ENCLOSURE 1

EXISTING APPENDIX A TECHNICAL SPECIFICATIONS
AND THOSE PROPOSED IN P.C. # 125

3.1 REACTOR COOLANT SYSTEM

3.1.1 MAXIMUM REACTOR COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum activity in the reactor coolant system at any time.

OBJECTIVE: To limit the consequences of an accidental release of reactor coolant to the environment.

SPECIFICATION:

A. The specific activity of the reactor coolant shall be limited to:

1. $\leq$ 1.0 uCi/gm dose equivalent I-131.

2. $\leq$ 100/$\bar{E}$ uCi/gm, where $\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant.

B. ACTION:

1. With the specific activity of the coolant determined to be >1.0 uCi/gm but <60 uCi/gm dose equivalent I-131, the reactor may be started up or operation may continue for up to 48 hours provided that operation under these circumstances does not exceed 800 hours in any consecutive 12 month period. Should the total operating time at a reactor coolant specific activity >1.0 uCi/gram dose equivalent I-131 exceed 500 hours in any consecutive six month period, the licensee shall report the number of hours of operation above this limit to the NRC within 30 days.

2. With the specific activity of the reactor coolant determined to be >1 uCi/gm dose equivalent I-131 for more than 48 hours during one continuous time interval or >60 uCi/gm dose equivalent I-131 or >100/$\bar{E}$ uCi/gm, have the reactor subcritical with the average temperature of the reactor coolant (T_{avg}) less than 535°F within 6 hours.

3. With the specific activity of the reactor coolant >1.0 uCi/gm dose equivalent I-131 or >100/$\bar{E}$ uCi/gm, perform the sampling and analysis requirements of item 1a.4.a of Table 4.1.2 until the specific activity of the reactor coolant is restored to within its limits. A reportable occurrence shall be prepared and submitted to the Commission within

30 days. This report shall contain the results of the specific activity analysis together with the following information:

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

b. Fuel burnup by core region,

c. Cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded,

d. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

e. The time duration when the specific activity of the reactor coolant exceeded >1.0 uCi/gram dose equivalent I-131.

C. The dose equivalent to I-131 shall be determined for those isotopes with their dose conversion factors as listed in Table III of TID-14844 (see reference). The effective activity of those isotopes shall be summed with the measured activity of I-131 to obtain the I-131 dose equivalance.

D. The provisions of Specification 3.0.4 are not applicable.

BASIS: <u>Specific Activity</u>

The limitations on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed the guidelines of 10 CFR Part 100 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 uCi/gram dose equivalent I-131, but <60 uCi/gm dose equivalent I-131, accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 uCi/gram dose equivalent I-131 but <60 uCi/gm dose equivalent I-131 must be restricted to no more than 800 hours in any consecutive 12 month period since the maximum allowable activity level increases the 2 hour thyroid dose at the site boundary significantly following a postulated steam generator tube rupture.

Reducing T_{avg} to < 535°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the dose equivalent I-131 exceeds 1.0 uCi/gram and following a significant change in power level to monitor possible iodine spiking phenomena to assure the activity remains <60 uCi/gm dose equivalent I-131.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

3.1.4 LEAKAGE

Applicability: Applies to reactor coolant system leakage

Objective: To ensure that leakage from the reactor coolant system does not exceed acceptable limits.

Specification:

A. The reactor coolant system shall be monitored for evidence of leakage.

B. Detectable leakage from the primary coolant system shall be investigated and evaluated. In any event, if the leakage exceeds 1 gpm and the source of leakage is not identified, the reactor shall be shut down. If the sources of leakage have been identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 6 gpm shall be permitted.

C. The reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of any of the following conditions:

1. An increase in primary to secondary leakage of 140 gpd (0.1 gpm) over a period of twenty-four hours in any steam generator.

2. Any primary to secondary leakage in excess of 215 gpd (0.15 gpm) in any steam generator; or

3. Measured increase in primary to secondary leakage in excess of 15 gpd (0.01 gpm) per day, when measured primary to secondary leakage is above 140 gpd.

Following reactor shutdown, leaking tubes will be repaired or plugged.

