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AUTH.NAME AUTHOR AFFILIATION
ZIMMERMAN,S.R. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Forwards revised reponse to request for addl info re final steam generator rept discussed during 830506 meeting. Responses to telescopied questions on 830323 also encl.

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CP&L

Carolina Power & Light Company

July 15, 1983

Serial: LAP 83-320

Director of Nuclear Reactor Regulation
Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
REQUEST FOR ADDITIONAL INFORMATION
"FINAL" STEAM GENERATOR REPAIR REPORT
REVISION 1

Dear Mr. Varga:

Please find attached Carolina Power & Light Company's (CP&L) revised response to your request for additional information regarding the "Final" Steam Generator Repair Report for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2) dated March 21, 1983. Included with this response are CP&L's responses to questions regarding this same subject which were telecopied to CP&L on March 23, 1983.

In a meeting with members of the NRC Staff on May 6, 1983 and in subsequent conference calls, CP&L's draft response to these questions was discussed and NRC Staff provided clarifications to some of their requests. These responses have been further updated as appropriate with the information available at this time. This response supersedes our previous response to you dated June 3, 1983, Serial: LAP-83-206.

Should you have any questions regarding this information, please contact a member of my staff.

Yours very truly,

SRZimmerman

S. R. Zimmerman
Manager
Licensing & Permits

DCS/lcv (6959DCS)

Attachments

cc: Mr. J. P. O'Reilly (NRC-RII)
Mr. G. Requa (NRC)
Mr. George F. Trowbridge, P.C.
Mr. Steve Weise (NRC-HBR)
Dr. David L. Hetrick (ASLB)
Myron Karman, Esquire (NRC-ELD)
Dr. Jerry R. Kline (ASLB)
Karen E. Long, Esquire (PS-NCUC)
Mr. Morton B. Marqulies (Chm.-ASLB)
Mr. B. A. Matthews (Hartsville Group)
Mr. John C. Ruoff (Hartsville Group)

A001 1/1

**Carolina Power & Light Company Response
to NRC Request for Additional Information
H. B. Robinson Unit No. 2
Final Steam Generator Repair Report**

1. Section 3.4.1.1 "Decontamination" of the report states that two different decontamination methods were being evaluated (fill and soak, and mechanical) for primary surface decontamination. Indicate the method you selected and decontamination solution chosen.

CP&L Response

A general specification for the decontamination of steam generator channel heads was prepared. This specification was sent to various contractors with a request for bid.

Following a review of the bids received, Carolina Power & Light Company (CP&L) has determined that the Westinghouse Electric Corporation presented the best overall method of decontamination. The method chosen, that was used at Turkey Point, is an alumina grit decontamination system utilizing a high pressure water-grit spray that impinges upon the surface of the channel head and abrades away the radioactive oxide film.

The system chosen incorporates a radioactive waste handling subsystem with appropriate shielding for processing the slurry waste water. Waste water may be either recycled or disposed of as radwaste following processing.

2. Section 3.4.7.1 "Radioactive Waste Volume and Activity" of the report states that various solid radwaste volume reduction techniques are being investigated. Describe in detail the technique selected including volume reduction ratio, solid radwaste process control program, activity content, and temporary storage area.

CP&L Response

Planning to determine the most appropriate volume reduction program for the steam generator outage has not been finalized at this time. Box compaction, offering a volume reduction ratio of approximately 4:1, is the most likely choice. Temporary storage areas have been identified on a preliminary basis. The overall radwaste program description, including volume reduction, methods, procedures and equipment will be available for review on November 1, 1983.

3. What are the processes and procedures to be used to preclude the probability of foreign materials entering the steam generators during construction activities?

CP&L Response

Special procedures will be developed to insure cleanliness and preclude the entry of foreign parts or objects into the steam generator components during disassembly and reassembly. Physical barriers, most probably constructed of sheet metal, with access control will be installed to prevent any inadvertent entry when the steam generators are open. Sign-in and sign-out logs for personnel, tools and equipment will be maintained by access control. Tube sheet protection will be installed prior to installation and special procedures will address removals and inspection hold points as required. Tube bundle and downcomer openings will be protected and access control established at secondary manways for controlling entries as noted above when systems are open. Prior to and during removal of these physical barriers, access control will be established to verify that no foreign objects enter those systems as they are being opened.

4. What are the processes and procedures to be used to conduct search and retrieval inspections of the secondary side of the steam generator to assure cleanliness and absence of foreign materials after construction is completed?

CP&L Response

Prior to putting the unit back into service, a boroscopic inspection of the secondary side of the steam generator will be conducted to insure that no objects have gained entry or remain in place. Mechanical severance of the secondary piping systems and immediate plugging will be a requirement of the procedures and physical inspection hold points will be required to fit-up and weld-out.

In addition, contractor personnel will be trained relative to CP&L cleanliness policy and procedures. QA surveillance personnel will insure that all procedures including hold points are strictly adhered to.

5. Describe whether any areas not previously disturbed during site preparation and plant construction will be needed to effect steam generator repairs.

CP&L Response

All of the areas which will be used during steam generator replacement at one time were used for laydown storage areas during initial plant construction. Many of these areas have become wooded since plant construction and are currently being cleared to provide storage areas, building locations, laydown areas and space for the steam generator storage compound. An existing rail spur which was installed during initial plant construction will be either removed or covered over to accommodate a heavy haul road from the equipment hatch to the storage compound. A small plot of wooded territory has been cleared to provide contractor vehicle parking during the steam generator effort.

6. Discuss any change in the amount of demineralizer wastes or steam generator blowdown discharged during operation of the new steam generators?

