Central Files
50-280

October 27, 1978

MEMORANDUM FOR: Gary Quittschreiber, Senior Staff Engineer
Advisory Committee on Reactor Safeguards

FROM: A. Schwencer, Chief, Operating Reactors Branch #1, DOR

SUBJECT: DRAFT SAFETY EVALUATION REPORT FOR SURRY STEAM GENERATOR REPAIR PROGRAM

I am forwarding you 16 copies of a draft Safety Evaluation Report concerning the Surry steam generator repair program and 16 copies of report NUREG/CR-0199, PNL-0199, "Radiological Assessment of Steam Generator Removal and Replacement."

These reports are for use by the ACRS Surry Subcommittee at its meeting on October 28, 1978.

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachments -
1. Safety Evaluation Report (16)
2. NUREG/CR-0199 (16)

DISTRIBUTION
Central Files
ORB#1 RF
ORB Subject file
V. Stello
D. Eisenhut
OELD
~~XXXXX~~ Docket File 50-280/281
NRC PDR
LOCAL PDR

7811060060

OFFICE	DOR:ORB#1/PM	DOR:ORB#1/C	DOR:A/D:S&P			
SURNAME	DNeighbors:pab	ASchwencer	DGEisenhut			
DATE	10/27/78	10/27/78	10/27/78			

Docket 50-280

OCTOBER 27, 1978

MEMORANDUM FOR: Gary Quittschreiber, Senior Staff Engineer
Advisory Committee on Reactor Safeguards

FROM: A. Schwencer, Chief, Operating Reactors Branch #1, DOR

SUBJECT: DRAFT SAFETY EVALUATION REPORT FOR SURRY STEAM GENERATOR REPAIR PROGRAM

I am forwarding you 16 copies of a draft Safety Evaluation Report concerning the Surry steam generator repair program and 16 copies of report NUREG/CR-0199, PNL-0199, "Radiological Assessment of Steam Generator Removal and Replacement."

These reports are for use by the ACRS Surry Subcommittee at its meeting on October 28, 1978.

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachments -
1. Safety Evaluation Report (16)
2. NUREG/CR-0199 (16)

DISTRIBUTION
Central Files
ORB#1 RF
ORB Subject file
V. Stello
D. Eisenhut
OELD
XXXXX Docket File 50-280/281
NRC PDR
LOCAL PDR

DOR:ORB#1/PM	DOR:ORB#1/C	DOR:ORB:S&P
DNeighbors:pab	ASchwencer	DGEisenhut
10/ /78	10/ /78	10/ /78

* * * * * *

SAFETY EVALUATION 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

* * * * * *

DRAFT

1.0 INTRODUCTION

By letter dated August 17, 1977, as supplemented December 2, 1977, April 21, June 2, June 13, June 30, September 1, and October 25, 1978, Virginia Electric and Power Company (the licensee) submitted a steam generator repair program for Surry Power Station, Units 1 and 2. We determined that the proposed program requires our review and approval. Also, the ACRS has expressed an interest in this program and our evaluation of it. Our evaluation of this program is presentd in this report. This evaluation will be supplemented to address additional matters raised by the ACRS, if any.

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.

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, it 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 ratio exceeds 2.6. In addition,

DRAFT

impurities in the feedwater, were not being adequately removed by blowdown. These phosphate precipitates tended to accumulate as sludge on the tubesheet and tube supports and associated crevices in the central region of the tube bundle where restricted water flow and high heat flux occur. This phosphate concentration (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 between 2.6 and 2.3. However, this did not correct the phosphate hideout problem causing wastage of the Inconel-600. 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 has occurred predominantly in rigid regions or so-called "hard-spots" in the tube support plates. These support plate hard spots are at 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. Also, it is worth noting that the hard spots do not contain the same array of adjacent 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 penetrate the support plates. The adhering corrosion product (magnetite) occupies approximately twice the volume of the material corroded. Thus, the continuing corrosion coupled with thermal cycling has, in the hard spots, exerted sufficient compressive 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 center of some of these slots have even closed.

DRAFT

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 initial 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 be required to initiate stress corrosion cracking of the Inconel-600 alloy tubing exposed to PWR reactor coolant.

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 compressive forces discussed 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.

