Central File

50-250

MAR 2 3 1979

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

FROM: G. Knighton, Chief, Environmental Evaluation Branch, DOR

SUBJECT: REVIEW OF TURKEY POINT STEAM GENERATOR REPAIR PROGRAM (TAC 7126)

The Environmental Evaluation Branch (EEB) has reviewed the Florida Power and Light Company's (the licensee) "Steam Generator Repair Report, Revision 6," for Turkey Point Units 3 and 4. The EEB contributions to the Safety Evaluation, Section 2.6, "Radiological Considerations," and Section 3.4, "Radiological Consequences of Postulated Accidents" are enclosed.

We have concluded that the replacement steam generators will not have any appreciable effect on the radiological consequences of postulated accidents and will not alter the conclusions reached in the March, 1972 Safety Evaluation.

We have also concluded that the licensee's efforts to maintain occupational exposure to ALARA values during the repair effort are reasonable and adequate radiation protection will be achieved.

We have further concluded that the radioactive effluents which may be released as a result of the repair effort are less than those expected during normal operations and can be maintained within the radiological effluent technical specifications and will not affect the health and safety of the public.

We have identified an item to which the licensee has not provided an adequate response. The licensee has not provided adequate justification for their decision not to monitor liquid releases for Fe-55 and Ni-63. Therefore, we will require the licensee to perform a monthly composite sample on all liquid effluents for Fe-55 and Ni-63 during the repair operations, to consider these isotopes when comparing actual releases to technical specification requirements and to report the releases in the semi-annual effluent report.

The licensee should be required to provide a report of the actual occupational exposures spent during the Unit 4 repair. This report should contain a summary of the techniques used to keep exposures ALARA and an evaluation of their effectiveness. The licensee should include a critique of their efforts, compare actual to estimated exposure, and include any proposed changes to their Unit 3 program if the Unit 4 experience indicates that changes should be made to keep exposures ALARA. The man-rem summary should be in sufficient detail

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OFFICE>					
SURNAME>	Contact: G. Cwalina, EEB/DOR				
DATE>	28066				

to identify exposures for the various steps of the process. A suggested format would be Tables 13, 14 and 15 of NUREG/CR-0199. The report should be submitted to the NRC 90 days after completion of the Unit 4 repair.

Original signed by
George W. Knighton

George W. Knighton, Chief
Environmental Evaluation Branch
Division of Operating Reactors

Enclosures:
As stated

cc: V. Stello
D. Eisenhut
B. Grimes
R. Vollmer
J. D. Neighbors
J. Collins
T. Murphy
W. Kreger
R. Ballard
A. Gibson, Region II
R. W. Houston
J. Scinto
R. A. Alexander
S. Kirslis
J. Minns
E. Adensam
Section B/EEB

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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 Turkey Point 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 lower value was estimated assuming credit for various techniques to reduce exposures, e.g., providing water shielding by maintaining high steam-generator water levels, remote tooling and distance where applicable. The licensee has committed to these types of techniques, consequently, it is appropriate to compare the PNL lower value with the licensee's estimates.

The licensee's occupational exposure estimates include a detailed estimate of doses based on major job functions of 1300 man-rem per Unit. These detailed estimates do not include dose savings from use of temporary shielding and local decontamination or dose costs from implementation of these. However, the licensee has estimated a range of doses for the replacement of from 650-1450 man-rem per Unit. The range of doses presented represents the licensee's best judgment with respect to the predicted worker dose considering uncertainties in prediction of job man-hours and radiation fields. The radiation field uncertainties consider the effectiveness of temporary shielding and the time required to place such shielding. Therefore, although the licensee has not included the effect of temporary shielding and local decontamination in their detailed estimate, they have considered it in their predicted range of doses. For comparison purposes in this report, we are evaluating the PNL lower estimate versus the licensee's detailed estimates.

The licensee's estimates are generally lower than PNL's lower values because the licensee used actual plant data which are lower than PNL's radiation field estimates. The licensee's dose estimates are based on a range of radiation field values from actual in-plant surveys at Turkey Point. The estimates assume occupancy is in an average radiation field. The licensee has stated that use of temporary shielding will be determined based on radiation surveys and an estimate of the dose savings from use of shielding compared to the dose incurred from installation of the shielding. Consequently, the actual radiation fields are expected to be within the range of values given in the licensee's report.

The licensee's estimates include an estimate of 200 man-rems for miscellaneous activities such as supervision, quality assurance and health physics. We have divided the estimate between the removal and installation phase in this evaluation to permit comparison with the PNL estimates.