CP&L Response

According to the steam generator manufacturer, the blowdown rate should stay about equal to the present flow rate (25 gpm/generator). The existing system will be modified to allow for higher blowdown flow rates during startup or as steam generator bulk water chemistry dictates. The higher flow rates will be for approximately 15% of the plant operating time. This rate will require the makeup water treatment system to operate a longer period of time. This longer operating period will increase regeneration waste amounts 15%.

7. Discuss any changes to the NPDES waste permit anticipated as a result of this action?

CP&L Response

There are no changes required to the NPDES waste permit.

8. Provide a description of the treatment and disposal of steam generator blowdown.

CP&L Response

The treatment and disposal of steam generator blowdown fluid from the higher capacity blowdown system will be no different than the present system. The present system passes the blowdown stream through heat recovery exchangers, into a flash tank and then through a radiation monitor before being discharged to the cooling lake.

9. Describe the areas/components that will be decontaminated and subsequently placed back in service as referenced in section 3.5.5 of the Steam Generator Repair Report. Describe the decontamination process including the decontamination fluid. Describe the tests that have been performed to show that decontamination fluids are benign and will not cause future corrosion.

CP&L Response

See response to question 1.

10. Provide the details of the lower steam generator assembly sealing prior to storage as related in Section 4.0 of the repair report discussion concerning on-site storage of components. Address the thickness of the seal plates and welds and the preparation of the interior of the assembly, i.e. drying, gas cover, etc.

CP&L Response

The steam generator (SG) lower assemblies will be sealed by metal boundaries before the assemblies are removed from inside containment. While the details of the necessary metal boundaries and installation procedures have not yet been finalized, they will consist of top and bottom SG lower assembly shield plates and probably plug welded pipes at locations where connections to the shell will be severed. These latter connections consist of one 1-inch shell drain, two 2-inch blowdown connections, and one 3/4-inch level tap. The secondary side handholes above the tubesheet are planned to be internally sealed, as access to the handhole bolting is required for the SG handling equipment.

The design requirements for the shield plates for the upper (transition cone) and lower (tubesheet/channel head) ends of the SG assemblies are as follows:

1. Geometric and material compatibility.
2. Leak-tight construction.
3. Combined installed weight of 35,000 pounds or less.

Conceptually, the shield plates may be about 3 inches thick and of a diameter slightly larger than the mating SG diameter. The plates may be installed using prefabricated curved steel segments which would be fillet welded to the SG and shield plate after fitup of the plate to the SG.

The upper shield plate will probably be a "hat" configuration to accomodate the tube bundle and the changing diameter in the core area. It would be installed shortly after removal of the upper steam dome. Provisions will be made for air ingress while subsequently draining the water (retained for shielding purposes) from the secondary side.

The channel head interior will be decontaminated at the beginning of the project. After separation of the tube bundle from the channel head, the SG lower assembly will be lifted to the operating deck where the lower shield plate will be installed. Upon completion of sealing of the SG assembly, it will be removed from containment.

11. Provide the secondary water chemistry control program including practices and changes to minimize future corrosion of the SG tubes. (EPRI, NP-2704-SR, Special Report, October 1982, "PWR Secondary Water Chemistry Guidelines" provides a part of the overall basis for Staff review on the subject.)

<u>CP&L Response</u>

The secondary water chemistry control program for operation with the new steam generators will incorporate operational and chemistry control practices such that the intent of EPRI and Westinghouse PWR secondary chemistry guidelines will be met. Steam generator protection will be provided by adherence to control limits and action levels for concentrations of contaminants known to cause tube corrosion in the steam generators. These limits and action levels will be provided for cold shutdown, hot standby, and power phases of operation. Also for these phases, feedwater chemistry control limits and action levels will be specified for the effective control of corrosion in this part of the secondary cycle, reduction of corrosion product transport to the steam generator from the feedwater, and control of contaminant input to the steam generators. Control limits will initiate investigative and corrective action, and action levels will be set beyond selected normal limits to minimize steam generator tube corrosion.

As appropriate, action levels will include limits on power operation when such operation will effectively minimize steam generator tube corrosion. For example, power operation action levels will be specified for the following steam generator bulk water parameters: sodium, cation conductivity, chloride, and oxygen. EPRI and/or Westinghouse-recommended concentration values will be specified, as appropriate, for the steam generators.

Control limits and action levels will be monitored through sampling and analysis of feedwater and steam generator blowdown. Continuous instrumentation and laboratory analysis programs will provide rapid identification and diagnosis of system problems. Methods and sampling frequency will be specified to ensure that EPRI and Westinghouse guideline objectives are met.

12. Locked Rotor Analysis

Justify using the ENC analysis as the base case of the safety evaluation of the locked rotor accident, and clarify the differences between the two analyses. Also justify the assumption that the non-affected RCPs keep operating, and that LOOP does not occur. In particular justify how the locked rotor analysis complies with GDC-17. The results of sensitivity analyses regarding the effect of LOOP should be provided if available.

The FSAR locked rotor analysis provided more conservative results than the ENC analysis. Thus, the FSAR values for peak pressure are 2440 psia for three loop operation and 2540 psia for two loop operation, while the ENC value for three loop operation is 2332 psia. ENC did not perform an analysis for two loop operation. The FSAR was more specific than the ENC analysis in terms of the results provided for DNBR and clad temperatures. For the DNBR analysis, the FSAR provided plots of minimum DNBR achieved by 90% and 95% of the fuel rods, and the hottest fuel rod versus time. Slightly less than 10% of the rods reach a DNBR lower than 1.3 for both two and three loop operation. The ENC analysis provides a plot of DNBR versus time, showing a minimum DNBR of 1.4, but is not specific regarding percentage of fuel rods represented. We would normally assume this to be the 95/95 DNBR. The FSAR provides plots for hot spot clad temperatures versus time, which show a steep increase during the transient. The peak clad temperature is 1810°F and a small amount of zirconium reacts with water. The ENC analysis evidences almost no rise in a plot of what appears to be average clad temperature versus time. Neither analysis assumes loss of offsite power (LOOP) in accordance with GDC 17 and both assume that all but the seized reactor coolant pump (RCP) keep operating.