DRAFT

Due to the continuing denting related problems (requiring plant shutdown and occupational radiation exposure), the certainty of 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 REPLACEMENT 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 land blocking devices, and modifying the feedring. These modifications will be retained or improved upon in the replacement steam generators. Also, additional modifications, as discussed below, will be incorporated into the replacement steam generators.

A flow distribution baffle plate, located 18" above the tubesheet, will be used in the replacement 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 replacement 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 will tend to deposit.

Unlike the existing design, all tubes in the replacement generators will be expanded to the full depth of the tubesheet 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 is the carbon steel now in use. Furthermore, corrosion of SA-240 will result in an oxide which occupies approximately the same volume as the parent material. Thus its corrosion product is not expected to exert the significant stresses observed with the present design.

As another important design change, the tube support plates in the replacement steam generators will be "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 (tube support plate annulus) eliminate the denting phenomenon.

The tubes in the replacement generators will be recessed slightly into the tubesheet holes and then welded to the tubesheet 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 replacement generators, modifications to the moisture separator equipment will be made to accommodate this increase.

DRAFT

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 aditional 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 replacement generators to improve drainage of the channel head. The replacement 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 Inconnel-600 tubing used in the replacement steam generators will be thermally treated after forming to reduce residual stresses imparted by tube processing, thereby improving its resistance to stress corrosion. Several benefits are expected to result from this reduction of residual stresses. These include improved resistance in stress corrosion cracking in NaOH, improved resistance to intergranular attack in oxygenated environments, and improved resistance to intergranualar attack in sulphur-containing species. The thermal treatment will be within a time-temperature band to avoid formation of a chromium depleted grain boundary layer (sensitization).

2.3 ASME Code and Regulatory Guide Implementation

All new component parts of the replacement 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.

DRAFT

DRAFT

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 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. The reactor vessel head will be removed for defueling and the actual removal of the fuel will commence seven days or more after shutdown. After all fuel has been removed from the reactor to the spent fuel storage facility, the reactor vessel head will be replaced. A protective cover will then be placed over the reactor vessel and local decontamination work will begin. The biological shield wall and a section of the pressurizer cubicle wall will be removed to provide access to the first steam generator. 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 shall 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

DRAFT

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 recontructed.

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.

DRAFT

DRAFT

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 generator.

The licensee has not yet completed the preparation of detailed procedures for preoperational testing and startup after completion of the steam generator repairs. The NRC's Division of Inspection and Enforcement will review the detailed procedures prior to fuel loading to verify that adequate testing will be performed to ensure safe startup of the Unit after completion of these repairs.

2.6 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.

DRAFT

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 are intended to be conservative and represent upper bound values. The licensee's estimates are based on actual plant data and should be more representative of the maximum expected values. 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 plant basis (i.e., repair of 3 steam generators) unless otherwise noted.

2.6.1 Occupational Radiation Exposure

Removal and reinstallation of the repaired steam generators, separation, 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 three major categories: post-shutdown preparation, steam generator removal, reinstallation of the repaired steam generators, and disposal of portions not reused in the repaired steam generators.

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 installed. 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 exposures should be similar to a normal refueling.

DRAFT

Following defueling and prior to starting removal of the first steam generator, the reactor coolant system will then be 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. PNL (NUREG/CR-0199) has estimated an upper boundary total occupational dose of 420-780 man-rem for these activities. The major portion of this dose estimate is attributed to installation of radiation control equipment such as shields and temporary contamination control structures. Following these preparation activities, the steam generators will be removed one at a time.

Removal

Removal activities include removal of the thermal insulation around the steam generators and pipe separation areas, and of the reactor coolant and secondary system piping. Main steam lines, feedwater, reactor coolant inlet and outlet

DRAFT

and miscellaneous pipe segments must all be removed to provide clearances in the steam generator area. The highest exposures would most likely be received during cutting of the reactor coolant piping. 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. The plasma arc cutting device will make these cuts to minimize the total personnel stay time in the radiation fields near this piping. In addition, extensive shielding of adjacent high radiation sources 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 familiariziation training should minimize the time required to perform the operations and thus, minimize time spent in radiation fields to an absolute minimum. The cut reactor coolant pipe ends will be covered with shields to reduce streaming from the internal surfaces.