D. In addition, in accordance with the Technical Specifications, the reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of primary to secondary leaks in excess of 0.3 gpm in any steam generator. Following reactor shutdown, an eddy current inspection will be performed as required by the Technical Specifications, any leaking steam generator tubes will be repaired or plugged and the NRC will be notified prior to resumption of plant operation.

Basis: Two basic kinds of leakage from the reactor coolant system are possible, namely:

1. To other closed systems.

2. Directly to the containment.

Systems into which leakage from the reactor coolant system could occur are designed to accept such leakage. However, leakage directly into the containment indicates the possibility of a breach in the coolant envelope. For this reason, the acceptable value for a source of leakage not identified was set at one gpm.

Once the source of leakage has been identified, it can be determined if operation can safely continue. Under these conditions, an allowable leakage rate of 6 gpm has been established. This is based upon the contingency of sustained loss of all off-site power and failure of the onsite generation. With 6 gpm leakage, decay heat removal can safely be accomplished for a period in excess of 12 hours. Within the 12 hour period, the reactor coolant system can be depressurized.

To comply with Paragraph IV.C.1(b)(4) of the "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors" adopted by the AEC on June 19, 1971, the maximum allowable identified leakage rate from the primary coolant system has been established as not exceeding 6 gpm. This value is based on operating experience regarding non-safety related equipment limitations which has shown that, under certain circumstances where primary system leakage is directed to the gas handling portion of the radwaste system, the capacity of this system would be exceeded during extended operation with a leakage greater than 6 gpm.

Detection of leaks from the reactor coolant system to the containment is accomplished through use of any or all of the following methods:

1. Sump level

2. Radiation monitoring

3. Humidity measurements

With these methods, a leak of one gpm can be detected in a matter of hours. Detection of leaks to other systems is accomplished through the use of radiation monitoring, level indications in the affected system, and water chemistry variations. In both cases, large leaks would be detected by indications from process variables in the reactor coolant and related systems.

The justification for the 0.3 gpm primary to secondary leakage limit is as described in the Basis for Technical Specification 4.16.

4.2.3 Safety Injection System Hydraulic Valve Testing (Surveillance Requirement)

An interim surveillance testing program shall be conducted during the remainder of the current fuel cycle which began in June 1981. At the next refueling outage, the interim program shall be supplanted by a long term surveillance testing program. It is intended that this long term program will be developed and submitted to the NRC for review and approval at least 60 days prior to the next refueling outage.

The interim surveillance program shall be as follows:

1. At least once every 92 days, (except when the inverval lapses while in mode 5 or 6, in which case the test may be delayed until a mode 3 or 4 operation prior to the next entry into mode 2) the unit shall be placed in mode 3 or 4 and a Hot SIS functional test (with the MOV-850 A, B&C valves locked closed) shall be performed. This test shall include a determination of the force required to open valves HV-851 A&B and the margin to available actuation force. This test shall be evaluated on the basis of the following criteria:

   a. If the measured actuator force for both the HV-851 A&B valves is less than 10,000 lb_f*, the unit may be returned to power.

   b. If the measured actuator force of either HV-851 A or B is between 10,000 and 22,000 lb_f, the Hot SIS test for both valves shall be repeated to again determine required opening force and available margin. The prediction will assume a straight line extrapolation from the following equation:

$$T = \frac{(22{,}000 - F_2)}{(F_1 - F)/T_L}$$

where F_1 = measured actuator force from the first Hot SIS test during the current surveillance test (lb_f)

F_2 = measured actuator force from the second Hot SIS test during the current surveillance test (lb_f)

*Upon receipt of satisfactory data from continuing testing and analysis, the NRC staff will consider a request from Southern California Edison Company to change this number to more accurately reflect existing conditions.

T_L = time (in days) since the last surveillance testing

F = the actuator force from the previous surveillance test (lb_f)*

If the calculated value of T does not exceed 92 days, the next surveillance test must be performed before T days had elapsed.

c. If the measured actuator force of either HV-851 A or B is greater than 22,000 lb_f, the valve(s) shall be declared inoperable. Test results shall be reported to the NRC along with proposed corrective actions and NRC approval obtained prior to returning the unit to service.

2. The first test shall be performed not less than 14 days nor more than 21 days following return to power from the current outage which began September 3, 1981.