CP&L Response

The Westinghouse (W) locked rotor analysis for the H. B. Robinson FSAR was performed in 1968. The GDC-17 was not issued until 1971, therefore the W analysis could not have incorporated the GDC-17 assumption of loss of offsite power (LOOP).

For your information, however, it has been W's experience with 3 loop plants that a loss of offsite power in the locked rotor analysis shows the peak reactor coolant pressure to increase by less than 20 psi and the peak clad temperature to increase by less than 30°F above the case without loss of offsite power.

The following is a comparison of the Exxon (ENC) and W analysis.

1) MDNBR Comparison

For the original FSAR analysis, a heat flux hot channel factor (F_Q) value of 3.23 and an enthalpy rise hot channel factor ($F_{\Delta H}$) value of 1.77 were used, resulting in an initial DNBR value of 1.63 and an MDNBR value of 0.82 occurring two seconds after transient initiation. The corresponding ENC values in ENC's most recent analysis (Reference 2) are 2.62 and 1.58 for F_Q and $F_{\Delta H}$ respectively. These values are closer to, yet still bound, the H. B. Robinson Technical Specification limits of 2.2 for F_Q and 1.55 for $F_{\Delta H}$. The ENC values used in Reference 2 result in an initial DNBR value of 2.3 and an MDNBR value of 1.58. Therefore, the difference in the FSAR and Exxon calculated MDNBR values is attributed to the unreasonably high peaking assumed in the FSAR analysis.

2) Cladding Temperature Comparison

The low MDNBR value of 0.82 calculated by the FSAR analysis implied the occurrence of DNB at the hot spot. The 1.58 value for MDNBR calculated by ENC is considered more accurate and it implies DNB does not occur and hence cladding temperatures will not differ substantially from coolant temperatures.

3) Primary System Pressure Comparison

The ENC PTSPWR code calculations for the locked rotor event as given in References 1 and 2 predict conservatively low system pressures which result in conservatively low Exxon generated MDNBR predictions. Since Exxon fuel will not significantly affect the system pressure transient for this event, and the FSAR values are substantially below the design limits, system overpressurization was not directly addressed by ENC.

4) Loss of Offsite Power (LOOP) Consideration

The Exxon analysis of the locked rotor transient for H. B. Robinson Unit 2 followed the original licensing bases presented in the FSAR. Therefore, no LOOP was considered for the locked rotor analysis.

If LOOP were to occur in accordance with GDC-17 and the two operating RCS pumps coasted down, the core coolant flow two seconds into the transient, the time at which MDNBR occurs, will be roughly 55% of the initial full flow value. This value is roughly 82% of the value calculated in the ENC analysis for the case where no pump coastdown in the unaffected loops occurs. The effect of this flow reduction would lower the MDNBR value to approximately 1.3. This result is still less severe than the FSAR, since DNB would still not be predicted and significant cladding temperature rises would not result.

5) Two Loop Operation

Power operation with less than three operational reactor coolant pumps is prohibited by the HBR2 Technical Specifications.

REFERENCES

1) "Plant Transient Analysis of the H. B. Robinson Unit 2 PWR for 2300 MWt", XN-75-14, Exxon Nuclear Company, Inc., Richland, WA 99352, July 15, 1975.

2) "Review of Plant Transient Analysis for Positive Moderator Temperature Reactivity Feedback for the H. B. Robinson Unit 2 Nuclear Power Plant", XN-NF-79-42, Exxon Nuclear Company, Inc., Richland, WA 99352, June 1979.

13. Steam Line Break (SLB) Analysis

As noted in our SER for the HBR-2 Cycle #9 fuel reload, the ENC SLB model appears deficient in not considering asymmetric core temperatures, nor the mass input and primary system cooldown due to accumulator actuation or SIS input. We require that the licensee provide additional information that justifies the adequacy and conservatism of the ENC model utilized in the SLB analysis prior to the next refueling.

CP&L Response

In CP&L's letter to Mr. Steven A. Varga (NRC) from Mr. L. W. Eury (CP&L), dated July 23, 1982, CP&L committed to provide additional information that justifies the adequacy and conservatism of the model utilized in the steam line break analysis prior to the next refueling. This information will be provided with our Cycle 10 reload license application by November 1, 1983.

14. Discuss the radiological consequences of accidents that could take place during the repair effort from drops, solution leakage, and accidental destruction of filters during cutting operations.

CP&L Response

1) Drops

Every effort will be made to minimize the potential for drops both inside and outside of the containment building. The movement pathways of the old lower assemblies were identified on Figure 3.1-3 in the Final Steam Generator Repair Report. The movement pathways of the new lower assemblies inside containment will be the reverse of those depicted on this drawing. These pathways will be traversed only when the reactor is defueled, the reactor cavity drained, the reactor head is in place on the reactor vessel, the missile shield is in place, and double isolation is achieved between the reactor cavity and the spent fuel pit. The primary system piping within the postulated drop zone will be drained and inoperable. With the exception of the reactor vessel, all vessels located inside of containment building which may contain radioactive effluents while the steam generators are being replaced are located outside of the containment polar crane wall and are not within the impact area of a postulated drop. Therefore, the potential for large radioactive liquid releases have been kept to a minimum. During handling it should be noted that the old lower assemblies will be dry (i.e., drained) and welded shield plates will be installed on both ends. In addition to this welded seal, the containment building will be maintained under negative pressure by venting through the normal containment vent system. If an old lower assembly was dropped and this seal was breached, this would still not be the cause of any significant radiological releases outside the containment building as it would be contained by a HEPA filter system and monitored by the vent monitoring system. It should be noted that the safe handling of the lower assemblies will be paramount for reasons of personnel safety and economic considerations.

