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4.7 QUALITY ASSURANCE

The quality assurance programs for Consumers Power Company, Bechtel Power Corporation, and Combustion Engineering, Inc. as applied to this project are described in this section.

4.7.1 CONSUMERS POWER COMPANY QUALITY ASSURANCE PROGRAM

The Consumers Power Company quality assurance program is described in the Consumers Power Company Quality Assurance Program Topical Report (CPC-1-A). This report is an integral part of the Consumers Power Company Quality Assurance Program Manual for Nuclear Power Plants and the provisions pertaining to major modifications will be invoked for the Palisades Steam Generator Repair Project. 4

4.7.2 BECHTEL POWER CORPORATION QUALITY ASSURANCE PROGRAM

Bechtel Power Corporation will perform its duties in accordance with Bechtel Topical Report BQ-TOP-1, Bechtel Quality Assurance Program for Nuclear Power Plants. This topical report will be invoked for activities within Bechtel's scope of responsibilities on this project. 4 Responsibility for the Palisades Steam Generator Repair Project has been assigned to the Ann Arbor office.

4.7.3 COMBUSTION ENGINEERING POWER SYSTEM GROUP NUCLEAR QUALITY ASSURANCE PROGRAM

The quality assurance program used by Combustion during the design and shop fabrication of the replacement steam generators is described in CE-NPD-210, Quality Assurance Program - A Description of the CE Nuclear Steam Supply System Quality Assurance Program.

4.8 REGULATORY GUIDE APPLICABILITY TO REPAIR PROGRAM

Section 2.1.4 discusses regulatory guide compliance during the manufacture of the steam generator units. This section discusses regulatory guide applicability to repair program activities other than steam generator manufacture.

a. Regulatory Guide 1.31 (5/77), Control of Ferrite Content in Stainless Steel Weld Metal

   Control of stainless steel welding complies with interim position on Regulatory Guide 1.31 (Branch

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D-1 Section 4.1.4 describes the various types of accidents which might occur while handling the steam generators. However, estimates of the dose consequences of these accidents are not presented. Estimate the dose consequences of 1) accidents which might rupture the old steam generators such as a steam generator drop accident, 2) accidents which might affect safety related equipment such as a crane drop on the diesel generator room or a dropped steam generator rolling into buildings housing safety related equipment, and 3) accidents which might cause the release of radioactivity such as a crane drop onto a storage tank. Explain the assumptions and parameters used in these estimates.

RESPONSE:

The dose consequences of various accidents which may occur while handling the steam generators are addressed below.

1) An accident involving the drop of the old steam generators could be assumed to rupture a closure on one of the primary nozzles. As discussed in SGRR Section 4.4.7, the radionuclides are immobilized in the tenacious corrosion layer, and a significant release from a drop is not expected. For the purpose of analysis, however, a removable contamination level of 10^6 dpm/100 cm^2 was assumed for the steam generator internal primary surfaces. The release of this material, approximately 10 mCi, could not result in an offsite dose of greater than 10^{-3} mrem.

2) The fuel will be removed from the reactor prior to the start of major construction activities inside containment. Once the fuel is removed, equipment inside the containment building will not be necessary to serve a safety function during shutdown. Outside the containment, it is necessary to maintain the integrity of the control room, the spent fuel pool, the spent fuel pool cooling system, component cooling system critical service water system, and the diesel generators during shutdown. The steam generator replacement route and construction activities associated with the replacement are far removed from these safety-related structures, systems, and equipment. Administrative controls will be applied to keep heavy construction objects from infringing upon the safety functions of these systems, components, or structures. Therefore, a radiological accident of this nature is not credible, and a dose assessment is not necessary. (Also see E-9).

3) Other radiological accidents are not envisioned except those addressed under Item 1.

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D-2 Section 4.3.6.3 identifies three potential sources of radioactive liquid wastes. Is the secondary coolant a potential source of release of radioactive liquids? Describe the normal liquid radwaste processing scheme. What is the estimated quantity and radioactivity of each of these wastes? Explain 1) what portions of the releases are due to normal refueling and work outage and 2) what portions are added by the steam generator repair effort? Describe the criteria which will be used to determine if the wastes will be processed through (1) filters, (2) evaporators and/or (3) demineralizers prior to release to the environment. For each of the three potential sources of radioactive liquid, evaluate their environmental impact. The evaluation should include estimates of the doses individuals in the public and the population within 50 miles might receive. Identify the equipment which will be available to process the wastes. Evaluate the possibility of the local decontamination solution(s) having deleterious effect(s) on the equipment employed to process it. Are the concentrations in Table 6.2-3 typical of refueling/major work outages? Include Fe-33 in your release estimates, and commit to sampling for Fe-55 and Ni-63 in liquid effluents during the repair program. During the Surry repair program, Fe-55 and Ni-63 were found in effluent analyses.

RESPONSE:

The secondary coolant is not considered a potential source of release of radioactive liquids. Activity accumulated during the repair effort from steam generator blowdown resins and resins from full flow condensate demineralizers would not be expected to exceed 50 mCi and would be shipped as dry radioactive waste. At present, any liquid releases associated with the steam generator repair outage are identical to the releases associated with a normal refueling or work outage. A newly installed laundry-dry cleaning system will eliminate any additional laundry waste water associated with increased outage work schedules. Local decontamination waste fluids will be solidified in a cement matrix and is not considered a potential source of radioactive liquid release.

