UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. DPR-31

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated August 18, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 95, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 31, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated August 18, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: August 31, 1983

# ATTACHMENT TO LICENSE AMENDMENTS

## AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NO. DPR-31

## AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-41

## DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| i & vi | i & vi |
| 1-5 | 1-5 |
| 1-6 | 1-6 |
| 1-7 | 1-7 |
| 3.1-5 | 3.1-5 |
| - | 3.1-5a |
| - | 3.1-5b |
| - | Figure 3.1-1 |
| Table 4.1-2 (Sheet 1 of 3) | Table 4.1-2 (Sheet 1 of 3) |
| Table 4.1-2 (Sheet 2 of 3) | Table 4.1-2 (Sheet 2 of 3) |
| B3.1-1 | B3.1-1 |
| B3.1-1a | B3.1-1a |
| B3.1-2 through 3.1-8 | B3.1-2 through 3.1-7 |

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## LIST OF FIGURES

## 1.14 DOSE EQUIVALENT I-131

The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", or in NRC Regulatory Guide 1.109, Rev. 1 October, 1977.

## 1.15 POWER TILT

The power tilt is the ratio of the maximum to average of the upper out-of-core normalized detector currents or the lower out-of-core normalized detector currents whichever is greater. If one out-of-core detector is out of service, the remaining three detectors are to be used to compute the average.

## 1.16 REACTOR COOLANT PUMPS

The reactor shall not be operated with less than three reactor coolant pumps in operation.

## 1.17 LOW POWER PHYSICS TESTS

Low power physics tests are below a nominal 5% of rated power which measure fundamental characteristics of the reactor core and related instrumentation.

## 1.18 ENGINEERED SAFETY FEATURES

Features such as containment, emergency core cooling, and containment atmospheric cleanup systems for mitigating the consequences of postulated accidents.

## 1.19 REACTOR PROTECTION SYSTEM

Systems provided to act, if needed, to avoid exceeding a safety limit in anticipated transients and to activate appropriate engineered safety features as necessary.

## 1.20 SAFETY RELATED SYSTEMS AND COMPONENTS

Those plant features necessary to assure the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents which could result in off-site exposures comparable to the guideline exposures of 10 CFR 100.

## 1.21 PER ANNUM

During each calendar year.

## 1.22 REACTOR COOLANT SYSTEM PRESSURE BOUNDARY INTEGRITY

For purposes of low temperature RCS overpressure protection, the RCS will have pressure boundary integrity UNLESS the RCS is open to containment and the minimum area of the RCS opening is greater than 2.20 square inches.

## 1.23 COOLANT LOOP

Each of the following is defined as being a Coolant Loop:

1. Reactor Coolant Loop A and its associated reactor coolant pump and steam generator with secondary side level greater than or equal to 10%.

2. Reactor Coolant Loop B and its associated reactor coolant pump and steam generator with secondary side level greater than or equal to 10%.

3. Reactor Coolant Loop C and its associated reactor coolant pump and steam generator with secondary side level greater than or equal to 10%.

4. Residual Heat Removal Loop A and its associated residual heat removal pump and heat exchanger.

5. Residual Heat Removal Loop B and its associated residual heat removal pump and heat exchanger.

1.24 $\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be 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, with half lives greater than 30 minutes, making up at least 95% of the total noniodine activity in the coolant.

e. After shutdown, corrective action shall be taken before operation is resumed.

f. Above 2% of rated power, two leak detection systems of different principles shall be operable one of which is sensitive to radioactivity.

g. Reactor Coolant System leakage shall be limited to 1 gpm total primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System.

4. MAXIMUM REACTOR COOLANT ACTIVITY

The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\overline{E}$ microcuries per gram.

With the above limits being exceeded, the following actions shall be taken:

1. When the reactor is critical or average reactor coolant temperature is greater than 500 F:

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3 within 30 days indicating the number of hours above this limit.

b. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1-1, be in a SHUTDOWN condition with average reactor coolant temperature less than 500 F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\overline{E}$ microcuries per gram, be in a SHUTDOWN condition with average reactor coolant temperature less than 500 F within 6 hours.

2. <u>For all modes of operation</u>

a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ microcuries per gram, perform the sampling and analysis requirements of item 1.h.1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.b. This report shall contain the results of the specific activity analyses 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. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operations, 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 primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