2) Solution Leakage

Precautions to prevent or minimize solution leakage inside containment while decontaminating the channel heads have been addressed. The following, as a minimum, will be invoked upon the channel head decontamination contractor.

a) Drip pans will be provided for all pumps and filters.
b) A seal system will be added to the channel head nozzles to isolate the channel head from the remainder of the primary system.
c) Operators engaged in the use of steam generator decontamination equipment will be trained in the operation of this equipment and will be instructed as to the accomplishment of the required decontamination.
d) The responsible operators of the decontamination equipment will be required to be in communication with each other.
e) All personnel engaged in the decontamination process will be instructed as to the hazards associated with the operation of radioactive liquid systems.
f) All operators will be required to check (visually) any and all high pressure lines and hoses to ensure there are no sharp kinks or bends in the system.
g) Prior to operation a dry run will be conducted to ensure the integrity of the complete deconning system.

In the event that there is a leak detected, the decontamination contractor will be instructed to close the system down and seal off all openings.

The existing containment drain system is designed for and capable of handling any liquid effluents which may leak due to the decontamination operation. The containment ventilation system will maintain a negative pressure inside containment and will monitor and control any radiological releases to the outside of the containment building.

3) <u>Accidental Destruction of Filters During the Cutting Operations</u>

As noted in CP&L's Final Steam Generator Repair Report, an extensive effort will be put forth to control airborne radioactivity. It was noted in the report that in addition to bulk containment atmosphere control of airborne activity, appropriate localized control will also be provided as necessary using temporary enclosures and HEPA filtration units. Personnel working in areas of potential airborne contamination will wear respiratory equipment as required. In the event that there is accidental destruction of a filtration unit during a cutting operation, the following will prevent any adverse radiological consequences:

The localized controls around the cutting operations of the lower assembly channel heads will consist of tented enclosures. These enclosures will be maintained under negative pressure by separate HEPA filtration units. If there is a failure in this localized ventilation system the only consequence would be the escape of airborne radioactivity to the ambient containment atmosphere. This containment atmosphere is controlled by the normal containment ventilation system which would require all radioactive airborne materials to pass through the HEPA filtration system which is built into it. Locally installed monitors would detect this breach of the localized containment, and alert surrounding personnel of the potential airborne situation. This situation would be handled with the existing Health Physics' procedures already in effect at H. B. Robinson Unit No. 2 Plant.

15. Regulatory Guide 8.8 recommends preparation and planning actions be completed before workers enter radiation areas (Section c.1.b, c.3.a). Verify that the planning and preparations will be completed prior to the initiation of the steam generator replacement task for the following ALARA measures planned for the steam generator replacement task:

a. general area decontamination

b. primary surface decontamination

c. use of temporary shielding

d. use of specialized tools

e. removal of selected valves and piping

f. establishment of low background wait areas

g. establishment of laydown areas

h. training for plant and contractor personnel

i. access control

j. equipment decontamination

This verification should ascertain that outage sequences will be or have been reviewed by ALARA coordinators prior to work initiation to determine the specific applications of the above measures and that manpower, materials, and work direction will be planned and committed. By example, laydown and "wait areas" should be clearly identified and prepared for the task start including provisions for decontamination, temporary shielding, posting, and access. Radiation, contamination, and airborne radioactivity surveys will be conducted as necessary to determine radiation protection measures.

CP&L Response

a. Planning for the deconning of the general area of the containment is proceeding.

   Generally, all areas in containment where work activites are conducted will be deconned to less than 1000 dpm/100 cm^2 if possible. Specific work areas and decontamination methods have not been identified.

   Presently, a decon crew is maintained onsite for general area deconning and radwaste package preparation. An expansion of this crew similar to what was employed at Turkey Point is planned for the SG repair outage. A specific work package for decontamination will be prepared by November 1, 1983.

b. Specifications for deconning the steam generator channel heads are written and have been sent to contractors with requests for bid. The channel heads of the three steam generators will be deconned during the SG repair program. See reponses to question 1 above.

c. Extensive radiation surveys were made during the April-May, 1983 mid-cycle outage. Decisions will be made concerning the type, amount, and location of temporary sheilding. Procedures are currently being written for installation, tracking, and removal of this temporary shielding. To the maximum entent possible specific uses of temporary shielding will be identified prior to the start of the outage.

d. The use of specialized tools for the purpose of personnel exposure reduction is being incorporated where feasible into the task procedures that will be implemented during the steam generator repair project. These procedures including the use of specialized tools will be established and evaluated on the basis of the company's commitment to an integrated system of dose limitation centered on the ALARA philosophy. Although the overall goal of exposure reduction is a bounded constraint of the repair project, the exact nature of the tools is highly dependent on the methodology and contractor that will be selected. It is anticipated that these planning and selection phases will be completed by December 1, 1983 based on a February, 1984 outage start date. At that time a more comprehensive list of tooling will be available.

e. The steam domes and reactor coolant pump motors will be removed from containment for modification in a low exposure rate area. No additional components are scheduled for removal. Exposure rate levels from hot lines or components will be reduced via shielding.