The operation of the radioactive waste treatment system is addressed in Section 11 of the FSAR.

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Total body doses to the population and average doses to individuals in the population from receiving water-related pathways to a distance of 50 miles from the site are calculated to be less than 1 manrem and 1 mrem, respectively.

As stated above, local decontamination waste fluids will be solidified in a sodium-silicate cement and will not be treated as part of the normal liquid radwaste processing scheme. Potential deleterious effects on radwaste processing equipment will be addressed when decontamination alternatives are addressed.

The estimated specific activity of laundry waste water as presented in Table 6.2-3 is based on a refueling/major work outage which occurred at the Palisades plant in 1976 and included a full offload of the core.

Analyzing for Fe-55 and Ni-63 in liquids is intended on a semi-annual basis during the repair program.

PALISADES PLANT SGRR

D-3 Your analysis and calculation of the inventory of activated corrosion products in a steam generator presented in Table 4.4-1 is not clear. The estimate of 3.5×10^5 inches of steam generator tube appears to be too low (from the FSAR, 8519 tubes times an average tube length of roughly 650 inches equals 5.5×10^6 inches of tube). Also, it is not clear what is meant by tubes per tube sheet, and the ratio of Co-58 to Co-60 is much higher than that ratio found at Turkey Point. Provide information to clarify these items.

RESPONSE:

External dose rate analysis has indicated that the dose rate decreases as the inverse square of the distance from the tubesheet with dose levels at the top of the tube to be in the low mrem/hr range. Therefore, only the tubesheet area has been used to estimate activity inventory. The steam generator activity presented in Table 4.4-1 assumes that the activity concentrations shown are distributed throughout the tubesheet which extends to 20.5 inches above the inlet and outlet plenums. The 3.5×10^5 inches of steam generator tubing estimate were obtained as follows:

(8,519 tubes/steam generator) (20.5 inches of tube per tubesheet) (2 penetrations for each tube through tubesheet) = 3.5×10^5 inches of tube

The ratio of Co-58 to Co-60 was based upon the isotopic analysis of undescaled 2-inch-long sections of steam generator tubing. The study is listed as Reference 2 in the repair report. In addition, isotopic concentrations were verified by smear surveys and an independent gamma isotopic survey performed in 1979 using a shielded lithium drifted germanium detector.

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D-4 Describe any effects which your modifications to the steam generator, blowdown drain, and sample systems may have on the various accident analyses presented in your FSAR.

RESPONSE:

No values associated with modifications to the steam generator blowdown drain or sample systems will be less conservative than those analyzed in the FSAR.

PALISADES PLANT SGRR

D-5 Provide estimates of the Fe-55, Ni-63 and transuranic content of the "old" steam generators. The estimates should be based on sample analyses similar to those presented in Table 4.4-1.

RESPONSE:

Samples are not available for the analysis of the radionuclides in question. However, if Fe-55 and Ni-63 were major constituents, the two isotopes would not be significant contributors to the radiation dose. Transuranics are not expected to differ significantly from levels observed at Surry.

PALISADES PLANT SGRR

D-6 Your analysis of gaseous releases from cutting reactor coolant piping presented in Table 6.2-1 does not appear to be correct. The equation presented does not yield the answer given and the area of cut values of 0.5 square inches is too low. Estimate the releases from cutting the blowdown, feedwater, and main steam piping using similar analyses. Also, estimate the releases from cutting the construction opening of the containment. Estimate the amount of gaseous radioactivity which will be released from routine refueling/work outage activities such as fuel movement. Also, estimate the added releases from steam generator repair activities such as complete defueling, local decontamination and construction activities within containment.

RESPONSE:

The area of the cut was presented erroneously, but it was used properly to obtain the results. The area of cut presented in the equation from Table 6.2-1 should be as follows:

(0.5 inch) (π) (inside diameter of pipe)

or

(0.5) (π) (30) = 47.12 sq in

and

(0.5) (π) (42) = 65.97 sq in

Total release for Cr-51 was estimated as follows:

Airborne activity near cut (μCi) =
(47.12 sq in) (0.589) (μCi/sq in) (8 cuts)
+ (65.97 sq in) (0.589 μCi/sq in) (4 cuts) = 0.378 μCi

No significant airborne releases are expected from cutting blowdown main steam or feedwater piping.

Prior to cutting the construction opening liner plate, the internal surfaces of the plate will be decontaminated to reduce the amount of transferable contamination to below 2,200 dpm/100 cm^2.

Gaseous radioactivity associated with repair outage will be similar and not expected to exceed the amount of gaseous effluents associated with other routine refueling/work outages. Gaseous effluents for the 1976 refueling/steam generator inspection outage, which included a complete offload of the core, totaled 1.31 Ci.

