coolant pipe ends will be covered with shields to reduce radiation streaming from the internal surfaces.

The steam generator upper shell will be cut and removed from the lower assembly and stored in the containment. Due to the low radiation fields at this location, minimal shielding will be required and flame cutting techniques will be used. 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 activitiy. 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 from radiation emanating from the lower shell internals. Flame cutting techniques will be used to cut the wrapper to minimize the time. The PNL dose estimate for cutting the wrapper assumed the cut would be performed from inside the steam generator upper shell in much higher radiation fields and takes no credit for shielding from keeping the water level high. The licensee's estimate of occupational exposure to cut the wrapper is based on lower radiation fields.

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. The upper shell moisture separation equipment will be replaced with new equipment except for the demisters which will be reused. The upper shell will be refurbished at low radiation level work locations inside containment and prepared for reinstallation on the new steam generator lower assembly. The contribution to the occupational exposures will be minimal due to the low contamination levels expected on secondary side portions of the steam generator and the ambient radiation levels at the work areas.

All three existing generators will be removed before any of the new generator sections are brought into the containment. The licensee has estimated an expected maximum total occupational exposure of 560 man-rem per Unit for the removal activities. PNL (NUREG/CR-0199) has estimated a dose of 1100 man-rem for the removal phase. The licensee's lower estimate is based on actual plant data and includes dose reductions from temporary shielding and local decontamination.

2.6.1.3 Installation of Repaired Steam Generators

The installation phase involves bringing in and installing the new lower shell assemblies, installing the moisture separation equipment, bringing in and attaching the upper shells, transporting and reinstalling all the removed piping and associated transition pieces, reconstructing the concrete walls removed earlier, removing all temporary work structures, cleanup, performing preoperational structural integrity tests, refueling and preparing the containment for startup tests prior to return to power. 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 reactor coolant system piping. To minimize radiation exposure, an automatic welding device will be used. PNL (NUREG/CR-0199) has estimated a savings of 500 man-rem per generator (1500 man-rem per Unit) from using remote welding as compared to manual welding. The licensee has estimated the maximum expected exposure for this phase to be 877 man-rem per unit of which 68 man-rem is due to reinstallation of the reactor coolant system piping. PNL (NUREG/CR-0199) has estimated an exposure of 1800 man-rem of which 1500 man-rem is due to reinstallation of the reactor coolant system piping. VEPCO has performed a more detailed estimate of the installation phase including such items as removal of extra temporary shielding and scaffolding, containment cleanup and painting. Consequently, the VEPCO dose estimate for this phase is higher than the PNL estimates for this phase. The PNL dose estimates did not include as much temporary shielding or consider some of the specific tasks considered by VEPCO.

2.6.1.4 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 2.6.5. The licensee has estimated a maximum of 35 man-rem per Unit will be expended for the onsite storage. PNL (NUREG/CR-0199) has estimated 30 man-rem per Unit. These estimates are essentially the same.

2.6.1.5 ALARA Considerations

The licensee has estimated a maximum of 2070 man-rem per Unit will be expended for the repair program. This estimate is based on dose rate survey data from the Surry reactors, estimates of man hours involved for the individual procedures and estimated savings from dose rate reduction techniques. PNL (NUREG/CR-0199) has estimated a total dose of 3380 man-rem per Unit for the whole repair program. The licensee's lower estimate is based on actual plant data and include dose reductions from temporary shielding and local decontamination as well as the remote tooling and control of steam generator water level assumed by PNL.

Extensive planning will be used in the repair effort, including the health physics aspects. An individual knowledgeable in health physics has been assigned full time to the repair effort and will be responsible for all radiation protection activities. He will participate in the planning phase and will supervise the health physics program during the repair program. A health physics manual written for the repair effort will be used. This health physics program will be required to be implemented to insure that exposure to occupational workers is ALARA. The repair effort will be performed using a "work package" method which will include all information necessary to complete a particular job. Dose rate reduction information such as shielding requirements will also be included. The shielding requirements will be based on radiation surveys taken after shutdown as part of the post-shutdown phase of the repair effort. Pre-operational briefings will be held to assure familiarity with the repair effort. Information gained during the effort will be factored into the work packages as a result of debriefing sessions.

Access to radiation areas will be controlled by use of a limited access path through the equipment hatch and use of radiation work permits. The entire repair effort will be continuously monitored by health physics personnel. Area monitors will be in service during the repair, and will provide warnings of high airborne radiation levels.

Employee training will be used throughout the repair program. Training activities will include health physics training with emphasis on biological effects of radiation. All contract personnel will be required to follow the licensee's health physics program. Training aids and mockups will be used to familiarize workers with tasks in order to reduce the time spent in radiation areas. These mockups include a full scale mockup to simulate welding of the steam generator upper shell and new lower assembly. In addition, more experienced personnel will be used whenever possible to maximize efficiency of an operation and thus minimize the total exposure time.

Decontamination can be an effective dose reduction technique because radiation fields can be significantly reduced. However, several factors must be considered where decontamination is being considered. Chemical compatibility of the decontamination fluid with the materials of the installed system must be proven. Additional exposure would result from installation and operation of decontamination equipment and processing of the radioactive waste generated. Based on present limited experience in large scale, high volume chemical decontamination of reactor coolant systems, we believe that considerable economic impact, e.g., increased reactor outage time and development of equipment and procedures, would result from the use of chemical decontamination. Also, the research necessary to prove the safety of such operations could have a major schedule impact. Because of these considerations, we conclude that chemical decontamination of the tubes is not a viable option for this program at this time. Local work area surfaces however, can and will be decontaminated using mild solutions. This should provide worthwhile radiation exposure reductions for several of these areas. The licensee will use such local decontamination wherever dose reduction benefit can be gained.

We have reviewed the licensee's submittal regarding occupational exposures and conclude that efforts being made to maintain occupational exposures ALARA are acceptable because the licensee is doing everything reasonable to reduce occupational exposure.

2.6.2 Radioactive Waste Treatment

Radioactive waste treatment will be used to provide treatment of activity generated as a result of the repair effort so that radioactive releases to the environment are kept to a minimum. The currently installed station waste treatment systems and temporary systems as discussed below will be used to process airborne and liquid wastes.