f. As stated in the current H. B. Robinson Steam generator repair report, low exposure rate waiting areas will be utilized by workers not actively engaged in a given task phase. Low exposure rate waiting areas will be provided for workers on the 228 ft. and 275 ft. elevations. The location on the 228 ft. elevation will be adjacent to the personnel elevator while the 275 ft. elevation area will be in the vicinity of the stairwell near "A" steam generator. The areas should be large enough to accomodate 25 people and the exposure rates in these areas should be less than 5 mR/hr. These areas will be re-evaluated as the outage progresses and may be changed should better areas be identified.

g. A comprehensive site master plan for space utilization, including laydown areas, is under development. Exact dimensions of various facilities are being finalized at this time. When this phase is complete, space allocation will be initiated. This will be accomplished in a time frame sufficient to allow necessary site preparation to be completed prior to the beginning of the outage.

h. An ALARA training module has been developed and is presently being presented to CP&L and contract personnel in General Employee Training. This module is scheduled to be modified and restructured by September, 1983. All persons working in radiation control areas will receive this training. Additionally, General Employee Training Level III which is a 40-hour course designed to provide an advanced level of knowledge in radiation protection and plant systems will be implemented on July 18, 1983.

i. The access control system to be employed during the replacement outage will be a manual system similar to the one currently in use at the H. B. Robinson Plant. CP&L's RIMS System will assist in maintaining exposure histories and provide computerized dosimetry records for tracking of radiation doses for individuals and the dose accumulated by the major outage tasks. The design of the dressout facility and the containment checkpoint are essentially complete although additional minor changes may be incorporated prior to the repair outage, no major changes are currently planned.

The dressout facility is a roughly L-shaped building of approximately 7,000 ft^2. The building will contain an undress area, a security station, checkpoints for entrance into the radiation control area (RCA), a dress out area, an HP office, a survey equipment room, and a room for respirator storage. The facility will also contain portal monitors and friskers which will be used by personnel on exit from the RCA.

The dress out facility will be connected to the containment and auxiliary buildings via a covered walk way. Personnel entering the containment vessel will pass through a containment checkpoint to assure that they are properly badged, dressed, and logged. A respirator issue station will also be available at the containment checkpoint. Upon exiting the containment, personnel will undress, frisk out at the checkpoint and pass through the health physics check out area before returning to the dress out facility.

j. In the past, the H. B. Robinson plant has utilized special equipment such as a freon vapor degreaser, a sandblasting glovebox, and a reverse electropolisher for equipment and tool decontamination. CP&L will lease or purchase similar equipment for use during the replacement outage. Specific equipment and facilities have not been selected at this time. A description and evaluation of the systems chosen will be provided by November 1, 1983.

16. Identify the number of portable air sampling instruments available for the replacement task as discussed in Regulatory Guide 8.8, Section C.4.

CP&L Response

Air sampling equipment currently in use at the H. B. Robinson facility include 20 Staplex Hi-Vol Samplers, 9 RADeCO Lo-Vol Samplers, and 3 Pneumotive Continuous Lo-Vol. Samplers. In addition to this equipment, the Robinson Plant has on order 9 continuous air monitors (CAMs).

The CAMs that have been ordered for the H. B. Robinson Plant are RADeCO Models GM 222A-1. These CAM's are compact, semi-portable general purpose monitors for measuring gross beta or gamma particulate airborne activities. The airflow rate is continuously monitored by utilizing a front panel monitored flow meter to measure the pressure drop across an internally mounted venturi tube. The monitor is equipped with both the Hi level and Hi-Hi Level visual and audible alarms as well as a visual fail safe alarm.

The purchase order for the CAMs were received by RADeCO on January 25, 1983. The shipment date for the CAMs has been set for August 1, 1983. As soon as the CAMs are received, calibration procedures will be developed. Once the procedures are in place the CAMs will be calibrated and ready for use. The exact locations of the CAMs within the plant are yet to be determined since their placement will depend on the finalized work layout of containment. However, it is known that at least one CAM will be placed in each steam generator bay. Other areas will be selected by December 1, 1983.

17. Verify that the H. B. Robinson counting facility will be adequate for the anticipated increased surveillance activities as in Regulatory Guide 8.8, Section C.4.

CP&L Response

The H. B. Robinson Counting Room presently has 1 Mulichannel Analyzer (MCA) with 2 GE(Li) detectors for gamma spectroscopy, 1 Liquid Scintillation Counter for tritium measurements, 2 Phoswich Detectors for evaluation of evaporated and filtered samples, 2 gas-proportional α/β smear counters, and 1 Phoswich α/β smear counter.

The H. B. Robinson Plant has on order a second MCA with 1 Ge (Li) detector Ge(Li) system to enhance gamma spectroscopy capabilities.

The plant also has access to the company's mobile counting laboratory as a contingency should extra equipment be needed on short notice. It contains 2 Intrinsic Germanium (IG) detectors and 1 Phoswich detector.

Since the need for additional fixed equipment is highly dependent on programatic development in the area of contamination control the plant health physics staff and support groups will continue to assess the requirements as they develop based on the control programs implemented.

In an effort to alleviate the crowded conditions of the present counting facility, space for counting activities has been alloted in the new E&RC Building. This building and its associated hardware, excluding the calibration facility, is scheduled for completion by December 31, 1983 and presently is proceeding on schedule.

18. Provide a commitment to satisfactorily resolve the outstanding deficiencies noted in Inspection Report 50-261/82-34 prior to commencing the steam generator replacement task.