PNL also provides 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 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. The time involved in defueling an entire core will be similar to the time involved in defueling, shuffling and refueling 1/3 of a core. Since the radiation fields will be essentially the same as a normal refueling, the expected occupational exposure should be similar to a normal refueling.

Following defueling and prior to starting removal of the first steam generator, temporary structures will be installed to facilitate the steam generator separation and removal activities. These structures

include contamination control envelopes around the reactor coolant piping at the separation points, temporary ventilation systems, scaffolding, and construction of a temporary platform with guide rails at the equipment hatch to facilitate removal of the steam generator lower assemblies.

The preparation activities also include radiation surveys and local decontamination. Portions of the concrete shield walls will be removed to permit later removal of the steam generator lower assemblies. Some small sections of containment internal structures must also be removed to permit removal of the lower assemblies.

The thermal insulation around the steam generator, reactor coolant and main steam piping will also be removed. A new 250 ton construction hoist will be placed on the polar crane bridge because the existing trolley is not capable of handling the lower assemblies. Load testing of the new hoist will assure that current OSHA safety standards are met.

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 257 man-rem per Unit (excluding refueling) for these post-shutdown preparation activities. The major portion of this dose estimate is attributed to installation of temporary structures, local

decontamination and removal of insulation. The licensee has not provided a detailed estimate for installation of temporary shielding. The licensee states that the need for temporary shielding will be treated on an individual case basis. The need for shielding will be evaluated based on the dose savings for performing the job with shielding versus the dose incurred during installation and removal of the shielding.

PNL (NUREG/CR-0199) has estimated an occupational dose of 450 man-rem for the post-shutdown preparation activities including 20 man-rem for defueling. The PNL estimate also assumes control of the steam generator secondary side water level to shield radiation emanating from the primary side corrosion products. PNL has included an estimate of 144 man-rem for radiation surveys, local decontamination and installation of shielding. It is the staff's opinion that some shielding and local decontamination will be necessary. Although the licensee has not provided an estimate for installation of shielding in the detailed dose estimate, the licensee's range of dose estimates does consider the effectiveness of temporary shielding and the time required for installation of the shielding, and is based on his knowledge of plant specific design and should be more representative of that actually spent.

2.6.1.2 Steam Generator Removal

Removal activities include removal of the main steam lines, feedwater reactor coolant inlet and outlet and miscellaneous pipe segments. These must all be removed to provide clearances in the steam generator area. The highest exposures will most likely occur during preparation and cutting of the reactor coolant piping and cutting and removal of

the steam generator upper internals because of the manhours required in the radiation areas to complete the cutting. The reactor coolant system pipe 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 licensee plans to utilize remote cutting tools wherever possible to minimize personnel stay time in radiation areas. It is planned to use mockups to familiarize personnel in the specifics of the cutting operations including space restraints, protective clothing, and special tasks required. The familiarization training should minimize time spent in radiation fields. The 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 on the containment operating floor. Remote cutting tools will be used wherever possible. The expected low contamination levels on the secondary side preclude the necessity of using contamination control envelopes at this location to control the spread of airborne activity. The steam generator wrapper and upper internals will be cut from outside the steam generator after the upper shell has been removed. The steam generator water level will be kept high to shield personnel making the cuts from radiation emanating from the lower shell internals. 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 because PNL assumed no credit for shielding

from keeping the water level high. The licensee's estimate of occupational exposure to cut the wrapper is lower than PNL's because it is based on radiation fields actually measured at Turkey Point which are lower than those assumed by PNL.

All openings in the steam generator lower shell will be sealed with welded metal seals prior to removal of the steam generator lower assemly 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 and most of the internal moisture separation equipment will be reused. The upper shell will be 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 a total occupational exposure of 436 man-rem per Unit for the removal activities. PNL (NUREG/CR-0199) has estimated a dose of 1100 man-rem for the removal phase.

2.6.1.3 Installation of Repaired Steam Generators

The installation phase involves bringing in and installing the new lower shell assemblies, 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. This yields a total estimated exposure of 1800 man-rem for the installation phase. The licensee has estimated the exposure for this phase to be 569 man-rem per Unit. The PNL estimate assumes worker occupancy in higher radiation fields than those estimated (based on plant surveys) by the licensee.