PALISADES PLANT SGRR

D-7 Explain what release path, treatment and effluent monitoring will be used for the air which is exhausted from the containment. Also, explain how releases through the construction opening will be prevented or controlled and monitored. Page 4-4 of your report states that the opening will be covered; page 4-25 states that air will be drawn in through the opening. Provide clarification.

RESPONSE:

Airborne radioactivity inside containment during the steam generator repair effort will be controlled, monitored, and ultimately released via the plant vent stack, which utilizes a high-efficiency particulate air (HEPA) filter. In addition to the stack filter (HEPA), the air will be conditioned, if necessary, for removal of airborne radioactivity by use of two installed recirculation filters (HEPA plus charcoal absorber) rated at 6,000 cfm.

A temporary construction covering will be provided to restrict the infiltration of dust, sand, and water into the containment. Air will be drawn through the construction opening and exhausted by the plant stack, preventing radionuclide release from the construction opening.

PALISADES PLANT SGRR

D-8 Provide estimates of the doses which 1) an individual living near the site and 2) the population within 50 miles might receive due to the gaseous effluents from the repair effort.

RESPONSE:

Total body dose and significant organ doses to an individual from gaseous effluents are estimated to be less than 1 mrem.

Total body doses to the population and average doses to individuals in the population from gaseous effluents within a distance of 50 miles from the site are estimated to be less than 1 manrem and 1 mrem, respectively.

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D-9 Estimate the volume and radioactivity content of the various types of solid radioactive waste which will be generated by the repair effort:

a) concrete

b) metal shavings

c) construction materials

d) solidified decontamination fluids

e) piping

f) paper, rags, clothing, etc.

g) spent resins, evaporator bottoms, spent filters, etc.

RESPONSE:

a) Concrete - The only concrete expected to be removed is at the containment construction opening. The concrete is covered by the liner plate on the inside and should have little or no contamination. This concrete (or part of it) will be disposed of by conventional methods if contamination can be reduced to acceptable levels (less than 2,200 dpm/100 cm^2). Otherwise it will be disposed of as solid radwaste. Volume is approximately 4,200 cubic feet.

b) Metal Shavings - The metal shavings will result from cutting piping and miscellaneous steel. The metal shavings volume is estimated at 3 cubic feet. Radioactive content is estimated to be less than 0.5 curie.

c) Construction Materials - Liner plate and other construction material is estimated at 40,000 cubic feet, containing less than 25 curies.

d) Solidified Decontamination Fluids - The solidified decontamination fluids are estimated to be 3,000 cubic feet, with a radioactive content of less than 4.5 curies if the primary piping spool pieces are reused. If new piping spools are used, approximately 2,000 cubic feet and less than 0.5 curie are in this category.

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e) Piping - If primary piping spool pieces are reused, it is estimated that piping will be approximately 25 cubic feet and contain less than 0.05 curie. The use of new primary piping would result in about 200 cubic feet of solid radwaste, containing less than 4.0 curies.

f) Paper, Rags, and Clothing - In this category, approximately 40,000 cubic feet are expected, containing about 50 curies.

g) Spent Resins - Based upon previous refueling outages, spent resins are estimated to be approximately 8,650 cubic feet, with about 80 curies.

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D-10 We have been informed that VEPCO experienced 214 man-rem in cutting reactor coolant piping at Surry. Fifteen cuts were made with a motorized, track mounted plasma arc device. You have estimated that 35 man-rem will be used to make 12 cuts at Palisades with a track mounted oxygen-acetylene torch. The dose rates at Surry were in the 40-60 millirem per hour range, similar to your estimated/measured dose rates. Reevaluate your estimate of 35 man-rem for Palisades in light of experience at Surry.

RESPONSE:

It is believed that experience gained from the Surry outage will help to reduce manhours associated with various outage-related tasks, and the above estimate is appropriate for the task as defined for the Palisades steam generator repair project.

PALISADES PLANT SGRR

D-11 Your discussion of radiation protection training on page 4-29 is not clear. Explain who will receive what levels of training. Explain who is included in "selected crafts." Explain what training will be given to training personnel entering 1) containment building 2) radiation areas, and 3) high radiation areas. Explain what training will be given to personnel who may wear respiratory protection devices.

RESPONSE:

A comprehensive course in radiological protection, in addition to the required orientation course, will be available to a selected group of craft personnel. This selection will be at the discretion of the health physics staff, but will necessarily include craft supervision or personnel responsible for directing craft work groups in radiation areas. As a minimum, all personnel entering the containment building, radiation areas, and high radiation areas will be required to successfully complete the orientation course described in the Palisades plant health physics procedures. Respiratory training requirements are described in the Palisades Health Physics Procedures (Section 7). The general training program will address, as a minimum, the following:

1. Discussion of the need for respirators

2. Classifications of hazards that may require the use of a respirator

3. General classes of respirators used for respiratory protection and a general discussion of their operation principles

4. Discussion of the types of respirators used at Palisades, including proper donning-operating instructions

5. The quantitative fitting program

6. User's responsibility

7. The evaluation program (wholebody counting, air sampling, and other bioassays)

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D-12 Discuss your plans for keeping radiation exposures ALARA when rewelding primary coolant piping. Evaluate the potential man-rem savings of decontaminating the cut-out sections of primary coolant piping. Indicate whether, based on that evaluation, the cut-out pipe sections will be decontaminated or not. At Surry decontamination of the primary coolant sections was successful in significantly reducing radioactive contamination. The lower contamination levels resulted in lower doses for reinstallation of the piping.