The deficiencies noted in Inspection Report 50-261/82-34 included the following:

1. More conclusive evidence is needed in ascertaining that high whole body counts are due to skin contamination and ingestion rather than inhalation of airborne radioactive material.

2. No constant air monitors (CAMs) in operation in the Auxiliary Building.

3. Inadequate surveillance of airborne radioactivity in controlled areas prior to work commencement.

CP&L Response

With regard to the deficiencies noted in Inspection Report 50-261/82-34 as stated above, all CP&L actions on items 1 and 3 have been taken and are awaiting NRC review. For item 2, as noted in 16 above, CAMs have been ordered for use.

19. Discuss the use of engineering controls which preclude the need for respiratory protection equipment (e.g. contamination control devices, local HEPA ventilation, flexible ducting, tents) as recommended in Regulatory Guide 8.8, Section c.2.d and identify specific applications.

CP&L Response

Temporary containment systems equipped with Hepa filters will be utilized during the channel head cuts to minimize the spread of airborne contaminants. They will also be utilized during initial portions of the re-welding. Additional uses are undetermined at this time. Other possible uses, however, include:

1. The use of a cover to be placed over the containment equipment hatch to prevent airborne activity from exiting the containment in the

unlikely event that the redundant purge system is lost.

2. The use of tents with HEPA ventilation systems in areas where welding will be done on contaminated equipment or components.

3. The use of portable gloveboxes around pipes and systems where high levels of contamination are expected.

A description of the containment systems and a list of their uses will be provided by November 1, 1983. A portable containment fabrication shop (Tent Shop) will be located on site to provide for rapid and custom construction of portable containments as the need arises.

20. Provide specific commitment to establish a program for ALARA internal and external contamination consistent with Section c.2.d of Regulatory Guide 8.8 in order to reduce the numbers of workers who receive detectable internal contamination, as well as to minimize the number of workers who become externally contaminated.

CP&L Response

The CP&L management commitment to the ALARA program is contained in the Corporate Health Physics Policy Statement and Corporate Radiation Protection and Control Manual. This commitment applies to exposure from all sources of ionizing radiation including both internal and external contamination. Though special effort will be exerted to minimize the number of personnel contamination incidents, CP&L management is committed to this through the existing policy statement and will track and trend these incidents throughout the outage in accordance with procedures.

An ALARA program is currently in place at the H. B. Robinson Plant to maintain personnel exposure to radiation to ALARA levels. This program applies to all sources of exposure, both internal and external. The steam generator replacement project will be conducted in accordance with this program. Planning for the project is currently being reviewed by the ALARA organization and will be followed to completion.

The ALARA organization consists of a full time ALARA specialist and a full time technician assigned to the plant health physics staff. In addition, the ALARA specialist chairs an ALARA committee which is composed of representatives from the various plant sub groups. This committee provides review of plant modifications from an ALARA standpoint to assure proper preparation and planning.

There is also a full time ALARA specialist position in the Construction Department at the H. B. Robinson Plant. When filled, this position will serve as an interface between health physics and construction and will assist the plant ALARA group during the steam generator replacement project.

21. Describe how decontamination facilities, to be provided for the replacement task, meet the criteria of Regulatory Guide 8.8, particularly Sections c.2.f and c.4.e.

CP&L Response

A fixed or portable facility to house dedicated decon equipment and provide a location for deconning tools and equipment is being evaluated. The type of decon equipment and the size and location of this facility will be determined before the SG repair outage begins.

22. Verify that adequate training facilities and training personnel will be available to conduct the committed training prior to initiation of the related tasks. The training program planned by the licensee includes measures to familiarize workers with their tasks, tools, equipment, and operational and radiological procedures by use of job-specific training, dry-run training, and mock-up training. Methods for handling and processing radioactive wastes, and the impacts of these wastes have been evaluated. Radwaste reduction techniques are being investigated.

CP&L Response

A new training facility is under construction and scheduled to be completed prior to the SG repair outage. This facility will have adequate space to conduct all formal classroom training. Additional permanent and/or contract personnel will be available to conduct committed training prior to initiation of related tasks.

Approximately thirty-two individual tasks have been identified that will require job-specific training. A training program made up of modules that meet or exceed CP&L standards will be developed to familiarize workers with their tasks, tools, equipment, and operational and radiological procedures. Dry-run mock-up training will be provided for, but not limited to, channel head workers. The development and implementation of all training will be coordinated through the CP&L Curriculum Development Unit and Robinson Training Unit. A training program for the handling of radwaste will be available for review by November 1, 1983.

The Robinson Training Unit will be responsible for tracking, quality control, and documentation of all training materials related to steam generator repair.

23. Provide a commitment to measure and evaluate the progress of the steam generator replacement task through dose tracking and on-going radiological assessment of specific tasks by radiological engineers/ALARA coordinators as is recommended in Regulatory Guide 8.8, Section C.1 and C.3.

CP&L Response

During the outage, personnel exposures as a function of task will be tracked and updated by the company's computerized radiation information management system (RIMS). In addition task procedures and progress will be monitored by the ALARA specialists in an on going effort to minimize personnel exposures.

24. In order for the NRC Staff to evaluate radiological results of the replacement project, and to determine if additional or different radiological controls need to be considered, the licensee should perform a radiological assessment as follows:

(1) The collective occupational dose estimate shall be updated weekly. If the updated estimate exceeds the person-rem estimates by more than 10%, the licensee shall provide a revised estimate, including the reasons for such changes, to the NRC within 15 days of determination.

(2) A final report shall be provided to the NRC within 60 days after completion of the repair. This report will include:

(a) A summary of the occupational dose received by major task.

(b) A comparison of estimated doses with the doses actually received.