*For the first surveillance test, the value of F shall be the average actuator force of HV-851 A&B valves from pre-operation testing (3135 lb_f). All subsequent surveillance testing shall assume the F_2 value from the previous surveillance test for each valve. If an F_2 was not required during the previous surveillance test, the F_1 value for each valve shall be assumed.

4.8 REACTIVITY ANOMALIES

Applicability: Applies to potential reactivity anomalies.

Objective: To require evaluation of reactivity anomalies within the reactor.

Specification: A. Following a normalization of the computer boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Atomic Energy Commission.

Basis: To eliminate expected errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a reactivity insertion of this amount would not result in pressure or temperature transients which exceed the design conditions of the reactor coolant system.

4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability:

Applies to the leakage of radioactive source materials.

Objective:

To verify the physical integrity of portable and fixed radioactive calibration sources.

Specification:

A. Byproduct material sealed sources which exceed the quantities listed in 10 CFR §30.71, Schedule B, and all other sealed sources containing greater than 0.1 microcuries shall be leak tested in accordance with Specification B, C and D below.

   Exception: Not withstanding the periodic leak test required by this specification, any licensed sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

B. Each sealed source containing radioactive material, other than Hydrogen 3, with a half life greater than thirty days and in any form other than gas, shall be tested for leakage and/or contamination prior to use out of storage and prior to transfer to another person and thereafter at intervals not to exceed six months. This test does not apply to sealed sources that are stored and not in use.

C. The leakage test shall be capable of detecting the presence of .005 microcuries of radioactive material. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate.

D. If testing reveals the presence of .005 microcuries or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or disposed of in accordance with applicable regulatory requirements.

Basis:

This Specification assures that leakage from radioactive material sources does not exceed allowable total body or organ limits. In the unlikely event that those quantities of radioactive byproduct materials of interest to this Specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10CFR70.39 (c) limits for plutonium.

4.16 INSERVICE INSPECTION OF STEAM GENERATOR TUBING

Applicability:

Applies to the inservice inspection and sampling selection for steam generator tubing.

Objective:

To monitor the integrity of the steam generator tube primary boundary and provide guidance for corrective action when imperfections are observed.

Specification:

A. GENERAL STEAM GENERATOR TUBE SELECTION

The steam generators shall be inspected when shutdown by selecting steam genertor tubes on the following basis:

1. Tubes for the inpsection shall be selected on a random basis except where experience at San Onofre Unit 1 or experience in similar plants indicates critical areas to be inspected.

2. Each inspection shall include at least 3% of the total number of tubes in each steam generator to be inspected.

3. Inservice inspections may be limited to one steam generator on a rotating schedule encompassing 3% of the total tubes of steam generators in the plant if the results of previous inspections indicate that all steam generators are performing in a like manner.

4. Every inspection shall include all non-plugged tubes in the steam generators to be inspected that previously had detected imperfections greater than 20%, except as specified in Specification C.1.

B. SUPPLEMENTARY INSPECTIONS

If the inspections in Specification A indicate imperfections, additional inspections shall be required as follows:

1. If any of the tubes inspected pursuant to Specification A.3 have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration since their last inspection, inspect 3% of the tubes in one of the uninspected steam generators.

2. If more than 10% of the tubes inspected in a steam generator have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have incrased more than 10% wall penetration since their last inspection, or one or more of the tubes inspected have an imperfection in excess of the plugging limit, inspect an additional 3% of the tubes in that steam generator, concentrating on tubes in those areas of the tube sheet array where tubes with imperfections were found and on that length of tube where the imperfections were found. In addition, the rest of the steam generators shall be inspected in accordance with Specification A.2.

3. If the additional inspection in Specification B.2 indicates that more than 10% of the additionally inspected tubes have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration since their last inspection, or one or more of the additionally inspected tubes have an imperfection in excess of the plugging limit, inspect an additional 6% of the tubes in that steam generator in the area of the tubesheet array where tubes with imperfections were found and through that length of tube where the imperfections were found.