RESPONSE:

Decontamination by electropolishing is considered a viable technique for reducing total manrems. A second alternative, replacing old spool pieces, is being considered at this time. The manrem estimate presented in Table C-1-5 (Question Set C) considers this potential manrem savings in Work Area 8C inside the reactor coolant pipe after decontamination. It should be noted that if the old spool pieces are replaced with new sections, radiation fields associated with Work Area 8C would be further reduced.

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D-13 Radioactive contamination on the outside of the steam generators could be released during storage or during transportation. Estimate the amount of radioactive contamination on the outside of the steam generators, and describe any plans for removing or fixing that contamination. Your estimate of removable activity should be based on actual measurements. Also, describe any plans for incorporating these features into the design of the steam generator storage building to control the release of radioactivity.

a) A sealed floor,

b) A floor sump from which water could be sampled for radioactivity (if any collected),

c) Particulate filters over ventilation or breathing openings in walls and ceiling.

RESPONSE:

Smear surveys on outside steam generator surfaces have shown removable contamination ranging from 500 to 50,000 dpm/100 cm^2. The outside of the steam generators will be decontaminated by scrubbing with detergent and water to reduce the amount of transferable contamination. Any transferable surface contamination above 2,200 dpm/100 cm^2 will be fixed by painting.

Steam generators will be sealed to prevent the release of radioactivity during transfer and subsequent onsite storage (see Section 4.3.5.2). The only significant radiological consideration associated with storage is the direct radiation from the steam generator (see Section 4.4.6). Shielding will be provided to ensure an acceptable radiation level outside the storage facility (see Section 4.4.2).

The steam generator storage building will be provided with a sump to collect liquid such as that resulting from roof leakage and condensation. The floor will be sealed if the floor permeability is found to be a concern. Because of efforts taken to seal and clean the steam generators, particulate filters in the storage building are not necessary. However, a surveillance program will be implemented to provide assurance that there is no unanticipated release of radioactivity to the environment (see Section 4.4.6).

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D-14 Describe any plans to maintain a high water level inside the steam generators to provide shielding during the repair effort. This measure was effective in reducing dose rates during operations such as pipe cutting at Surry.

RESPONSE:

Unlike Surry, a circumferential cut will not be required on the upper shell of the steam generators. Consequently, this technique is not expected to reduce exposures significantly. However, a high water level on the secondary side will be maintained to an extent practical.

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D-15 Evaluate the ability of your radwaste solidification system to acceptably solidify the solutions which will be used for local decontamination in the repair effort. The current burial ground criteria for acceptance of solid radwaste call for the waste to contain no free water.

RESPONSE:

Our present solidification system using sodium-silicate has been used and found adequate for solidification of decontamination solutions. The repair outage is not expected to generate any waste incompatible with the present system.

PALISADES PLANT SGRR

D-16 Describe what precautions and post-repair tests will be utilized to ensure that the effectiveness of ventilation and safety-related air filter systems is not impaired.

RESPONSE:

Equipment tests specified in the Palisades Plant Technical Specification, Table 4.2.2, Item 11 will be conducted to ensure that the effectiveness of ventilation and safety-related air filter systems is not impaired during the post-repair period.

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E-1 Items C.3 and E.4 in Table 2.1-1 reflect the increase of water level in the new "SG." Modify the Technical Specifications (Item 5 in Table 2-3.1) "the low steam generator water level trip."

RESPONSE:

In Table 2.1-1 of the SGRR, Item C.3 shows the increase in overall height of the steam generators, which is due solely to the increased height of the new main steam outlet nozzles. Item E.4 shows that the new steam generators will contain a greater mass of water on the secondary side at full power; this is due to changes in the internals and thermal hydraulic (void fraction) changes. This serves to increase the water inventory below the "low steam generator water level trip" as specified in the technical specifications, thereby enhancing safety. The normal water level will be approximately the same. Therefore, this technical specification for the new steam generators need not be changed.

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E-2 In 2.2-1-1, if the total tube outside area A is to be decreased in order to maintain $(UA)_n = (UA)_o$, and since the tube sheet dimensions seem to be unchanged, how is the "A" reduction going to be achieved? (a) less number of tubes with the same original tube length, or (b) same number of tubes with less than the original tube length and consequently less tube height at two-90^o bends?

2-1: If A is to be decreased, and (b) above is the chosen option, evaluate the net effect on natural circulation (eg, during a loss of coolant flow, a station blackout, etc).

2-2: If A is not to be decreased, explain qualitatively the varying effects that will change U.