(c) A discussion of ALARA measures employed, and

(d) A summary of decon efforts and radwaste generation.

(3) Interim reports which summarize each 90-day period of the repair effort shall be provided to the NRC within 60 days of the completion of each such period.

CP&L Response

Carolina Power & Light Company will provide the NRC with the information requested in items (2) and (3). This information will be submitted in the time frame specified by the NRC staff in the request.

In response to item (1), the collective occupational dose estimate for the SGRR was derived on a task by task basis. Weekly estimates of collective dose will be established prior to the beginning of the outage based on these task estimates and scheduling requirements. The collective and individual task totals will be updated frequently and tracked in order to provide guidance for subsequent work. Due to the scope of the outage and potential schedular changes based on outage progress, it is impossible to guarantee that these tasks will be performed as originally scheduled. Therefore, actual weekly doses can vary from the estimates due to purely schedular changes. For these reasons, the Company feels that it would be overly burdensome to require the issuance of a report anytime the actual weekly dose exceeds 10% of the weekly estimate. However, the Company does agree to compare the estimated and actual dose on a quarterly basis and to present these comparisons to the NRC in the 90 Day Progress Reports. If at this time the actual collective dose exceeds the estimate by 10%, the collective man-rem estimate will be revised and any increase will be justified accordingly.

25. Describe stress relief heat treatment procedures of welded joints to ensure compliance with ASME Code requirements. Also, indicate how the stresses would be mininized on cladded components during cutting, welding, and stress relief heat treatment.

CP&L Response

Post weld heat treatment (PWHT) temperature shall be 1150°F nominal ± 50°F. A temperature of 1200°F shall not be exceeded. The temperatures for each weld joint are specified for cutting, post heating and postweld stress relief heat treatment in the attached table. (Table 1)

The heat treatment cycles (i.e. duration of heating rate, holding temperature and time, cooling rate) shall be in accordance with the ASME Code Section III for the appropriate P numbers of material involved. During PWHT above 800°F, the rate of heating and cooling in any hourly interval shall not exceed 400°F divided by the thickness in inches of material being heat treated, but shall not exceed 400°F/hour and may not be less than 100°F/hour. During heating and cooling there shall not be a greater variation in temperature than 250°F within a 15 foot interval of weld length. The PWHT method shall be local heating of each weld joint consisting of resistance heating elements located in a circumferential band around the components weld joint.

Thermocouples will be either placed in contact with the material by use of the low energy capacitance discharge method (Ref. Code Case N-266) or placed in blocks in contact with the material.

Recorder time-temperature charts shall be produced for all PWHT operations when the temperature exceeds 600°F. Thermocouples for monitoring the coolant pipe at the steam generator nozzle/pipe weld joint shall be attached on the steam generator nozzle side.

In order to minimize stresses on cladded components, additional thermocouples will be installed and the temperature recorded continually and monitored to ensure maximum temperature ranges are not exceeded. In addition, insulation shall also be placed in areas where temperature limits are critical. See Table 1 attached.

26. Describe the details of the tests and evaluations which will be conducted after the steam generator repair to assure the integrity of the reactor coolant system and compliance with applicable codes and standards.

CP&L Response

The preoperational and start-up test program is being developed but many details remain to be determined. The objective of the test program will be to ensure that the plant is returned to safe and reliable full power operation. The steam generator replacement project will comply with the requirements of Section XI of the ASME Code and, of course, the plant

Technical Specifications. These basic requirements will assure the integrity of the reactor coolant system.

IWB-5222 of Section XI of the ASME code identifies the required test pressure as a function of test temperature. Also, Technical Specification 3.1.2 (Heatup and Cooldown) provides limit curves on the allowable combination of pressure and temperature in the primary system. Because of the technical specification required (pressure) test temperature, the ASME Section XI test pressure would be about 2307 psig. However, Technical Specifications 3.1.2.1.c and 4.3.1 will set the test pressure at 2335 psig for the primary system.

The inpsections during the pressure test will satisfy IWA-5246 of Section XI of the ASME Code and assure that the pressure boundary is acceptable.

27. Describe the preoperational testing program which will be conducted to provide the necessary assurance that the steam generator and other affected components can be operated in accordance with design requirements and in a manner that will not endanger the health and safety of the public.

CP&L Response

The steam generator replacement project is essentially only for the purpose of replacing the three steam generator tube bundles; any other work to be performed is not anticipated to be major. Thus, the preoperational testing will be oriented to this component and will not be as extensive as the testing involved in startup of a plant. The testing will satisfy the H. B. Robinson Technical Specifications.

The testing and inspections will involve cleaning (see Question 30), checkout of the fuel handling equipment, pressure testing (see Question 26), checkout of important instrumentation, and functional testing. The purpose of the functional tests are as follows:

Thermal Expansion Testing; to verify that the steam generators can expand and contract without obstruction during heatup to operating conditions and return to cold shutdown conditions. This testing will also include observation of the affected piping and instrumentation.

SG Water Level Stability And Control Testing; to verify stability of the automatic level control system including step load changes.

SG Thermal Output And RCS Flow Testing; to measure RCS flow using primary and secondary calorimetrics and to measure the SG thermal output at steady-state conditions.

SG Moisture Carryover Testing; to verify that the moisture carryover in the steam leaving the SGs satisfies the design/performance requirements.

28. Provide details of the tube/tubesheet joint in the replacement steam generators.

Following insertion into the tubesheet hole, tack rolling, welding and gas leak testing, the tubes are hydraulically expanded to the full depth of the tubesheet holes. Field experience with operating units in which the tubes were partially rolled into the tubesheet indicated that a full depth expansion, which would essentally close the crevice between the tube and tubesheet, would add margin by minimizing the possibility of crevice corrosion.