C. SPECIAL STEAM GENERATOR TUBE INSPECTIONS

In addition to the general steam generator tube inspections performed in Specifications A and B, every inspection shall include the following special inspections:

1. Every inspection shall include all nonplugged tubes in one of the steam generators that previously had been noted as having discretely quantifiable imperfections greater than 30% at antivibration bar (AVB) intersections, and all

This page included
FOR REFERENCE ONLY

non-plugged tubes in that steam generator that previously had been noted as having imperfections at AVB intersections which were not discretely quantifiable but which were identified during previous inspections as being in the 30 to 50% range.

2. At each steam generator inspection, all previously identified restricted tubes in either steam generator A or C shall be gauged by using eddy current probes to determine restriction sizes.

D. INSPECTION FREQUENCY

The inspections in Specifications A and B above shall be performed at the following frequencies:

1. Inservice inspections shall be not less than 10 nor more than 24 calendar months after the previous inspection.

2. If two consecutive inspections indicate that less than 10% of the tubes inspected have either (a) new imperfections greater than 20% or (b) previous imperfections that have increased more than 10% since their last inspection, the inspections shall be not less than 10 nor more than 40 calendar months after the previous inspection.

3. Unscheduled inspections shall be conducted in accordance with Specification A in the event of primary-to-secondary leaks exceeding Specification 3.1.4.C, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

E. ACCEPTANCE CRITERIA

1. As used in this specification:

   a. Imperfection means an exception to the dimensions, finish, or contour required by drawing or specification.

   b. Defect means an imperfection of such severity that the tube is unacceptable for continued service.

This page included
FOR REFERENCE ONLY

c. Plugging limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50% of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 40% of the nominal sleeve wall thickness.

2. If, in the inspections performed under Specification A,

   a. Less than 10% of the total tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

   b. No tube inspected exceeds the plugging limit,

   plant operation may resume.

3. If, in the inspections performed under Specification B,

   a. Less than 10% of the total tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

   b. No more than 3 of the tubes inspected exceed the plugging limit,

   plant operation may resume after performance of the corrective action in Specification F.

4. If, in the inspections performed under Specification B,

   a. More than 10% of the tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, or

   b. More than 3 of the tubes inspected exceed the plugging limit,

   the situation shall be reported to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

5. If in the inspections performed under Specification C.1, wear rates are observed at AVB intersections which are inconsistent with the 50% plugging criterion, the situation shall be reported to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

6. If in the inspections performed under Specification C.2 progression of the denting process is observed to be recurring, the situation shall be reported to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

F. CORRECTION ACTION

All leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limit shall be repaired or plugged.

Basis: The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = .3 gallons per minute per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads

imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of .3 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require shutdown during which the leaking tubes will be located and plugged and additional inspections performed.

If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 50% of the tube nominal wall thickness, except where sleeves are installed, in which case the plugging limit is 40% of the nominal sleeve wall thickness. A plugging limit of 50% for tubes and 40% for sleeves ensures that defects will not occur between inspection intervals.

The results of tube ID gauging and dent detection conducted in San Onofre Unit 1 steam generators demonstrate that the denting process has been arrested. Continuing assurance of this condition can be provided by performing a program of limited tube ID gauging and dent detection in either steam generator A or C on a refueling outage frequency. Adequate surveillance of denting related tube restrictions can be maintained at refueling intervals by noting any new restrictions during the conduct of general surveillance and AVB inspections and by gauging tubes which have previously been noted as being restricted. Progression of denting can also be monitored in either steam generator A or C by evaluating third and fourth support plate denting data obtained from the general surveillance and AVB inspections as well as from the ID gauging program and comparing these results with those of previous inspections.

The results of AVB area inspections conducted in San Onofre Unit 1 steam generators demonstrate that AVB modifications installed during the Cycle VI refueling outage were successful in eliminating significant growth of tube wall penetration indications at AVB locations. Continuing assurance of this condition can be provided by performing U-bend inspections at refueling outage intervals of tubes having wall penetration indications in excess of 30% at AVB locations.

6.9.2 UNIQUE REPORTING REQUIREMENTS

The special reports shall be submitted as required:

a. Inservice Inspection (Technical Specification 4.7)

b. Reactor Vessel Surveillance Program (Technical Specification 4.9)

c. Fire Protection Systems (Technical Specification 3.14)

d. The results of required leak tests performed on sealed sources (Technical Specification 4.12) shall be reported annually if the tests reveal the presence of 0.005 uCi or more of removable contamination.

e.

f.