RESPONSE:

The replacement steam generators will have 8,182 heat transfer tubes of a 0.750-inch outside diameter (od) and a 0.042-inch average wall. The original steam generators have 8,519 heat transfer tubes of a 0.750-inch od and a 0.048-inch average wall. The number of heat transfer tubes for the replacement steam generator was established, using an equivalent UA basis; i.e., (UA) replacement steam generators equal (UA) original steam generator. This equivalent UA basis will provide the replacement steam generators with essentially the same thermal performance capability as the original steam generator. The tube bundle geometry is arranged, using a 1-inch triangular pitch pattern for the tube holes and is identical to the spacing pattern utilized on the original generators. The equivalent UA basis, therefore, results in the replacement steam generators having fewer tubes and a smaller average tube length because the overall heat transfer coefficient U for 0.042-inch average wall tubes is greater than that for the 0.048-inch average wall tubes of the old steam generator.

2-1: This design difference will not significantly affect the natural circulation because most of the natural circulation heat transfer occurs just above the tubesheet.

2-2: This is not applicable. See also the discussion regarding UA above.

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E-3 In 6.1.1, what are the new design features that will improve the flow distribution in the new SG in light of the fact that you decided not to use the flow distribution baffle?

RESPONSE:

Design features which enhance the flow distribution and circulation included in the design of the replacement steam generators are eggcrate type tube supports, tube lane divider plate, and revised blowdown duct design. The eggcrate type tube support system used on the replacement steam generator offers several advantages over the drilled tube support plates used on the original generators from a hydraulic standpoint. Because an eggcrate support is of a relatively open type construction, localized crevices adjacent to tube surfaces are minimized. The eggcrate tube support system in the steam generator provides a minimum number of potential, localized, steam-blanketed areas which might be present in the annular gaps between tubes and drilled support plate holes. The open flow area of each eggcrate support is calculated to be 69% of the total flow area. From a different point of view, the eggcrate support obstructs only 31% of the available secondary flow area. The large open flow areas available in the eggcrate support system avoids the accumulation of boiler water deposits by minimizing local flow eddies and flat surfaces which are present in other commonly used tube bundle support systems. The tube support design on the replacement steam generator has also been designed to reduce the number of supports, both horizontal and vertical grids in the tube bend region. In addition, the vertical tube pitch in the bend region has been increased from 1.0 to 1.75 inches, which significantly reduces flow resistance in the bend region. The resultant effect is that the tube support system selected for the replacement steam generator is a design which reduces the potential for localized corrosion and increases internal recirculation.

On the replacement steam generator, a divider plate is mounted in the tube lane between the hot and cold leg sides of the tube bundle. This divider plate, in conjunction with the revised blowdown duct, prevents the preferential bypass of the tube bundle by recirculating water exiting from the downcomer along the tube lane, thereby forcing the recirculation fluid from the downcomer to be directed across the tube bundle. The overall effect is to ensure relatively high radial fluid velocities throughout the region above the tubesheet.

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E-4.1 In Section 6.1.3 explain the effect of core inlet temperature on blowdown, reflood, and peak clad temperature.

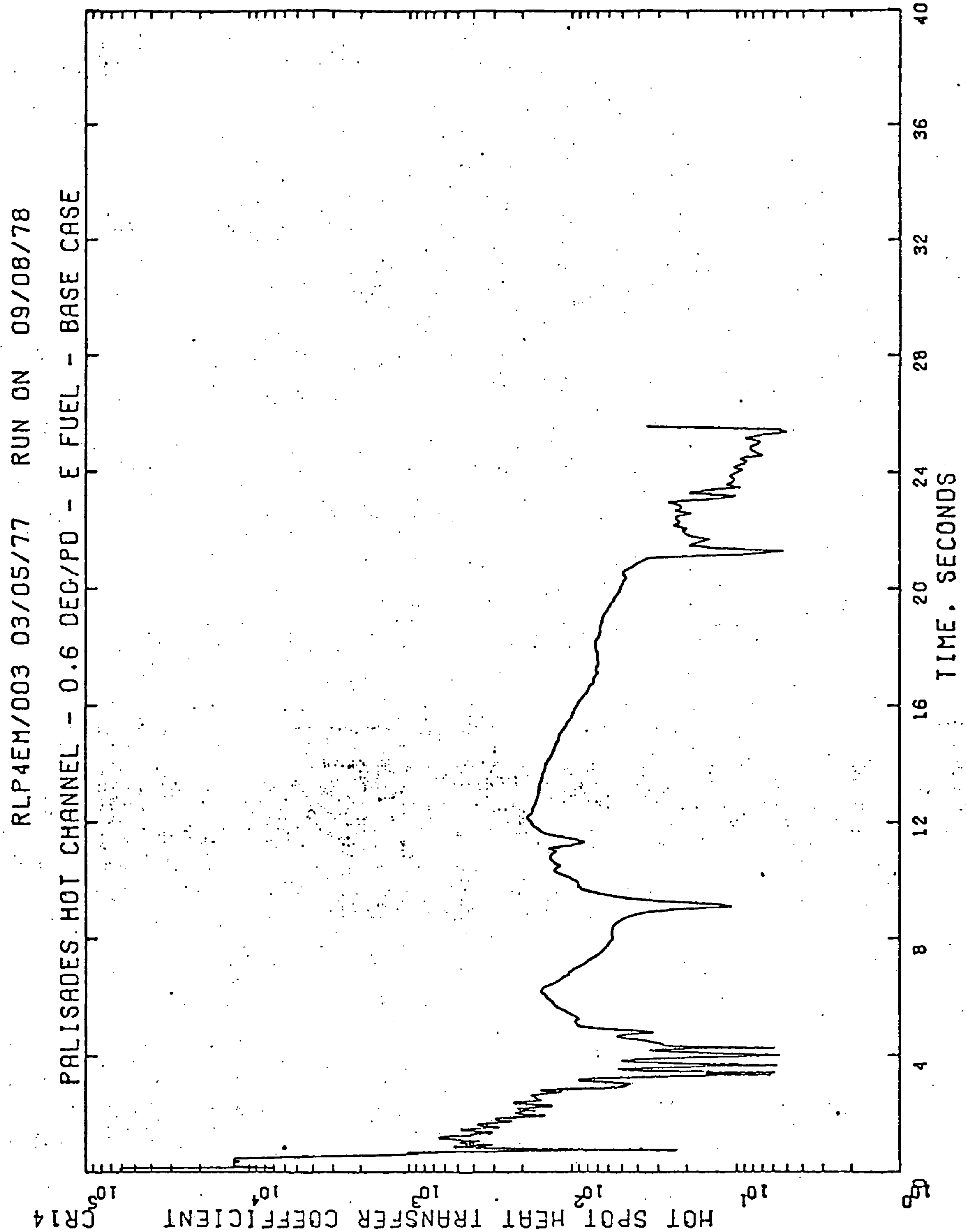
RESPONSE:

The core inlet temperature sensitivity study showed fuel rod heat transfer during blowdown to be relatively insensitive to core inlet temperature variations. The primary effect of increased core inlet temperature was observed in the reflood phase of the LOCA. With a higher core inlet temperature, slightly more energy would be released to the containment during the LOCA. This results in higher containment pressures and therefore increased core flooding rates due to less steam binding in the steam generators.

E-4.2 In Section 6.1.3 explain the effect of primary system pressure on blowdown, reflood, and peak clad temperature.

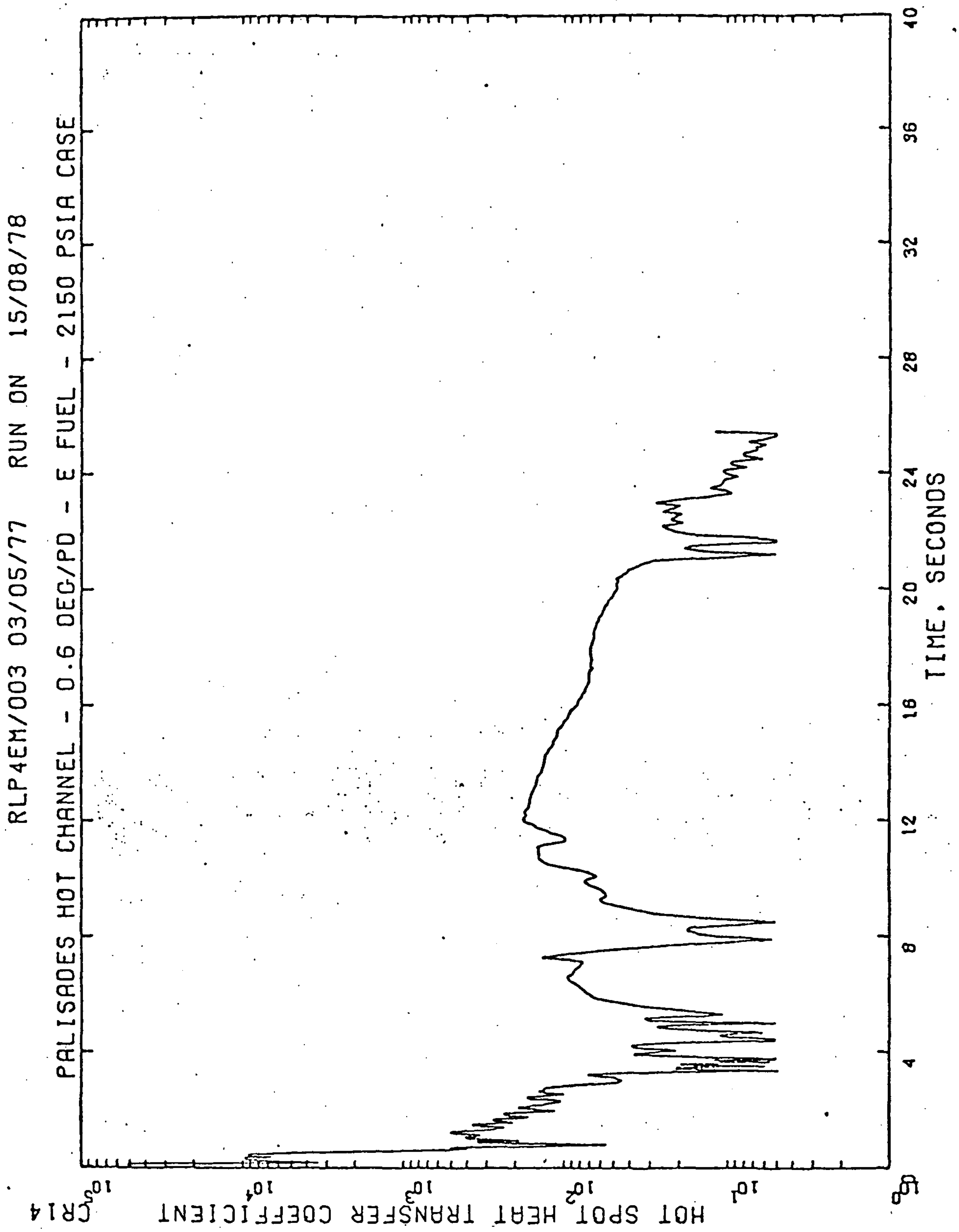
RESPONSE:

The major effect of increased primary system pressure was observed during the blowdown phase of the LOCA. With a higher pressure (2,150 versus 2,060 psia), flow through the core and hot assembly during blowdown was degraded sufficiently to result in poorer fuel rod heat transfer during blowdown and ultimately a higher peak clad temperature. For comparison purposes, the calculated heat transfer coefficient at the hot spot in the core is shown in Figures E-4.2-1 and E-4.2-2 for the base case and for the 2,150 psia case, respectively.



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

HOT SPOT
HEAT TRANSFER COEFFICIENT
FOR 2060 PSIA
Figure E-4.2-1



PALISADES PLANT
STEAM GENERATOR REPAIR REPORT

HOT SPOT
HEAT TRANSFER COEFFICIENT
FOR 2150 PSIA
Figure E-4.2-2

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E-4.3 At the end of page 6.1e, it is stated that "because of higher expected primary system flowrates and higher expected secondary steam pressures with the new steam generators as compared to the reference analysis." Explain this statement in relation to the last sentence in Section 2.2.1.1 and items A.6 and B.4 of Table 2.1-1.

RESPONSE:

Section 2.2.1.1 and Table 2.1-1 refer to steam generator design parameters. However, the referenced analysis in the sentence in question is the most recent LOCA analysis performed for Palisades. This analysis assumed actual operating pressures, temperatures, and flowrates as measured in the plant with the present level of tube plugging. In addition, for conservatism, approximately 500 additional tubes were assumed to be plugged. With the replacement steam generators installed, the primary system flowrate is expected to significantly exceed that assumed in the referenced analysis. Further, with the increased flow, high primary coolant temperatures, and therefore secondary steam pressures, will be permissible based on current core thermal margin limits

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E-5.1 In 6.1-4, justify the conclusion reached in the second paragraph that the LOCA mass/energy release for the replacement SG at full power would produce less containment peak pressure than that produced with the original SG.

RESPONSE:

An explicit LOCA analysis for effects on containment peak pressure was not performed. The new steam generator primary volumes are slightly less than the original steam generator; therefore, there is less reactor coolant system fluid available. Less initial inventory will produce lower peak containment pressure.

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E-5.2 Provide the LOCA mass/energy release curves for both the original and replacement SG. Also provide the containment pressure curve for the original SG.

RESPONSE:

Because the primary side volume of the replacement steam generators will be slightly less than the respective volume of the original units, the total LOCA mass/energy release to containment will actually be less with the replacement steam generators. Thus, the containment pressure response for the largest LOCA (the 42-inch, double-ended guillotine rupture) will be no more adverse than that shown in Figure 14.18-2 of the FSAR. It should be noted that the FSAR containment analysis was based on a core thermal power level of 2,650 MW t.

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E-5.3 Briefly describe the analytical methods used for full power MSLB mass/energy release analyses and compare the similarities and differences compared to the NRC accepted method.

RESPONSE:

The analytical methods used for a full-power main steam line break (MSLB) are the NRC-accepted methods. The digital computer code SGN III, which has been accepted on other dockets, was used. A particularly detailed discussion can be found in Appendix 6.B of Combustion Engineering's CESSAR-PSAR. In addition, credit was taken for the steam outlet nozzle restrictors and main steam isolation on high containment pressure as explained in Section 6.1.4 of the Steam Generator Repair Report.

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E-6 Refer to Page 2-9

In the new steam generators, how will the thermal sleeve design minimize leakage (to reduce the likelihood of steam gen/FW water hammer)?

Can this joint be welded to eliminate leakage entirely?

RESPONSE:

The feedwater liner design for the replacement steam generator incorporates an expanded (explosively expanded) seal on the outboard piping end of the nozzle and an overlapping slip joint at the liner sparger joint. The feedwater sparger assembly, internal to the steam generator, includes a series of top discharge elbows. This top discharge design, as opposed to the underside discharge port design and the overall feedwater sparger design, prevents immediate draining of the feedwater ring in the event of a loss of feedwater flow, thereby minimizing the possibility of a water hammer.

If the liner to the sparger joint were to be welded, this would fix the feedwater thermal liner at both the inboard and outboard ends. The free movement of the liner due to thermal temperature differentials would then be prevented, a condition which could induce a fatigue failure of the liner sparger joint.

PALISADES PLANT SGRR

E-7 Refer to Page 4-3

Will the defueling of the reactor deviate from the usual and ordinary defueling practices? If so, describe any details that represent a deviation.