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 39 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 1301 man-rem per Unit will be expended for the repair program. This estimate is based on dose rate survey data from the Turkey Point reactors, estimates of man hours involved for the individual procedures and estimated savings from dose rate reduction techniques as addressed previously. In addition, the licensee estimated a range of exposures from 650-1450 man-rem/Unit based on uncertainties regarding job man-hours, radiation fields and the effectiveness of temporary shielding. PNL (NUREG/CR-0199) has estimated a total dose of 3380 man-rem per Unit for the whole repair program.

The licensee has committed to making every reasonable effort to keep radiation exposures ALARA in accordance with 10 CFR Part 20.1(c). The radiation protection program followed during the repair effort will be in accordance with the FPL Health Physics Manual and its implementing procedures.

The FPL plant procedures contain sections relating specifically to health physics, including such items as protective clothing, personnel monitoring, radiation surveys, use of temporary shielding and treatment of contaminated personnel. The licensee has stated that the Health Physics Manual reflects a management commitment to maintain occupational exposures ALARA and that the plant Health Physics Supervisor is responsible for ensuring that the ALARA policy is implemented.

The licensee has stated that additional facilities will be provided for the repair effort, including a radiological protection training facility and an additional health physics area.

All craft personnel will be required to take training in radiological protection. The course will include instructions and demonstration in radiation protection principles, theory and practice, emergency planning and the FPL Radiological Protection Program. Personnel will be required to pass a comprehensive examination to have unescorted access in the radiation controlled area. Those failing to pass the exam or those who take only a short basic course will need an escort in the controlled area.

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

Other models include a scale model of the steam generator internal's details and a model of laydown space requirements inside containment. We have concluded that use of the models will be a helpful tool in planning an efficient repair program and will serve to reduce occupational exposures by reducing potential occupancy in radiation fields.

The licensee has stated that full scale mockups will be used to train craft personnel in steam generator cutting and welding operations. This training will serve to reduce occupational exposures by familiarizing personnel with the operations which should reduce the time spent for the actual operation.

The licensee has stated that use of temporary shielding and local decontamination will be evaluated on an individual job basis. The man-rem expenditure for installing and removing the shielding will be compared to the man-rem savings when using the shielding to determine the value of the shielding.

Low radiation background areas will be established inside the containment. Personnel not engaged in an activity will be required to wait in these areas in order to keep their exposures low.

The licensee has stated that special tools such as remote equipment for cutting and welding will be used whenever possible. Use of remote and automatic tooling will save exposure by reducing personnel man-hours to perform the job, allowing personnel to keep away from high radiation sources and allowing personnel to remain behind shielding to keep their exposures low.

Decontamination can be an effective dose reduction technique because radiation fields can be significantly reduced. However, several factors must be considered when 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 be decontaminated using mild solutions. This should provide worthwhile radiation exposure reductions for several of these areas. The licensee will evaluate the use of local decontamination wherever dose reduction benefit can be gained similarly to the evaluation for use of temporary shielding.

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 radioactivity 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. The major 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. All containment releases will be exhausted by the purge system via the plant vent. Releases will be monitored by the

existing sample station and monitor on the plant vent. 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 1.1×10^{-2} Ci of radioactivity 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 cutting kerfs. This activity will pass through the local HEPA filters to the containment atmosphere and then through the containment purge exhaust system to the environment. Although the HEPA filters will be purchased to a removal efficiency of 99.97%, a filter efficiency of 99% was assumed for the filters. We have independently estimated 0.27 Ci may be generated locally by cutting of the RCS piping resulting in a release of 2.7×10^{-3} Ci to the environment assuming a 99% efficiency for removal of particulates by the local HEPA filter. The difference between the licensee's estimate and the staff's estimate is due to the assumption of a different size cutting kerf. Our estimates are based on the information given by PNL in NUREG/CR-0199. In addition, PNL has estimated that 8.1×10^{-3} Curies may be released from secondary system piping cuts. We, therefore, estimate the total release for pipe cutting for removal of three steam generators to be 1.1×10^{-2} Curies. These projected releases are less than the actual average airborne radioactivity releases during 1976 and 1977. For 1976 these releases were 3.8×10^{-2} Ci of

particulates and 0.3 Ci of halogens. During 1977, they were 2.6×10^{-2} Ci of particulate activity and 0.7 Ci of halogens per Unit.