Following removal of the cut pipe sections, the steam generator upper shell wrapper and upper internals will be cut to permit removal of the upper shell. Due to the low radiation at this location, minimal shielding will be required and hand cutting may be used. The low radiation levels expected to be present are due mainly to previous tube leaks. The washing effect of the steam has kept the resulting deposital radioactive solids on the shell side at a low level. The expected low contamination levels on the secondary side also preclude the necessity of using contamination control envelopes at this location to control the spread of airborne activity. The steam generator wrapper and upper internals will be cut from inside the steam generator dome with access through the manways. Because the work area inside the steam generator is limited and because of the low radiation fields involved at this location, manually operated torches may be used with no significant increase in exposures when compared to the use of remotely operated equipment and installation of extensive shielding. This is because of the time involved in installation and removal of such equipment. Radiation fields in the areas where personnel are located during the cutting operations will be minimized by maintaining the secondary side water level as high as possible to effect maximum shielding of the radiation from the steam generator tubes. All openings in the steam generator lower

DRAFT

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 severed, 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 assemblies. 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 generators are brought into the containment. The licensee has esimated an expected maximum total occupational exposure of 560 man-rem per Unit for the removal activities. PNL (NUREG/CR-0199) has estimated an upper bound dose of from 1100-1700 man-rem for the removal phase of which 210-570 man-rem would be from the reactor coolant system pipe cuts.

Installation

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 exposures, a remotely operated 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 while PNL (NUREG/CR-0199) has estimated an upper bound exposure of 1800-3300 man-rem.

Disposition 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 4.3. 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.

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 upper bound dose of between 3300 and 5800 man-rem per Unit for the whole repair program. This estimate is based on the same methods used by the licensee. The lower PNL value is based on shielding and remote tooling techniques which would be used, as proposed by the licensee, to keep exposures ALARA.

Extensive planning will be used in the repair effort, including the health physics aspects. An individual knowledgeable in health physics has been assigned fulltime 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. 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.

DRAFT

We have reviewed the licensee's submittal regarding occupational exposures and conclude that efforts being made to maintain occupational exposures ALARA are acceptable.

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 if necessary temporary systems will be available to process the maximum expected level of airborne and liquid wastes generated.

2.6.2 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 (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.

DRAFT

DRAFT

The licensee has estimated that a maximum of 6×10^{-4} Ci of particulate activities per Unit will be released to the environment as a result of the RCS piping cuts via filtered ventilation systems. This assumes 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. A filter efficiency of 95% was assumed for each set of filters in a 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 of 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 released per Unit.

2.6.3 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 leakage 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

DRAFT

the total release to the environment 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 committed to use solutions which will be compatible with the liquid radwaste system. The licensee has estimated the total release from 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 excess radioactivity levels are detected, 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^{6} 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.

Based on the above consideration, we expect that the total estimated release of radioactivity in liquid effluents will be less than those encountered during reactor operations.

2.6.4 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 2.6.5.

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. The other structures will be exposed to such contamination as may result from cutting off the secondary system piping; this is expected to be insignificant.

To facilitate the steam generator lower shell 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 removal 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 (approximately 300 gallon volume) will be solidified and disposed of as solid waste in a licensed burial site. 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 staff has estimated the Curie content of the solidified decontamination solution to be approximately 30 Curies.

The staff has estimated that the total amount of solid waste to be shipped for disposal to a licensed burial site consists of 81,000 ft^3 and 37 Ci of radioactivity per Unit. This compares with the annual average amount of radioactive solid waste shipped during 1973, 1974 and 1976 of 26,800 ft^3 and 80 Ci, for both Units (or 13,800 ft^3 and 40 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 2,600 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 twice a year's worth of solid waste for each Unit. This amounts to an increase of about 15 percent over what is expected during the licensed life of each Unit. 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.

DRAFT

2.6.5 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.

DRAFT

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. 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. The assemblies will be sealed 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. The licensee has estimated 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 reasonable.

The onsite storage facility will be a concrete structure on a poured structural slab. The facility will be approximately 105 ft x 50 ft with a height of 19 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

DRAFT

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. 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 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.

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 licensee's ongoing effort 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 10, 1977 from Mr. Heletemes to Mr. Eicheldinger.

DRAFT

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 adequate controls exist within the approved programs for the proposed work activities. 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.

We concur that a flow distribution baffle plate should minimize, or at least reduce, the number of tubes exposed to 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 potential for tube wastage since fewer tubes will be exposed to less sludge.