2.6.3 Airborne Radioactive Releases

The Unit will be shutdown and the core unloaded; therefore, no gaseous wastes will be generated from reactor operations during the repair period which is expected to last about six months. However, some airborne radioactivity will be generated as a result of the fuel unloading. This is expected to be similar to the activity associated with a normal refueling. The potentially significant source of airborne radioactivity generation associated with the repair program will come from activities such as concrete removal and cutting and weld preparation work on open radioactive coolant piping. The major source of radioactivity is expected to be particulates generated from cutting the reactor coolant system (RCS) piping. These cuts will be performed in a local contamination control envelope which is ventilated to the containment through a local high efficiency particulate air (HEPA) filter. The secondary system piping cuts and concrete removal will not require local contamination control envelopes because of the low contamination levels in the secondary side piping and on the concrete. To assure that airborne radioactive releases to the environment are kept to a minimum, all containment releases will be processed through a temporary ventilation system containing a HEPA filter. There will be a slight negative pressure on the containment to prevent release through the access hatches.

The licensee has estimated that a maximum of 6×10^{-4} Ci of particulate activity per Unit will be released to the environment as a result of the RCS piping cuts via filtered ventilation systems. Based on expected contamination levels on the reactor coolant side surfaces and expected kerfs, it was estimated that the cuts release 0.25 Ci per Unit to the contamination control envelope. This activity will pass through the local HEPA filters to the containment atmosphere and then through the containment ventilation system HEPA filters to the environment. Although the HEPA filters will be purchased to a removal efficienty of 99.97%, a filter efficiency of 95% was assumed for each set of filters in series. We have independently estimated 0.33 Ci may be generated locally by cutting of the RCS piping resulting in a release of 8.3×10^{-4} Ci to the environment assuming a 95% efficiency for removal of particulates for each series filter. Our estimates are based on the information given by PNL in NUREG/CR-0199. The licensee has estimated the maximum expected total airborne release from each Unit to the environment from the repair effort will be 3.1×10^{-3} Ci of particulate activity, 4.5×10^{-3} Ci of iodine and 8.5 Ci of tritium. Most of this will be from airborne activity generated during the fuel unloading operations. This compares favorably with the average actual airborne radioactivity releases during 1976 and 1977. For 1976 these releases were 4.1×10^{-2} Ci of particulates, 0.7 Ci of halogens and 186 Ci of tritium released per Unit. During 1977, they were 1.03×10^{-3} Ci of particulate activity, 0.24 Ci of halogens and 440 Ci of tritium per Unit.

The estimated gaseous radioactive effluent resulting from the repair effort are small compared to Surry historical data and those projected from future operations. Therefore, we conclude that the releases will be within the Appendix I to 10 CFR Part 50 Design Objective and therefore, will be ALARA.

2.6.4 Liquid Waste

During the steam generator repair outage, radioactive liquid waste may be generated from (1) disposal of reactor coolant water, (2) disposal of secondary coolant water, (3) local decontamination solutions and (4) laundry waste water.

The reactor coolant will be stored in the boron recovery tanks for reuse after the steam generator repair. Therefore, there should be no significant release from this source.

Secondary coolant water will be significantly contaminated only if the Unit operates with a steam generator tube leak immediately prior to shutdown. We do not discount this possibility. However, even if such a leak exists, based on experience with previous leaks, the activity levels are expected to be low and would not contribute significantly to the total activity released. The licensee has estimated the total release to the environment from the release of secondary coolant water to be 0.22 Ci of mixed radionuclides for one Unit assuming a 0.25 gpm reactor coolant leak into the secondary system. Actual releases from secondary side water should be much less than this value because no reactor coolant leakage is expected. The secondary water will be released to the discharge canal as is normal steam generator blowdown.

Local decontamination will be used to lower radiation levels in the plant. The licensee has estimated the total release from local decontamination for one Unit to be 0.051 Ci of mixed radionuclides. The total volume of water is 18,000 gallons based on a release of 100 gallons per day for the expected 180 day outage.

The major volume of liquid radioactive effluent release will be from laundry waste water. The licensee has estimated that about 12,240 gallons per day will be released. The waste water is expected to be of low specific activity and should not require processing before release. However, it must be sampled. If radioactivity levels would result in releases which exceed those allowed by the Technical Specifications, the waste water will be processed to acceptable levels prior to release. The licensee has estimated the maximum expected release to the environment from laundry wastes to be 7×10^{-2} Ci per Unit with Co-60 making up 29 percent of the total activity and Co-58 making up 37 percent of the total activity.

The licensee has estimated a total maximum expected liquid release of 0.34 Ci of radioactivity (except tritium) and 2.3×10^{8} gallons of waste water for the repair effort for one Unit. We have independently estimated the total liquid release from laundry and general decontamination wastes to be 0.9 Ci. Our estimate is based on the radioactivity releases given in Table 2-20 of NUREG-0017 (April 1976) adjusted for the licensee's estimated release volume. For comparison, the average release of mixed fission (not including dissolved noble gases), and activation products was 17 Ci of radioactivity in 4.5×10^{7} gallons per Unit in 1976 and 24 Ci in 7×10^{7} gallons per Unit in 1977.

The estimated liquid radioactive effluent resulting from the repair effort are small compared to Surry historical data and those projected from future operations. Therefore, we conclude that the releases will be within the Appendix I to 10 CFR Part 50 Design Objective and therefore, will be ALARA.

2.6.5 Solid Waste

Solid wastes generated during the repair effort will include building materials used to construct temporary structures, concrete removed during the repair, miscellaneous piping, disposable protective clothing and solidified liquid wastes, and portions of the steam generators not reused. The disposal of the lower sections of the steam generators is discussed in Section 2.6.6.

The building materials used in temporary work structures should be free of any significant contamination. Only those materials used for a temporary contamination envelope around the reactor coolant piping will be exposed to significant contamination from airborne particulates resulting from the cutting operations. The other structures will be exposed to such contamination as may result from cutting the secondary system piping. The secondary system contamination levels are very small and cutting will not generate significant contaminants.

To facilitate the steam generator lower assembly removal some concrete will be removed from the biological shield surrounding the steam generators and from other structures. The licensee has estimated a total of 1,450 ft^3 of concrete will be removed per Unit with a total activity of less than 0.04 Ci. PNL (NUREG/CR-0199) has estimated that approximately 1,620 ft^3 of concrete may be removed per Unit.

In addition to concrete removal, portions of the steam generator moisture separation equipment and secondary system piping will be replaced and not reused. These portions consist of the feedwater and main steam piping, primary moisture separator, feedwater ring, thermal sleeve, telescoping deck plate, downcomer guard assembly and feedwater nozzle. The present generator insulation, upper steam generator support rings and support ring legs will also not be reused. The removed portions will be shipped offsite as radioactive solid waste due to some low level contamination. The licensee estimates that these will result in approximately 12,600 ft^3 of solid waste consisting of about 0.33 Ci of radioactivity.