Reserved for Future Use. SCE Proposed Change No. 83

g.

h.

ENCLOSURE 2

PROPOSED REVISIONS TO APPENDIX A TECHNICAL SPECIFICATIONS

3.1 REACTOR COOLANT SYSTEM

3.1.1 MAXIMUM REACTOR COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum activity in the reactor coolant system at any time.

OBJECTIVE: To limit the consequences of an accidental release of reactor coolant to the environment.

SPECIFICATION: The specific activity of the reactor coolant shall be limited to:

1. $\leq$ 1.0 uCi/gm DOSE EQUIVALENT I-131.

2. $\leq 100/\bar{E}$ uCi/gm, where $\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant.

ACTION:

A. With the specific activity of the coolant determined to be >1.0 uCi/gm but <60 uCi/gm DOSE EQUIVALENT I-131, STARTUP or POWER OPERATION may continue for up to 48 hours provided that operation under these circumstances does not exceed 800 hours in any consecutive 12 month period. Should the total operating time at a reactor coolant specific activity >1.0 uCi/gram DOSE EQUIVALENT I-131 exceed 500 hours in any consecutive six month period, within 30 days the licensee shall report to the NRC, pursuant to Specification 6.9.2, the number of hours of operation above this limit.

B. With the specific activity of the reactor coolant determined to be >1 uCi/gm DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or >60 uCi/gm DOSE EQUIVALENT I-131 or $>100/\bar{E}$ uCi/gm, be in at least HOT STANDBY with the average temperature of the reactor coolant (T_{avg}) less than 535°F within 6 hours.

C. With the specific activity of the reactor coolant >1.0 uCi/gm DOSE EQUIVALENT I-131 or $>100/\bar{E}$ uCi/gm, perform the sampling and analysis requirements of item 1a.4.a of Table 4.1.2 until the specific activity of the reactor coolant is restored to within its limits. A Licensee Event Report shall be prepared and submitted to the Commission pursuant to Specification 6.6. This report shall contain the results of the specific activity analysis together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the reactor coolant exceeded >1.0 uCi/gram DOSE EQUIVALENT I-131.

D. The provisions of Specification 3.0.4 are not applicable. PC. #125

BASIS: Specific Activity

The limitations on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed the guidelines of 10 CFR Part 100 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 uCi/gram DOSE EQUIVALENT I-131, but <60 uCi/gm DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 uCi/gram DOSE EQUIVALENT I-131 but <60 uCi/gm DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours in any consecutive 12 month period since the maximum allowable activity level increases the 2 hour thyroid dose at the site boundary significantly following a postulated steam generator tube rupture.

Reducing T_{avg} to < 535°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the DOSE EQUIVALENT I-131 exceeds

1.0 uCi/gram and following a significant change in power level to monitor possible iodine spiking phenomena to assure the activity remains <60 uCi/gm DOSE EQUIVALENT I-131.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

3.1.4 LEAKAGE

APPLICABILITY: Applies to reactor coolant system leakage.

OBJECTIVE: To ensure that leakage from the reactor coolant system does not exceed acceptable limits.

SPECIFICATION:

A. The reactor coolant system shall be monitored for evidence of leakage.

B. Detectable leakage from the primary coolant system shall be investigated and evaluated. In any event, if the leakage exceeds 1 gpm and the source of leakage is not identified, the reactor shall be shut down. If the sources of leakage have been identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 6 gpm shall be permitted.

C. The reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of any of the following conditions:

1. An increase in primary to secondary leakage of 140 gpd (0.1 gpm) over a period of twenty-four hours in any steam generator.

2. Any primary to secondary leakage in excess of 215 gpd (0.15 gpm) in any steam generator; or

3. Measured increase in primary to secondary leakage in excess of 15 gpd (0.01 gpm) per day, when measured primary to secondary leakage is above 140 gpd.

Following reactor shutdown, leaking tubes will be repaired or plugged.

D. In addition, in accordance with the Technical Specifications, the reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of primary to secondary leaks in excess of 0.3 gpm in any steam generator. Following reactor shutdown, an eddy current inspection will be performed as required by the Technical Specifications, any leaking steam generator tubes will be repaired or plugged and the NRC notified pursuant to Specification 6.9.2 prior to resumption of plant operation.