How much time will elapse between reactor shutdown and off-loading of the core into the spent fuel pool. Does this deviate from the time approved to maintain maximum allowable pool water temperatures?

RESPONSE:

In preparation for conducting the steam generator repair program, the defueling of the reactor is not expected to deviate from usual and ordinary defueling practices. It is estimated that a period of 4 to 5 weeks will elapse between reactor shutdown and the completion of offloading of the core into the spent fuel pool. This does not deviate from the time approved to maintain maximum allowable fuel pool water temperatures.

PALISADES PLANT SGRR

E-8 Refer to Page 4-8

In the transportation of the steam generator, will there be administrative controls in force to limit the maximum speed of the loaded transporter? If so, what will be the maximum speed that will be compatible with the maximum grade of 7% and the minimum turning radius of 42 feet?

RESPONSE:

During the transportation of the steam generator, administrative controls will be exercised to limit the traveling speed. The traveling speed compatible with a 7% grade and a 42'-0" turning radius is approximately 1/4 mile per hour.

PALISADES PLANT SGRR

E-9 If a storage site is necessary for the new steam generators, what assurance will there be that the access route from the storage location will not adversely expose safety-related equipment to possible damage?

RESPONSE:

The storage site for new steam generators, if necessary, will be located where there are no safety-related equipment or structures in the immediate vicinity of the storage or along the path of the access route as presently contemplated. The nearest Seismic Category I structure is the containment at the time of off-loading and loading steam generators through the containment construction opening. However, the reactor will have been defueled at the time of operation. Therefore, a new steam generator storage site and transportation will not adversely expose the safety-related equipment or structures to damage.

The electric transmission tower adjacent to the containment is not a Seismic Category I structure, but supports offsite power lines. In the unlikely event of any damage to the transmission tower, diesel generators will be used for supplying power to the safety-related equipment until the offsite power is restored. The capability to restore offsite power within 1 week will be available onsite.

PALISADES PLANT SGRR

E-10 Refer to Page 4-6, 4-9

Will the rigging be tested and prequalified before removing the old steam generators?

RESPONSE:

All rigging equipment outside containment will be prequalified prior to removal of the steam generators. Rigging equipment inside containment will be tested in place using the existing steam generator as a test load. The weight of the existing steam generator will be augmented, as required, with additional load to provide the required test load.

PALISADES PLANT SGRR

E-11 Is the fuel pool cooling pump supply, breaker, and its control power supply for the cooling pump monitored and alarmed in the control room?

RESPONSE:

The Palisades plant has two fuel pool cooling pumps, P-51A and P-51B. A 480-volt, 3-phase supply to pump P-51A is taken from motor control center (MCC) 7, breaker 52-717, whereas pump P-51B is fed from breaker 52-817 on MCC 8.

Loss of the 480-volt, 3-phase supply at the MCC or at the breaker, or the loss of a 120-volt control power supply for these pumps leads to a visual alarm indication on panel C08 in the control room when both the red and green indicating lights go off.

Further the auto-trip of these fuel pool cooling pumps is alarmed on window 7 of the safeguards, safety injection, and isolation annunciator on main control room panel C13.

PALISADES PLANT SGRR

E-12 Is the radiation monitoring instrumentation in the fuel pool area available during this period? Provide details.

RESPONSE:

Radiation monitoring instrumentation in the fuel pool area consists of two radiation area monitors and two process monitors. These monitors will be available during all steam generator repair outage activities. The monitors are designated as follows:

a. RE 2313 - spent fuel pool area monitor

b. RE 5709 - radwaste demineralizer (elevation 649 feet) area monitor

c. RE 5711 - radwaste addition ventilation monitor

d. RE 5712 - fuel handling radwaste ventilation monitor

The area radiation monitors (2313 and 5709) are coaxial ion chambers, and the process monitors (5711 and 5712) are geiger-mueller tubes.

Readout and display equipment is located in the control room. Detector ranges and sensitivities are chosen to enable monitoring to within the requirements of 10 CFR 20.

PALISADES PLANT SGRR

E-13 Section 4.7.1 references the Consumers' Topical Report for "those quality assurance activities within Consumers' scope of responsibilities on this project." This reference is not clear. However, the Consumers Power Company QA Topical Report adequately describes quality assurance provisions that pertain to major modifications. The second sentence of 4.7.1 should be changed to:

"This report is an integral part of Consumers Power Company Quality Assurance Program Manual for Nuclear Power Plants and the provisions pertaining to major modifications will be invoked for the Palisades Steam Generator Repair Project."

RESPONSE:

Section 4.7.1 has been modified as suggested, and a revised Page 4-46 has been issued with Revision 4 to the Steam Generator Repair Report.

PALISADES PLANT SGRR

E-14 Similarly, the reference to the Bechtel QA Topical Report in Section 4.7.2 is not clear. The second sentence of 4.7.2 should be changed to:

"This topical report will be invoked for activities within Bechtel's scope of responsibilities on this project."

RESPONSE:

Section 4.7.2 has been modified as suggested, and a revised Page 4-46 has been issued with Revision 4 to the Steam Generator Repair Report.