A major volume of solid radioactive waste will be compacted rags, trash and disposable protective clothing and equipment. The licensee has estimated about 7,644 ft^3 of such waste containing 6.5 Ci of radioactivity will be packaged and shipped in 55 gallon drums. This should result in about 1,040 drums.

It is also planned to decontaminate the section of reactor coolant system piping, which will be removed during the repair, by electropolishing. This work will be contracted to an outside firm for decontamination after removal. The contractor will provide all equipment necessary for processing the decontamination solutions. No liquid releases are expected from the electropolishing because the spent decontamination solution will be processed by the contractor with the chemical solutions being saved for reuse and the radioactive waste being solidified and disposed of as solid waste. The volume of solid wastes expected to be generated from electropolishing is a small fraction of the volume expected to be produced during the rest of the repair effort. The licensee has estimated that the solidified decontamination waste will consist of two 55 gallon drums containing approximately 12 Curies of radioactivity. Based on expected reactor system contamination levels the staff has estimated the Curie content of the solidified decontamination solution to be approximately 30 Curies in up to ten 55 gallon drums of solidified waste.

The licensee has estimated the repair of one Unit will result in a total solid waste volume of 26,000 ft^3 containing 19 Curies being shipped to a licensed burial facility. PNL (NUREG/CR-0199) has estimated a total of 81,000 ft^3 of solid radwaste will be generated during the repair of one Unit. We have estimated 37 Ci of radioactivity will be contained in this radwaste. The major difference between the licensee's and our activity estimate is the estimate of activity in the solidified decontamination solutions. This compares with the annual average amount of radioactive solid waste shipped during 1973, 1974, 1976 and 1977 of 27,000 ft^3 and 320 Ci, for both Units (or 13,500 ft^3 and 160 Ci per Unit). The year 1975 was not included in this average because of the exceptionally large volume of wastes shipped that year, 325,000 ft^3 containing 26,000 Ci. Thus, exclusive of the lower sections of the steam generator, wastes expected to be generated during the steam generator repair effort for one Unit will amount to about three times a year's worth of solid waste for both Units. This amounts to an increase of about 8 percent over what is expected during the licensed life of both Units. Because of the low specific activity of these solid wastes, shipment will cause no significant effect on the health or safety of the public. All radioactive waste shipments will conform to NRC and DOT regulations.

2.6.6 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) Immediate cut-up and shipment to a licensed burial facility;

(3) Onsite storage until facility decommissioning.

Because of the size and packaging involved, the only method for immediately shipping the assemblies intact would be by barge. At present, there are no licensed burial facilities with receipt capabilities available. Therefore, this option is not viable for the immediate disposition but may become an option in the future.

Immediate cut-up and shipment is possible now with transportation by truck or rail. The assemblies could be cut into suitably sized segments and packaged and transported as low specific activity material. Cutting of the assemblies and subsequent handling would result in significant occupational exposures due to the activity on the surfaces exposed to reactor coolant. Some dose reduction could be achieved by remote cutting of the assemblies. The licensee has estimated a total exposure of 1000-2000 man-rem for the immediate cut-up operation. PNL (NUREG/CR-0199) estimated 1700 man-rem exposure for disposal of 3 assemblies by immediate cut-up and shipment. Further reduction in activity 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 approximately 21% 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. PNL (NUREG/CR-0199) has estimated a total exposure of 810 man-rem for immediate cut-up and shipment following chemical decontamination.

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

The licensee has proposed long term onsite storage to 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 licensee responsibility and control of access and exposure to the assemblies until offsite shipment can be arranged, until the Unit has been decommissioned or until the radiation has decayed to levels that will allow easy disposal. 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. The licensee has stated that this activity will be contained during transport by either fixing the decontamination with a paint or epoxy coating or covering the assemblies with a herculite cover prior to removal from containment. Therefore, no release to the environment should result from transport of the assemblies to the onsite storage facility. There may be some dose to the public due to onsite storage from direct radiation from the steam generators. Based on the maximum expected radioactive inventory of the steam generators and the shielding of the storage facility the licensee 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 outside the facility walls are below the levels for unrestricted areas specified in 10 CFR 20.105. If upon completion of the storage phase the licensee finds levels in excess of 10 CFR Part 20.105 he will be required to provide adequate control and posting pursuant to 10 CFR Part 20.203.

The onsite storage facility will be a concrete structure on a poured structural slab. The facility will be approximately 110 ft x 55 ft with a height of 20 ft. It will be divided into 2 cells with storage for 3 steam generators per cell. The outside walls will be about 3 ft thick. No water accumulation is expected in the facility; however, an internal sump will be provided to collect water. The sump will be checked periodically with a dipstick. Any water that accumulates will be treated as radwaste. Natural ventilation will be provided to allow expansion and contraction of the air in the cell. Although no airborne particulates are expected to be released from the stored assemblies, a HEPA filter will be provided in the ventilation path to minimize particulates from the building. The filter will be changed periodically. The licensee has stated that periodic surveys will be taken to assure that no airborne contaminants are being released from the facility. We have reviewed the licensee's proposed monitoring program for the storage facility and find it acceptable. We conclude that the monitoring program will provide adequate assurance that effluents from the storage facility will be monitored and controlled. Entry into the storage facility is not necessary to change the HEPA filter, check the sump level, or check the airborne radioactivity levels. No electrical power will be provided to the storage facility.

The use of an onsite storage facility will minimize immediate occupational exposures since no disassembly or packaging for shipment is necessary. In addition, the long storage time will allow for significant decay of radioactivity so that ultimate disposal at the end of station life will not be a significant environmental or occupational dose impact. Therefore, we conclude that use of an onsite storage facility is in accordance with ALARA philosophy.

We have reviewed the licensee's proposed method of storage and conclude that there is reasonable assurance that this storage will not endanger the health and safety of the public. 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.

2.7 Quality Assurance

The quality assurance program for the repair of the steam generators will be in accordance with the Virginia Electric and Power Company (VEPCO) Topical Report number VEP-1-3A, "Quality Assurance Programs". Topical Report VEP-1-3A, approved by letter dated February 22, 1977 from Mr. Heltemes to Mr. Baum, outlines the quality assurance program developed to satisfy the requirements of Appendix B to 10 CFR Part 50 for the operations phase.

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. 8A, approved by letter dated September 16, 1977 from Mr. Heltemes to Mr. Eicheldinger. The Westinghouse QA program contains the requirements and controls for the design and fabrication which comply with the requirements of Appendix B to 10 CFR Part 50 and the applicable regulatory guides and standards contained in Chapter 17 of the NRC Standard Review Plan.