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

A quatrefoil support plant design will be used in the repaired steam generators. In contract, tubes in the exiting 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 annuli tube/tube support plate holes to the extent that the annular gap closes completely. The broached or quatrefoil hole 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, to prevent sludge deposits or scales in this tube/tube sheet area. Additionally, the open areas provide substantial space for expansion of the tube support plate metal without exerting major compressive forces on the tubes, even if one assumes that there will continue to be substantial magnetite growth which, as noted below, is rendered insignificant by the use of stainless steel for tube support plates rather than carbon steel.

The quarterfoil 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 distribution 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.

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. 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 coefficients 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 concur that support plate wear and tube fatigue should not be a problem in the new steam generators.

DRAFT

DRAFT

The use of "J-tubes" in the repaired steam generators and the possibility of fatigue problems resulting from flow induced vibration has been addressed by the licensee. J-tubes are very still 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.

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.

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. The installation contractor will be required to have an ASME "N" stamp applicable to the work performed by that contractor.

(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 generators.

DRAFT

DRAFT

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 temperature of the water in the pool is expected 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.

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. As an added precaution, the diesel fuel lines passing under the roadway will be tested after the heavy loads have been hauled over them.

If the haul vehicle were to experience both a brake failure and a transmission failure, the transporter could possibly run away. To preclude runaway, a guard vehicle will be used behind the transporter when the lower assemblies are hauled up the grade to the steam generator storage facility.

DRAFT

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, 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 assembly (the heaviest load to be lifted during this repair program) either inside or outside of the containment building has been considered. 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 pit 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.

We have concluded that the falling of the crane boom on these safety related structures would not prevent the safe shutdown of an operating Unit and would not prevent adequate cooling of the fuel assemblies in the spent fuel pool.

3.2.2 Other Interactions with Operable Station Units

The normal and emergency electrical power distribution system 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) supply power to the 4160 volt emergency buses for Units 1 and 2.

DRAFT

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 (~5 MW) compared to the emergency load for one Unit (~35 MW). Therefore the RSSTs are capable of simultaneously supplying the emergency 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 block 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 will provide additional 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.

DRAFT

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, traditional amounts of combustible materials will necessarily be introduced into containment including protective clothing, cleaning fluid, charcoal filters and plastic sheeting.

The fire protection for the containment consists of outside hydrant hose houses accessible to both containment buildings. Portable fire extinguishers and emergency lighting area 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 before the initiation 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 extinghuishers 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 area 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 traiing, 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.

DRAFT

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 minimial 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 conservatively used 102% power (2490) and T_{avg}+4° (578.4)

DRAFT

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 the approved model will be required.

The licensee will submit a reload report prior to operation which will evaluate any values not bounded by the values in the FSAR. Also because of a Westinghouse calculational error, the licensee is required to submit an ECCS analysis using the revised and approved Westinghouse model. We expect to receive and evaluate these submittals 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.

DRAFT

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 seady 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.

DRAFT

DRAFT

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.

DRAFT

DRAFT

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 a no decrease in their ability to remove heat. Also, since the dimensions of the steam generators will not be changed, the FSAR concludes 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 and 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.

DRAFT

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

The staff considers 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 has committed to 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. Therefore, the current small break LOCA analysis are acceptable for the plant with the replacement steam generators.

DRAFT

DRAFT

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 remain valid.

3.4 Radiological Consequences of Postulated Accidents

The repaired steam generators will not 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 changes 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

DRAFT

DRAFT

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 x 10^6	N.C.
Total Heat Transfer Surface Area, ft.^{2}	51,500	N.C.
Heat Transferred at 100% load, Btu per hr	2778 x 10^6	N.C.
Steam Conditions at 100% load, Outlet Nozzle:		
Steam Flow, lb per hr	3.5 x 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 x 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 x 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

DRAFT

DRAFT

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

DRAFT

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.

3.5.6 Special License Conditions and Technical Specifications

Following on-site storage of the lower assemblies, the licensee will be required to monitor the storage facility for sump water level, airborne activity and conditions of the HEPA filter on a periodic basis. Initial frequency shall be monthly to be re-evaluated in six months.

During the repair program itself, certain additional temporary Technical Specifications or license conditions will also be imposed. These will include: (a) the temporary containment and local ventilation systems shall be operable for all cutting operations; (b) significant deviations from expected doses shall be immediately brought to the NRC's attention; (c) significant deviations in the repair program presented to and for approval by the NRC staff shall require prior NRC approval.

3.6 Security

The licensee has identified measures 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.