A change in configuration was made in the early 1970s to full depth mechanical rolling which sealed the tubesheet crevice up to approximately a quarter inch from the top of the tube sheet. Full depth mechanically rolled steam generators now in operation include CHI 1 and KORI 1; these have operated for approximately five years. In that same time period, a number of steam generators with partially expanded tubes were full depth expanded in the field by an explosive process called WEXTEX. These include Trojan, Beaver Valley 1, Salem 1, Farley 1, and North Anna 1 and 2. Significant operating experience with full depth expansion is provided by these steam generators, operating for as long as seven years in the case of Trojan.

In the mid-1970s, development efforts started on an alternate expansion process that would combine the reduced deformation and the low residual stress transition of the WEXTEX process with the tight sealing of the hard mechanical roll.

Hydraulic expansion was adopted as the optimum and is the reference process for the current steam generator design including the replacement lower assemblies for the Robinson Nuclear Plant. The replacement Surry 2 steam generators are the first operating units with hydraulically expanded tubes. Refinement of the hydraulic process has resulted in expanding all but a small crevice of about one eighth inch average depth at the top surface of the tubesheet.

The benefits of the hydraulic expansion process are the reduction of the cold working caused by the mechanical hard rolling and the lower residual stresses at the transition of the expanded to unexpanded region of the tubes. Analyses and experiments have shown these tensile stresses to be of the order of 20 ksi on the OD surface and 20-30 ksi on the ID, which are about half the stresses for a mechanical roll.

In addition to the change to hydraulic expansion, the tubing material was also changed to take advantage of the increased corrosion resistance of thermally treated Inconel 600 to stress corrosion cracking in both primary and secondary environments. The occurrence of SCC and IGA, which has been observed in some partially expanded units is expected to be minimized by the the combination of the full depth hydraulic expansion and the thermal treatment of the Inconel 600 tubing.

See Figure 2.2-2, Tube-To-Tubesheet Junction, page no. 18 of the FSGRR.

29. Verify that no new postulated piping break locations would result from the replacement of the steam generators and modifications to the affected piping systems.

CP&L Response

The main steam piping is planned to be separated at the applicable steam generator nozzles and another location in the pipeline. The piping will be essentially restored to its original configuration, including the support system, following the steam generator replacement. The work will be done per the requirements of the Power Piping Code, B 31.1.

The main feedwater piping will be separated at the applicable steam generator nozzle and another location in the pipeline. The piping will be essentially restored to its original configuration, including the support system, following the steam generator replacement. The work will be done per the requirements of the Power Piping Code, B 31.1.

As the piping will be essentially restored to its original pre-replacement configuration and the work will be in accordance with the applicable code, no new potential failure locations will be created.

30. As described in section 4.0 of the repair report relating to post-installation activities, provide a commitment to inspect, after hydrotest, the interior of the steam generator to assure metal-clean surfaces in accordance with Regulatory Guide 1.37.

CP&L Response

The H. B. Robinson - Unit 2 Updated FSAR, Volume 1, Page 1.8.0-6, addresses the site applicability of ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants". Regulatory Guide 1.37 (issue dated 3/16/73) modifies this standard.

The steam generator replacement project will be conducted as a "construction" project. However, as this activity affects an operating plant, the project probably will not be started until major components have arrived on site.

The anticipated outage time is about 9 months which is not long compared to the construction time of a power plant. Cleanliness protection of the new steam generator assemblies will be provided during shipment, storage at the plant, and to the extent practical, during the actual installation. For example during the installation, the tubing ends will be plugged and/or covered to prevent welding fumes and contaminants from entering the tubes. At the completion of the construction portion of the project, the steam generator will be inspected to assure that the surfaces are clean prior to returning the unit to operation.

TABLE 1

STEAM GENERATOR REPAIR PROGRAM
HEAT TREATMENT SUMMARY *

	Preheat	Post Heat		PWHT	
	°F (min)	Temp, °F	Time, hrs.	Temp, °F	Time, hrs.
Thermal Cutting					
Upper Girth Rough Cut	250	N/A	N/A	N/A	N/A
Upper Girth Precision Cut	250	N/A	N/A	N/A	N/A
Feedwater Nozzle Cut	250	N/A	N/A	N/A	N/A
Welding Operations					
Upper Shield Plate	See Note 1	N/A	N/A	N/A	N/A
Bottom Shield Plate	See Note 1	N/A	N/A	N/A	N/A
Misc. Closure Devices	N/A	N/A	N/A	N/A	N/A
Upper Girth Joint	300	400-500	4	1100-1200	2.5
Lower Girth Joint	300	400-500	6	1100-1200	3
Cladding Replacement (Note 2)	200	N/A	N/A	1100-1200	3
Main Steam Flow Limiter	300	400-500	2	1100-1200	2
Main Steam Nozzle-to-Reducer	250	N/A	N/A	N/A	N/A
Main Steam Reducer-to-Pipe Elbow	200	N/A	N/A	N/A	N/A
Main Steam Pipe-to-Pipe (maybe 2)	200	N/A	N/A	N/A	N/A
Feedwater Nozzle Extension-to-Remnant	300	400-500	3	1100-1200	2.5
Feedwater Ring Support Lugs (3)	300	400-500	2	1100-1200	2.5
Feedwater Nozzle Extension-to-Reducer	300	400-500	2	1100-1200	1

* Quantities indicated are for each steam generator.
Post Heat is only required when preheat cannot be maintained until PWHT.

Notes:
1. To be addressed pending shield plate design.
2. Only first two (2) cladding passes require preheat.