We have reviewed the above reports with specific consideration for the proposed steam generator replacement. Based on our review we find that the repair activity is within the scope of the approved programs and that the controls within the approved programs for the proposed work activities comply with Appendix B to 10 CFR 50. Accordingly, we find the provisions established for the quality related activities acceptable.

3.0 EVALUATION

3.1 Design Changes to Eliminate Tube Degradation

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

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 replacement 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 new generators. Along with the absence of phosphates, planned condenser retubing and the installation and use of condensate polishers will 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, tubes in the existing steam generators penetrate support plates through close fitting circular holes. The majority of flow through existing plates 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 holes (annuli) to the extent that the annular gap closes completely. The broached hole or quatrefoil design has circulation in the lobes in the tube holes. This permits substantial tube/tube sheet flow. This results in a continuous flushing and scouring action, thus tending to wash out this area and prevent sludge deposits or scales.

The quartrefoil 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 the licensee has suggested that this will not be a problem in recirculating designs, we feel that the phenomenon is not well enough understood to assume that recirculating type designs will be free of 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 hole arrangement and should be less prone to denting. No denting has been observed in the once-through steam generators.

The repaired steam generators will use SA-240 Type 405 ferritic stainless steel for both the tube support plates and flow distributi baffle plate. The corrosion data provided indicate that, under the test conditions, Type 405 stainless steel will be greatly improved material for tube support plates over the carbon steel presently used. If denting reactions were to 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 (stress relieving) 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.

We have also evaluated the response to a concern regarding fatigue and wear of steam generator tubes that could possibly result from flow induced vibration. Conservative calculations show that the maximum value of the alternating stress is well below the endurance limit for the tube material, even if clearances between tubes and support plates are assumed to increase due to mechanical wear. Additionally, average values of wear co-efficients of the new support plate material, Type 405 stainless steel, are much lower than average values for the old, carbon steel, support plate material. Therefore, we conclude that support plate wear and tube fatigue should not be a problem in the new steam generators.

Based on the information discussed and the evaluation made above, we conclude that the new steam generator design has incorporated features to eliminate the potential for various forms of tube degradation observed to date. Periodic inspection of the steam generator tubes will detect any degradation and ensure that tube integrity is maintained. The inspections will be required by Technical Specifications which will be issued prior to startup with the newly repaired steam generators.

The use of "J tubes" on the feedwater rings in the repaired steam generators and the possibility fo fatigue problems resulting from flow induced vibration has been addressed by the licensee. J-tubes are very stiff and, therefore, have a very high fundamental frequency relative to frequencies of any concern in a seismic or vibrational analysis. The J-tubes meet the ASME Code fatigue requirements. Also, fatigue failures of J-tubes in operating Units have never been encountered. We find the use of J-tubes in the repaired steam generators to be acceptable.

3.2 Effects of Repair Activities

3.2.1 Protection of Safety Related Equipment

The licensee 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 the licensee include the following:

(1) All fuel will be removed from the reactor vessel prior to starting 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 VEPCO Nuclear Power Station Quality Assurance Manual and Section XI of the ASME Code.

(4) The containment boundary will not be disturbed except to open the equipment hatch. Use of the personnel hatch is permissible.

(5) The polar crane will be inspected and tested prior to removal of the old steam generator lower assemblies.

The specific potential hazards considered included the dropping of a steam generator lower assembly, a transporter accident, toppling of a crane, the interaction of systems shared by both Units and fires. Each of these is discussed below.

Defueling of the reactor will begin approximately seven to ten days after shutdown and will be completed in three days. 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 and based on operating experience with the pool cooling systems, the licensee expects the temperature of the water in the pool to be 120°F or less when the pool contains a fresh full core offload in addition to the spent fuel elements currently being stored. We independently estimated the cooling capability of the fuel pool cooling system in our March 23, 1978 Safety Evaluation issued with our approval of the increased storage capacity of the pool. That evaluation demonstrated that the present cooling capacity of the spent fuel pool will be adequate to accommodate the complete defueling of the reactor as planned here with a maximum temperature of 137°F, which is well below the boiling point of water.

In assessing potential hazards associated with the transportation of the removed steam generator lower assemblies, failures of the transporter (which consists of a semitrailer and a haul vehicle) were considered. Structural failure, overturning, runaway and road failure were also considered. To avoid structural failure, the transporter will have a high factor of safety between its rated capacity and the actual load. In considering overturning, the licensee found that one or more tire failures would not cause overturning and the side slopes of the haul route were far below the slopes required for overturning. Administrative limits will be placed on the turning radius and speed of the transporter to preclude overturning.

The tire loading will be well within the capability of the haul route roadways and safety related facilities that pass under it, such as the diesel fuel lines. To provide additional assurance that the diesel fuel lines passing under the roadway have not been damaged, they will be tested after the heavy loads have been hauled over them.

The new steam generator lower assembly will be hauled up a grade near the cooling water intake structure and the old lower assembly will be hauled up a grade in the vicinity of the containment structure for Unit 1. If the haul vehicle were to experience both a transmission failure and a brake failure or the trailer coupling were broken, the vehicle with the new steam generator could possibly roll back toward the intake structure or likewise the vehicle with the old steam generator could roll back toward the operating Unit. There are intervening structures between the grade to the steam generator storage facility and the fuel storage facility that would prevent direct impact of the transporter on the fuel storage facilities. Nevertheless, a guard vehicle will be used behind the transporter when the steam generator assemblies are hauled up a grade. The guard vehicle will prevent a transporter collision with safety related structures.

Most of the haul route will be along the water intake canal for the power station. However, the canal is separated from the roadway by a five foot berm and thus a hauling accident would not impact the canal. Therefore, the cooling water supply for the station would not be jeopardized.

Based on our review of the haul route in relation to safety related structures and components, and our consideration of vehicle failures, overturning, runaway and road failure, we find that the licensee has proposed adequate precautions to prevent accidents associated with the on-site transportation of the steam generator lower assemblies.

The consequences of dropping a steam generator lower assembly (the heaviest load to be lifted during this repair program) either inside or outside of the containment building have been evaluated. Since there will be no fuel in the containment building while heavy loads are being lifted, there will be no significant radiological hazard associated with lifts in the containment building. With regard to dropping a steam generator assembly outside of the containment building, the safety related structures such as the radioactive waste facility and the fuel storage building are not within the range of the devices used to lift the steam generators from the equipment hatch platform to the transporter. We have concluded that dropping a steam generator lower assembly or other identified heavy load associated with this repair program will present no undue risk to safety related structures.