The estimated gaseous radioactive effluent resulting from the repair effort are small compared to Turkey Point historical data. The projected airborne releases from the steam generator are expected to be well below the station radiological effluent technical specifications. The licensee has submitted information to show conformance with the design objectives of Appendix I to 10 CFR Part 50. The staff has not completed its evaluation of this information at this time, however, comparisons with the staff's evaluation given in the Final Environmental Statement (FES)(July, 1972) for Turkey Point indicate that the doses will be less than the Appendix I design objectives. The FES doses are based on releases of 0.8 Curies per year of iodines and particulates and over 3600 Ci per year of noble gases, which are much greater than the projected releases from the repair effort. Therefore, we conclude that the releases will be within the Appendix I to 10 CFR Part 50 Design Objective and 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 licensee is planning to store the reactor coolant for reuse after the repair is complete. Therefore, there should be no release to the

environment from reactor coolant. However, the licensee has estimated a release if it becomes necessary to discharge the coolant. The licensee has stated that the reactor coolant will be treated by the chemical and volume control system prior to release to the environment. The licensee has estimated that a maximum of 0.08 Ci of mixed fission and activation products may be released from the reactor coolant system.

The licensee has stated that if reactor coolant water is discharged it can be processed through a mixed bed demineralizer and the boric acid evaporator. Based on the reactor coolant system activities given in Table 2-2 and the decontamination factors given in Table 1-3 of NUREG-0017 (April 1976), we have estimated the release to the environment of 2×10^{-2} Ci from discharging the reactor coolant system. Actual releases will depend upon actual coolant concentrations at the time of the outage and use of the equipment. The plant technical specification requirements regarding liquid effluents must be met during the repair effort.

Secondary coolant water may be contaminated 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 relatively low and would not contribute significantly to the total activity released.

Local decontamination will be used to lower radiation levels in the plant. The licensee has stated that decontamination wastes are expected to be minimal and will be treated as part of the normal liquid radwaste processing stream. Wastes will be collected and sampled and processed or discharged as dictated by the plant technical specification.

The major volume of liquid radioactive effluent releases will be from laundry waste water. The licensee's maximum estimates are based on 22,000 gallons per day being generated and released during a 300 day outage. The waste water is expected to be of low specific activity and should not require processing before release. However, it must be sampled to verify it is low in radioactivity concentration. 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 0.47 Ci per Unit with Co-60 making up 27 percent of the total activity and Co-58 making up 36 percent of the total activity. The licensee has stated that he expects only 10,000 gallons per day will be released, thus, the total activity released should be only .20 Ci per Unit.

The licensee has estimated a total maximum liquid release of 0.55 Ci of radioactivity (except tritium) and 6.6×10^6 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 2.4 Ci. Our estimate is based on the radioactivity releases given in Table 2-20 of NUREG-0017 (April 1976) adjusted for the licensee's maximum estimated release volume. For comparison, the average release of mixed fission (not including dissolved noble gases) and activation products was 4.3 Ci of radioactivity in 1.7×10^7 gallons per Unit in 1976 and 4.5 Ci in 1.3×10^7 gallons per Unit in 1977.

The estimated liquid radioactive effluent resulting from the repair effort are small compared to Turkey Point historical data. The plant technical specifications limit the radioactivity in liquid effluents from Turkey Point Units 3 and 4 combined to 20 Ci per calendar quarter (excluding tritium and dissolved gases). Consequently, the projected releases will be well within the station technical specification limits.

The licensee has submitted information to show conformance with the design objectives of Appendix I to 10 CFR Part 50. The staff has not completed its evaluation of this information at this time. It is not expected that the liquid effluent limits will be reduced by this evaluation such that releases from steam generator repair activities will be well within the Appendix I design objectives for liquid effluents.

2.6.5 Solid Waste

Radioactive solid wastes generated during the repair effort will include contaminated building materials used to construct temporary structures, concrete removed during the repair, miscellaneous piping, disposable protective clothing, solidified liquid wastes and portions of the upper internals 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 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 1600 ft^3 of concrete will be removed per Unit with a total activity of 3.1 μCi. PNL's estimate (NUREG/CR-0199) agrees with the licensee's.

A major volume of solid radioactive waste will be rags, trash, disposable protective clothing and miscellaneous tools and building materials. The licensee has estimated about 25,000 ft^3 of such

waste containing approximately 100 Ci of radioactivity will be packaged and shipped to a burial facility. In addition, the licensee has estimated 30 Curies of activity will be contained in evaporator bottoms and spent resins.