T_L = time (in days) since the last surveillance testing

F = the actuator force from the previous surveillance test (lb_f)*

If the calculated value of T does not exceed 92 days, the next surveillance test must be performed before T days had elapsed.

c. If the measured actuator force of either HV-851 A or B is greater than 22,000 lb_f, the valve(s) shall be declared inoperable. Test results shall be reported to the NRC pursuant to Specification 6.6 along with proposed corrective actions and NRC approval obtained prior to returing the unit to service.

2. The first test shall be performed not less than 14 days nor more than 21 days following return to power from the current outage which began September 3, 1981.

---

* For the first surveillance test, the value of F shall be the average actuator force of HV-851 A&B valves from pre-operation testing (3135 lb_f). All subsequent surveillance testing shall assume the F_2 value from the previous surveillance test for each valve. If an F_2 was not required during the previous surveillance test, the F_1 value for each valve shall be assumed.

4.8 REACTIVITY ANOMALIES

APPLICABILITY: Applies to potential reactivity anomalies.

OBJECTIVE: To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION: A. Following a normalization of the computer boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made within 30 days and reported to the NRC pursuant to Specification 6.9.2.

BASIS: To eliminate expected errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safety limit since a reactivity insertion of this amount would not result in pressure or temperature transients which exceed the design conditions of the reactor coolant system.

4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

APPLICABILITY: Applies to the leakage of radioactive source materials.

OBJECTIVE: To verify the physical integrity of portable and fixed radioactive calibration sources.

SPECIFICATION:

A. Byproduct material sealed sources which exceed the quantities listed in 10 CFR 30.71, Schedule B, and all other sealed sources containing greater than 0.1 microcuries shall be leak tested in accordance with Specification B, C and D below.

   Exception: Not withstanding the periodic leak test required by this specification, any licensed sealed source is exemptfrom such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material of 10 microcuries or less of alpha emitting material.

B. Each sealed source containing radioactive material, other than Hydrogen 3, with a half life greater than thirty days and in any form other than gas, shall be tested for leakage and/or contamination prior to use out of storage and prior to transfer to another person and thereafter at intervals not to exceed six months. This test does not apply to sealed sources that are stored and not in use.

C. The leakage test shall be capable of detecting the presence of .005 microcuries of radioactive material. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate.

D. If testing reveals the presence of .005 microcuries or more of removalbe contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or disposed of in accordance with applicable regulatory requirements and reported in the subsequent annual report filed pursuant to Specification 6.9.1.4.

BASIS: This Specification assures that leakage from radioactive material sources does not exceed allowable total body or organ limits. In the unlikely event that those quantities of radioactive byproduct materials of interest to this Specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

c. Plugging limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50% of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 40% of the nominal sleeve wall thickness.

2. If, in the inspections performed under Specification A,

   a. Less than 10% of the total tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

   b. No tube inspected exceeds the plugging limit,

   plant operation may resume.

3. If, in the inspections performed under Specification B,

   a. Less than 10% of the total tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

   b. No more than 3 of the tubes inspected exceed the plugging limit,

   plant operation may resume after performance of the corrective action in Specification F.

4. If, in the inspections performed under Specification B,

   a. More than 10% of the tubes inspected have imperfections greater than 20% that were not detected during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

   b. More than 3 of the tubes inspected exceed the plugging limit,

   the situation shall be reported to the Commission in accordance with Specification 6.6 for approval of the proposed remedial action.

5. If in the inspections performed under Specification C.1, wear rates are observed at AVB intersections which are inconsistent with the 50% plugging criterion, the situation shall be reported to the Commission in accordance with Specification 6.6 for approval of the proposed remedial action.

6. If in the inspections performed under Specification C.2, progression of the denting process is observed to be recurring, the situation shall be reported to the Commission in accordance with Specification 6.6 for approval of the proposed remedial action.

F. CORRECTIVE ACTIONS

All leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limit shall be repaired or plugged.

BASIS: The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that thre is evidence of degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = .3 gallons per minute per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads.

6.9.2 SPECIAL REPORTS

Special reports shall be submitted to the NRC Regional Administrator, within the time period specified for each report.