The toppling of a crane having a 160 foot boom with a 30 foot jib extension was considered. The potential consequences of such an accident were considered with respect to the safety related structures, systems and components of the other Unit at the station, including: fuel building walls and roof, low level intake structure, high level intake structure, cooling water discharge tunnel, auxiliary building walls and roof, containment building, control room, service water pumps in the service building, primary grade water storage tanks, refueling water storage tanks, main steam valve house, and offsite power supply lines.

The fuel building, the low level and high level intake structures, the cooling water discharge tunnel, the auxiliary building, the containment, the control room, and the service water pumps in the service building were determined able to withstand the boom impact; no penetration would occur that would result in functional failure of equipment necessary for safe shutdown and continued residual heat removal of the operating Unit or functional failure of the spent fuel pool cooling system.

If the crane boom dropped on either or both primary grade water storage tanks, it would not prevent the safe shutdown of the operating Unit because the refueling water storage tanks would be available as an alternate source of borated water. Conversely, if the crane boom dropped on either or both of the refueling water storage tanks it would not prevent the safe shutdown of the operating Unit because the primary grade water storage tanks would be available as an alternate source of borated water.

If the crane boom were to drop on a main steam valve house it might disable the auxiliary feedwater system and the atmospheric dump valves. This would not prevent the safe shutdown of the operating Unit because auxiliary feedwater from the Unit under repair can be directed to the operating Unit by the operation of switches in the control room. If the atmospheric dump valves could not be opened, the safety valves would open and the hot shutdown condition would be maintained until additional steam relief capability were obtained.

It was determined that the crane boom drop on the offsite power supply lines could not affect all of the lines at once. Therefore, all offsite power could not be interrupted by a postulated crane boom drop.

Based on our review of the proposed hauling and lifting activities associated with the steam generator repair program including consideration of postulated transporter failures, dropping of the heaviest load and toppling of the crane, we have concluded that adequate precautions have been proposed to prevent accidents associated with on-site transportation of the steam generator lower assemblies. We have also concluded that the falling of the crane boom on safety related structures would not prevent the orderly 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 Operable Station 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 operable Unit. The results of that review are discussed below.

Reserve Station Service Transformers

The station service transformers supply 4160 volt power to the station auxiliaries during Unit operation. During start-up and shutdown conditions of normal Unit operation, three reserve station service transformers (RSSTs) (30 MVA or 27 MW each) supply power to the 4160 volt emergency buses for Units 1 and 2.

Temporary loads including construction loads, which are required for the repair of the steam generators, will be supplied by the RSST through each Unit's 4160 volt emergency bus. These loads are relatively small ($\sim$5 MW) compared to the station service load for one Unit ($\sim$35 MW). Therefore the RSSTs are capable of simultaneously supplying the service load to the operable Unit and the temporary load to the Unit under repair.

Emergency Diesel Generators

Units 1 and 2 each have an independent, dedicated diesel generator and they share a swing diesel generator. A safety injection signal on either Unit would normally close the swing diesel-generator breaker to the emergency bus of the Unit in which the safety injection signal occurs and blocks closure of the breaker to the other Unit's emergency bus. Also, Surry has a manual mechanism for the operator to close the diesel generator breaker to the Unit which has had an actual safety injection signal.

During the repair of steam generators, the swing diesel generator will be dedicated to the operable Unit and the interlocking circuit with the other Unit will be disconnected. This will ensure that the Unit under repair will not have any effect on the ability of the swing diesel generator to perform its safety function for the operable Unit.

Temporary Loads

Temporary loads for repair of the steam generators consist mainly of welding equipment. The peak temporary load is anticipated to be 5 MW which is less than 20 percent of the normal Unit emergency load.

These temporary loads will be connected at a junction box located inside the containment and powered from the 480 volt buses. Existing motor control centers and circuit breakers will provide protection against overcurrent and undervoltage. In addition to the existing protection devices, temporarily installed protection devices, in series with and on the temporary load side of the existing protection devices, will provide addition isolation from temporary loads. These circuit breakers will be able to isolate any fault occurring at a temporary load circuit and prevent adverse interaction with the common bus which is shared with the operating Unit.

The administrative controls in use for the existing electrical systems will remain in effect and will be employed to identify and monitor the status of temporary loads.

We have concluded that the proposed protection devices are adequate to isolate faults on temporary load circuits so that power for the operable Unit will not be adversely affected.

3.2.3 Fire Protection

An evaluation of the fire protection program for the Surry Station including the containment buildings of both Units was included in the "Fire Protection Systems Review" for normal plant operation and maintenance activities submitted to the NRC on July 1, 1977. This information was later supplemented by VEPCO's report "Steam Generator Repair Program, Surry Power Station, Units Nos. 1 and 2" which addressed the specific fire hazards associated with the steam generator repair outages.

The use of combustibles in the containment will be minimized to the extent practicable. Metal or fire retardant scaffolding will be used. Good housekeeping will assure that wooden 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. The measures taken above, combined with the licensee's attention to fire protection, demonstrated by the appointment of an assistant fire marshall for the repair effort, provides reasonable assurance that combustibles will be controlled to a minimum.

The fire protection for the containment consists of outside hydrant hose houses accessible to both containment buildings. Portable fire extinguishers and emergency lighting are available at the personnel entrance to containment. Communications for manual fire suppression activities would be by the normal page-type communication system or by portable radio.

The licensee will provide a permanently-installed hose standpipe system in each containment during the early stages of the steam generator repair program. The number of hose stations and the amount of hose at each station will be sufficient to reach all combustibles in containment. In addition, the following measures will be implemented for the duration of the steam generator repair outage:

1. Additional hoses, couplings, and related equipment will be maintained at the two hose houses near the containment equipment hatches. Both 1-1/2 inch and 2-1/2 inch hoses and nozzles will be available to fight fires inside containment.

2. Additional portable fire extinguishers will be placed in containment in areas where flame cutting and welding activities are performed.

3. Additional emergency lighting will be available at the equipment hatch and steam generator cubicles.

4. Portable "bull horns" will be available at or near the equipment hatch.

Although there are no fire detectors in containment, the building will be continuously manned during the steam generator repair outage. In addition, during that phase of the outage when the reactor is fueled, a fire watch will be stationed in areas containing redundant cables for the residual heat removal system. It should be noted here that once the reactor has been defueled in preparation for the repair work no fuel may be inside containment until after the reactor system pressure boundary has been retested for structural integrity and all gear associated with the repair itself has been removed from containment.