The licensee has estimated the repair of one Unit will result in a total solid waste volume of 26,000 ft^3 containing 130 Curies being shipped to a licensed burial facility. The licensee's estimates are based on typical quantities and types of wastes generated during a normal refueling outage. 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. This compares with the average amount of radioactive solid waste shipped (per Unit) of 25,400 ft^3 and 240 Ci during 1976 and 19,000 ft^3 and 210 Ci during 1977. 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 dispositon 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. Cutting of the assemblies and subsequent handling would result in increased occupational exposures due to the activity on the surfaces exposed to reactor coolant. Some dose reduction could be achieved by decontamination of the reactor coolant surfaces. However, effective decontamination factors may not be achievable due to presence of a significant number of plugged tubes which would prevent decontamination chemicals from entering approximately 19% 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 the radiation has decayed to levels that will allow easy disposal (e.g., Unit decommissioning). Based on decay of the expected radioactive corrosion products it is estimated that storage for 30 years can reduce the radiation levels to less than 1% of those expected when the assemblies are removed from containment. The assemblies will be sealed with steel plates or plugs prior to removal from containment to eliminate airborne particulates from being released from internal surfaces. Internal decontamination will not be necessary because of the seals. Some surface contamination will be present on the outside of the assemblies. The licensee has stated that the external surfaces will be decontaminated such that removable contamination levels will be less than 2200 dpm/100 cm^2 prior to removal from containment. Therefore, any release to the environment from transport of the assemblies to the onsite storage facility should be negligible.

The onsite storage facility will be a concrete structure approximately 110 ft x 60 ft with a height of 17 ft. The outside walls will be approximately 2 ft thick. The facility floor is earthen with no provisions for collecting water. No water accumulation is

expected since the roof is watertight and the generators will be drained prior to storage. Because the external contamination levels will be <2200 dpm/100 cm^2 airborne releases from the external surfaces of the generators are not expected. The licensee has proposed quarterly surveillance of the facility consisting of visual inspections and random swipes of the generators and area radiation surveys to assure that no airborne contaminants are being released from the facility. There will be a limited amount of direct radiation which penetrates the storage building walls. Based on the maximum expected radioactive inventory of the steam generators and the shielding of the storage facility 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 on contact with the outside of the facility walls are approximately the levels for unrestricted areas specified in 10 CFR Part 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.

We have reviewed the licensee's proposed surveillance program for the storage facility and find it acceptable. We conclude that the program will provide adequate assurance that there will be no releases from the storage facility.

The use of an onsite storage facility will minimize occupational exposures since no immediate disassembly and packaging for equipment 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 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.

3.4 Radiological Consequences of Postulated Accidents

3.4.1 Accidents During Operation with Repaired Steam Generators

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

The reactor coolant system parameters which affect the main steam line failure accident also remain unchanged. Assuming the same

concentration of radionuclides (pre-existing inleakage of reactor coolant), the increased mass of the secondary side will result in a slight increase in offsite doses. The contribution to the doses from additional reactor coolant inleakage during the accident itself would be unchanged. Because the secondary volume increases by 5 percent and most of the dose is a result of "fresh" reactor coolant inleakage, the total offsite doses will increase by much less than 5 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 March 15, 1972 Safety Evaluation for the Turkey Point Plants are not changed due to the steam generator's repair.

3.4.2 Accidents During Repair

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

The licensee has also evaluated the potential for a steam generator being dislodged from the rigging and striking the radwaste or fuel handling building. The licensee has stated that both buildings are capable of withstanding all postulated impacts with no breach of integrity.

We have evaluated the licensee's report and concur with his conclusions. Therefore, there will be no radioactive release to the environment from these construction related accidents.

The licensee has analyzed the potential consequences of rupturing the steam generator boundary due to mechanical shock. He concludes that even if the primary side boundary is breached, the tenacious nature of the corrosive film would result in insignificant releases to the environment.

We have independently analyzed the potential consequences of a steam generator drop. We have assumed that dropping of a replaced steam generator could 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. Radioactivity which may become loosened due to the drop will mostly be deposited on the large surface areas inside the steam generator lower plenum because there will be little air movement between the steam generator internal air spaces and the outside atmosphere. Consequently, 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 0.02 rem to the lung. The assumptions used in the calculation and the results are given in Table 3.4-1.

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)	5.5×10^{-5}
Lung Inhalation Dose Conversion Factor ** ($\frac{mrem}{pCi}$)	7.46×10^{-4}
Breathing Rate ($\frac{m^3}{S}$)	3.47×10^{-4}

Radiological Consequences of Postulated Steam Generator Drop

	Lung Dose (Rem)
Site Boundary	0.02

* All activity is assumed to be Co-60.

** From Regulatory Guide 1.109.

MAR 29 1979