Administrative controls related to fire protection are presently in effect at the station and are applicable during the steam generator repair outage. Additional fire protection personnel will be assigned to the repair activities in the containment. As a minimum there will be an assistant fire marshall appointed for the outage activities and fire leaders appointed for each shift. A fire team of at least five men, with appropriate fire training, will be maintained. The station Fire Marshall will direct these additional personnel in fire-related duties. Written procedures will govern the steam generator repair activities and will identify 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 protection measures to be taken to protect safety related structures, systems and components, we have concluded that there is reasonable assurance that the proposed repair activities can be conducted without significantly increasing the potential for damage to safety related systems.

3.3 Transient and Accident Analyses

3.3.1 Discussion

This section discusses the effect the replacement steam generators have on the transient and accident analyses.

As can be seen from Tables 3.3-1 and 3.3-2, the majority of the relevant design parameters and plant operating parameters will not be changed from those of the original steam generators. Therefore, the initial performance of the repaired steam generators during steady state and transient conditions is expected to be comparable to that of the original steam generators prior to tube plugging. The impact of this repair activity on the transient and accident analyses will, therefore, be minimal and the licensee's analyses presented in the FSAR remain valid.

The events analyzed in the FSAR are discussed in the following sections. The following plant conditions were used in those analyses:

Thermal design flow, gpm/loop	88500
SG tube plugging, %	0
*Power level, MWt (100%)	2441
*T_{avg} at 100% power, °F	574.4
ΔT at 100% power, °F	62.8
Steady state DNBR	1.73
$F^{N}_{\Delta H}$	1.55
FQ maximum	2.55

*The analyses (were conducted at) 102% power (2490) and T_{avg} +4° (578.4) to account for uncertainties in determination of the value of these parameters.

It should be noted that for this evaluation the FSAR constitutes the reference cycle. Therefore, although not anticipated based on available information, 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 us should be in accordance with Regulatory Guide 1.70, Revision 2.

It should also be pointed out that the current 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 an 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. Also because of a Westinghouse calculational error, the licensee is required by Order for Modification on Unit 2 dated April 7, 1978, and an Exemption conditioned on Unit 1 dated June 30, 1978, to submit an ECCS analysis using the revised and approved Westinghouse model. We will receive and evaluate the ECCS submittal prior to initial operation with the repaired steam generators.

3.3.2 Non-LOCA Accidents and Transients

In our evaluation, only the 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 withdrawal and rod 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 rod drop accident, the neutron flux redistribution is the limiting consideration. Since this is not dependent on the steam generator performance either, this analysis is not affected by the repair.

For the loss of reactor coolant 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. Therefore, these analyses are not affected by the repair.

For a chemical and volume control system malfunction, the boron dilution rate depends on the charging pump characteristics. The operator must recognize the malfunction and take action to terminate the event. Since the repair of the steam generator will not change the reactor coolant volume from its FSAR value, the repair will not affect the analysis of this event.

The steam generator repair may affect those events for which the transient reactor coolant conditions result from an interaction with the secondary system. These remaining events, which are generally concerned with reactor coolant heatup or cooldown through the secondary side, are discussed in the following sections.

3.3.2.1 Excessive Load Increase

This event involves a rapid increase in 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 percent increase in steam flow from full power can be accommodated without reactor trip. The replacement steam generators, which have a higher (8%) 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

For the case where the stop valves in the inactive loop are open, this event involves the injection of cold water into the reactor vessel and a significant increase in core flow. This results in a rapid increase in core power and reactor trip. The loop transport time is such that the cold water in the inactive loop would not reach the core prior to reactor trip. Therefore, this event is not affected by the steam generator repairs.

For the case where the loop stop valves are initially closed, this event involves the addition of cooler water of low boron concentration into the core. This results in boron dilution and a decrease in available shutdown margin. This event is terminated by operator action. The FSAR analysis assumes the isolated loop contains zero boron. The reactivity insertion rate depends only on the active loop boron concentration and the reactivity coefficients since the maximum flow rate is fixed. The steam generator repairs do not affect this event.

3.3.2.3 Excessive Heat Removal Due to Feedwater System Malfunctions

This event involves the addition of excessive feedwater to the steam generator or the inadvertent 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 have a higher full power inventory, the cooldown rate would be slower. However, the same endpoints condition will be reached. The no-load case will be unaffected since the repaired steam generator conditions will be unchanged from the FSAR.

3.3.2.4 Loss of External Electrical Load

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

3.3.2.5 Loss of Normal Feedwater

The loss of normal feedwater decreases the ability of the secondary system to remove the heat generated in the core. Since the repaired steam generators have a higher full power secondary side inventory, there will be no decrease in their ability to remove heat. Also, since the dimensions of the steam generators will not be changed, the FSAR analysis shows that the tubesheet in the steam generators receiving auxiliary feedwater will remain covered and adequate heat transfer capability will be maintained following loss of normal feedwater. Therefore, there are no adverse effects on this event due to the steam generator repairs.

3.3.2.6 Loss of All AC Power to the Station Auxiliaries

The loss of AC power with turbine 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 average tube height will be increased, thereby increasing the driving head for natural circulation flow. Also, the tubes are recessed slightly into the tubesheet holes, thus causing a lower pressure drop at the entrance to the tubes. The smaller frictional pressure drop enhances the 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 a corresponding increase in core reactivity. The FSAR analysis was performed for end of cycle, hot shutdown conditions. This event will be unaffected by the repaired steam generators because the no load fluid inventory of the steam generators, the flow area of the main steam line, the reactivity coefficients and the emergency shutdown system are unchanged.

3.3.2.8 Steam Generator Tube Rupture

For this event, none of the relevant plant operating parameters or steam generator design parameters are being changed. Therefore, the FSAR analysis of this event is unaffected by the steam generator repair program.

3.3.3 Loss of Coolant Accident

The design and operational differences of the replacement steam generator, such as number of tubes, full power fluid inventory, and pressure drop across the steam generator, are not expected to greatly affect the LOCA analysis. The reduction in flow area due to the lesser number of tubes is approximately equivalent to 1.5% of the tubes in the original steam generator being plugged. The reactor coolant volume is essentially unchanged because fewer numbers of tubes are compensated by the longer tubes.

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 current ECCS analysis, based on the currently approved model, has been performed assuming a significant number of steam generato tubes plugged.

The staff cons rs the ECCS analysis of record to be conservative for plant operation with the replacement 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 licensee will submit such an analysis prior to operation following the replacement of the steam generators. The analysis will accompany a request for Technical Specification changes which will remove certain operating restrictions imposed as a result of tube plugging.

The replacement steam generators do not have a significant effect on the small break LOCA since the steam generators are essentially the same and the thermal-hydraulic characteristics are unchanged. Therefore, the current small break LOCA analyses are acceptable for the plant with the replacement steam generators.

3.3.4 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 fuel 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 an 8% 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 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 secondary 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 side. 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 in leakage of reactor coolant), the increased

Table 3.3-1

STEAM GENERATOR DESIGN DATA (PER STEAM GENERATOR)

	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	650/600	N.C.
Reactor Coolant Flow, lb per hr	33.57×10^6	N.C.
Total Heat Transfer Surface Area, $ft.^2$	51,500	N.C.
Heat Transferred at 100% load, Btu per hr	2778×10^6	N.C.
Steam Conditions at 100% load, Outlet Nozzle:		
Steam Flow, lb per hr	3.5×10^6	N.C.
Steam Temperature, degrees F	516.1	N.C.
Steam Pressure, psig	770	N.C.
Feedwater Temperature at 100% load, degrees F	430	N.C.
Overall Height, ft-in.	67-8	N.C.
Shell OD, upper/lower, in.	178/135	N.C.
Shell Thickness, in.	2.813	2.9
Number of U-tubes	3388	3342
U-tube OD, in.	0.875	N.C.
Tube Wall Thickness (nominal) in.	0.050	N.C.
Number of Manways/ID, in.	6/16	N.C.
Number of Manholes/ID, in.	2/6	6/6 + 2/2
Reactor Coolant Water Volume, ft^3	1077	N.C.
Primary Side Fluid Heat Content, Btu	27.5×10^6	N.C.
Secondary Side Water Volume, ft^3	3581.8	N.C.
Secondary Side Steam Volume, ft^3	1976.7	N.C.
Secondary Side Fluid Heat Content, Btu	95.0×10^6	N.C.
Secondary Side Mass, lb (100% load)	109,000	117,000
Secondary Side Mass, lb (0% load)	170,000	N.C.

*No change

Table 3.3-2

COMPARISON FOR PARAMETERS FOR ORIGINAL AND REPAIRED STEAM GENERATORS

Reactor Coolant Side Pressure Drop	Decreased by 0.1 psi
Fouling Factor	Unchanged
Flow Area	Decreased by ~1.5%
Equivalent Tube Length	Increased by ~1.5%
Nominal Reactor Coolant Temperatures	Unchanged
Nominal Secondary Coolant Temperatures	Unchanged

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 8 percent and most of the dose is a result of "fresh" reactor coolant inleakage, the total offsite doses will increase by much less than 8 percent. This slight increase in total offsite doses will not result in estimated consequences in excess of the 10 CFR Part 100 guidelines, and the conclusions concerning these accidents reached in the February 23, 1972 Safety Evaluation for the Surry Power Station are not changed due to the steam generators repair.

3.4.2 Accidents During Repair

The licensee has analyzed the potential consequences of postulated accidents during the repair effort. The most severe accident which could result in a release of activity to the environment is the dropping of a replaced steam generator which ruptures the steam generator reactor side boundary and allows some of the deposited corrosion products to escape to the atmosphere. The licensee has estimated an offsite dose of 24 mrem to the lung from this accident.

We have independently analyzed the potential consequences of a steam generator drop. We have assumed that dropping of a replaced steam generator will rupture the reactor coolant side boundary, thus exposing the contaminated reactor coolant side surfaces. It is expected that most of the activity on the reactor coolant side is tightly bound to the piping surfaces. This is evident by the fact that the activity was not removed by the high velocity reactor system flowrates during operation. Based on our knowledge of the adherence of the radioactivity to reactor coolant side surfaces, that activity which may become loosened will mostly be deposited on the large surface areas inside the steam generator, and that there will be little air movement between the steam generator internal air spaces and the outside atmosphere we have conservatively assumed that 0.1 percent of the activity in the steam generator becomes airborne and is released to the atmosphere. The resultant dose to the critical organ of an individual at the site boundary is 0.6 rem to the lung. The assumptions used in the calculation are listed in Table 3.4-1 and the results are given in Table 3.4-3.

We have also analyzed the potential radiological consequences of a crane drop onto the refueling water storage tank (RWST) or reactor grade water storage tank of the operating Unit. Since the reactor grade water storage tank contains a smaller volume than the RWST and is clean reactor grade makeup water, the drop onto the RWST is the more severe accident. The RWST is maintained at 45°F to promote containment cooling in the event of a loss-of-coolant accident. Since, at this temperature the water will not flash or readily evaporate and most of the radioactivity will remain in the liquid phase, we have assumed that 1 percent of the volative radioactivity becomes airborne. We have conservatively assumed the activity in the tank to be at reactor coolant levels diluted by the tank volume with no credit taken for decay of the nuclides. The resultant dose is 0.4 rem to the thyroid. Plant measurements of the activity in the RWST indicate that the actual doses will be much less. The assumptions used in the calculations are listed in Table 3.4-2 with the results given in Table 3.4-3.

3.5 Special License Conditions and Technical Specifications

During the repair program certain additional temporary Technical Specifications or license conditions will be required. There will be an operability requirement for the temporary containment and local ventilation systems for all cutting operations, a requirement for removal of all fuel from the reactor vessel and storage in the spent fuel pool, a requirement to submit a program for preoperational testing and startup prior to fuel loading, a requirement to submit periodic reports summarizing the occupational doses and effluent releases, and a requirement to implement a health physics program.

3.6 Security

The licensee identified measures in a submittal dated October 25, 1978 that will be implemented during the repair program to assure that the security program in effect at the Surry Power Station is not degraded as a result of steam generator repair program activities. We have reviewed the licensee's program in light of these measures and have concluded that the program will not be degraded.

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 proposed steam generator repair program, (2) such activities will be conducted in compliance with the Commission's regulations and (3) approval of the proposed modifications will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 15, 1978

TABLE 3.4-1

ASSUMPTIONS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR DROP

Activity in Steam Generator (Ci)*	1400
Fraction of Activity Becoming Airborne	0.001
Site Boundary χ/Q (S/m^3)	1.6×10^{-3}
Lung Inhalation Dose Converson Factor ** ($\frac{mrem}{pCi}$)	7.46×10^{-4}
Breathing Rate ($\frac{m^3}{S}$)	3.47×10^{-4}

* All activity is assumed to be Co-60.

** From Regulatory Guide 1.109.

TABLE 3.4-2

ASSUMPTIONS USED IN CALCULATING RADIOLOGICAL

CONSEQUENCES OF CRANE DROP ONTO REFUELING

WATER STORAGE TANK

Tank Volume (gallons)	350,000
Isotopic Concentration in Tank ($\frac{\mu C}{cc}$)*	
I-131	.027
I-132	.011
I-133	.039
I-134	.005
I-135	.021
Fraction of Activity Becoming Airborne	0.01
Site Boundary χ/Q ($\frac{S}{m^3}$)	1.6×10^{-3}
Breathing Rate ($\frac{m^3}{S}$)	$3.47 \times .10^{-4}$
Thyroid Dose Conversion Factors ($\frac{Rem}{Ci}$)	Regulatory Guide 1.25

* Assumes Reactor coolant activity diluted by RWST volume.

TABLE 3.4-3

RADIOLOGICAL CONSEQUENCES OF

POSTULATED ACCIDENTS DURING REPAIR

Accident	Dose (Rem)
Steam Generator Drop	0.6*
RWST rupture	0.4**

* Dose to Lung

** Dose to Thyroid

APPENDIX A

Questions by Commonwealth of Virginia in Letter Dated January 17, 1978

As a result of the interest in the Steam Generator Repair Program by the Commonwealth of Virginia, we requested the licensee to answer the questions posed by the Commonwealth. The answers are contained in the licensee's report titled "Steam Generator Repair Program, Surry Power Station, Units 1 and 2." This appendi· was prepared to give the NRC staffs' evaluation of the licensee's response.

Question 1

In the opinion of our consultant, the tubes in the repaired steam generator may be subject to damage due to cross-flow vibration. This damage may take the form of a circumferential rupture of the tube at the tube sheet (where the tube is fixed) due to movement of the tube within the clearance space between the tube and the support plates. This effect may be increased over time if the clearance space (and the amplitude of the tube's movement therein) increases due to wear of the support plate material. In the event of a loss of coolant accident the fatigued tubes might fail causing secondary water and steam to enter the primary system with potential adverse effects on the injection of emergency core cooling water. Alternating stress conditions in the tube, in the vicinity of the tube sheet, should be calculated for the amplitudes expected throughout the steam generator lifetime in addition to those calculated for the new, unworn tube support plate clearances to assure that the alternating stresses will be within the allowable stress fatigue limits of the Inconel 600 material for the lifetime of the steam generator.

We understand that other steam generators of a broached quatrefoil design similar to the design proposed by Westinghouse have experienced such damage. Furthermore, we are informed by telephone consultation with Westinghouse that this phenomenon has been investigated by Westinghouse, at least with respect to its occurrence assuming the initial clearances prior to any wearing of the tubes or the tube support plates. In view of the foregoing, the Commonwealth requests that you resolve two questions arising out of this issue, and specifically confirm in your staff Safety Evaluation that:

1a. The alternating stresses described above are within the fatigue life limit of the steam generator tube material, for the predicted life of the steam generator.

1b. The alternating stresses will remain satisfactory during the life of the steam generator in view of the probability of wear and increased clearance between tube and tube support during the steam generator lifetime.

NRC Evaluation of 1a and 1b

Our evaluation is found on page 29 of the SER.

Question 2

In addition to the issue identified above, there are several issues which we examined with respect to whether the repaired steam generators can be expected to operate with reasonable efficiency throughout the remaining operating life of the Surry units. Although we are satisfied that these issues do not significantly increase the probability of accidents or radioactive release to the environment, any such future repairs could involve increased employee exposure to radiation. Moreover, a large part of the justification for the repairs (Section 5 of the application) is that operation of the plant will be more efficient after the repairs. The potential for further repairs in the future would weaken this justification.

For these reasons, we suggest that the Staff satisfy itself on the following issues and address them in the safety evaluation:

2a. The heat treatment of the steam generator tubes should be justified with regard to whether those tubes in which residual stress would be significant after bending would be stress relieved. (Application Section 2.3.15, 2.4.5).

NRC Evaluation of 2a

Our evaluation is found on pages 6 and 29 of the SER.

2b. Due to the increase in length of the tube bundle, the top of the bundle will be closer to the steam separators. We believe that you should attempt to confirm that no deleterious effect on steam separation will result from this change. (Application Section 2.5.3, 2.3.25).

NRC Evaluation of 2b

In the licensee's response to this question, page SSGP 9.E.3-1 of the Steam Generator Repair Program Report, it is stated that with the use of shorter separators, the minimum clearance between the top of the tube bundle and the bottom of the lower deck plate has actually increased.

2c. So-called "J-tubes" are welded to the feedwater distribution ring in place of the usual orifices. We believe that you should confirm that these tubes will not be subject to fatigue failure or other flow induced phenomena. (Application Section 2.6.5).

NRC Evaluation of 2c

Our evaluation is found on page 30 of the SER.

2d. VEPCO intends to cut certain components of the primary system by flame cutting. We believe the safety evaluation should describe measures to prevent the entry of debris from flame cutting into the primary system.

NRC Evaluation of 2d

We have reviewed the measures proposed by the licensee in its report on page SSGP 9.E.5-1 and have concluded that the measures described provide assurance that any debris resulting from the flame cutting will not be present in the reactor coolant system upon completion of the steam generator replacement effort. The measures taken by the licensee include (1) melting a narrow kerf which minimizes the amount of metal, (2) selection of cut locations which minimizes the entrance of debris such as having no vertical pipes into which debris could fall, (3) closing the reactor coolant isolation valve and (4) use of detailed procedures with appropriate supervision. Some debris may enter adjacent reactor coolant piping during cutting and there is no need to prevent this from occurring, because, prior to re-welding, the piping between the cut and the reactor coolant isolation valve, will be cleaned, thus removing any debris which may have entered the reactor coolant system.

2e. Depressions caused by metal stamping the tube numbers on the tube sheet may collect radi-active crud and potentially increase radiation exposure to personnel working in the area of the tube sheet.

NRC Evaluation of 2e

The licensee has responded to this question on page SSGP 9.E.6-2. The staff has reviewed the licensee's response and concurs with the licensee's conclusions. The depression caused by the stamping is only 10 mils deep and a relatively smooth surface. Considering the irregular surface of the tube sheet (penetrations for tubes) and mechanics of crud deposition, the stamping should not affect the general radiation levels in the channel head. In addition, the marking of the tubes will significantly reduce the time involved in tube identification for plugging and thus reduce the time spent in plugging and the resulting occupational exposures. Therefore, we conclude that the tube marking will serve to reduce total occupational exposures and thus is in accordance with an ALARA philosophy.