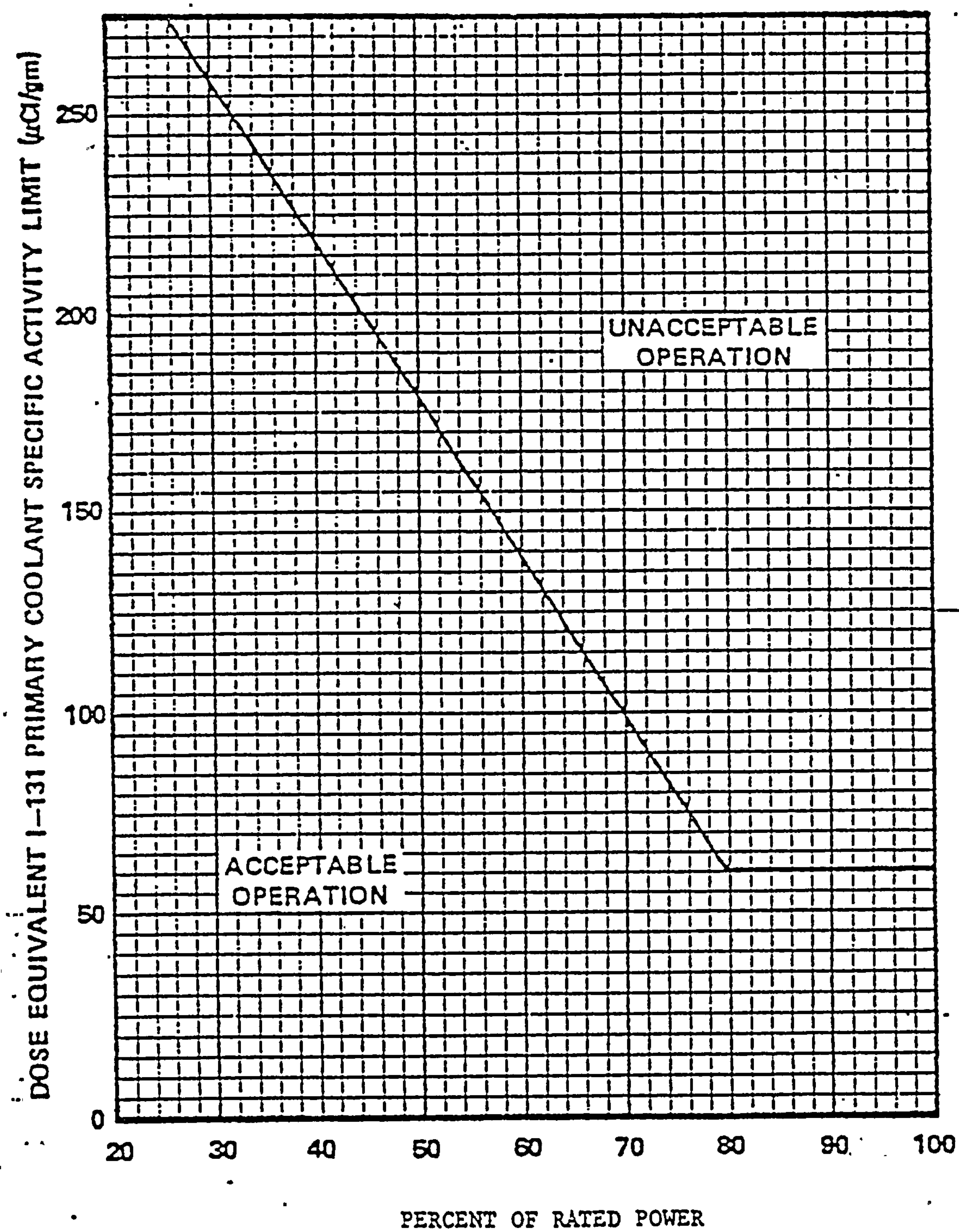


FIGURE 3.1-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131.

TABLE 4.1-2 (Sheet 1 of 3)

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

| | Check | Frequency | Max. Time Between Tests (Days) |
|---|---|---|---|
| 1. Reactor Coolant Samples | a) Radiochem. ($T_{1/2} > 30$ Min) | Monthly | 45 |
| | b) Cl and $O_2$ and F | 5/Week | 3 |
| | c) Tritium Activity | Weekly | 10 |
| | d) Gross β, γ Activity (μCi/cc) | 5/Week | 3 |
| | e) Boron Concentration | 2/Week | 5 |
| | f) $\overline{E}$ Determination | Semi-annually | 30 Weeks |
| | g) Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | Biweekly | 18 |
| | h) Isotopic Analysis for Iodine including I-131, I-133 and I-135 | 1) Once per 4 # hours whenever the specific activity exceeds 1.0 μCi/gm DOSE EQUIVALENT I-131 or 100/$\overline{E}$ μCi/gm | NA |
| | | 2) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the RATED POWER within a 1 hour period. | NA |
| 2. Refueling Water Storage Tank Water Sample | Boron Concentration | Weekly † | 10 |
| 3. Boric Acid Tank | Boron Concentration | 2/Week | 5 |
| 4. Boron Injection Tank* | Boron Concentration | Monthly † | 45 |
| 5. Control Rods | Rod drop times of all full length rods | For all rods at least once per 18 months and following each removal of the reactor vessel head. For specifically affected individual rods following maintenance on or modification of the control rod drive system which could affect the drop time of those specific rods. | |

* See reference (11) on Page B3.4-2

\# Until the specific activity of the primary coolant system is restored within its limits.

TABLE 4.1-2 (Sheet 2 of 3)

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

| | Check | Frequency | Max. Time Between Test (Days) |
|---|---|---|---|
| 5. Control Rods (cont'd) | Partial movement of full length rods | Biweekly while critical | 20 |
| 6. Pressurizer Safety Valves | Set Point | Each refueling shutdown | NA |
| 7. Main Steam Safety Valves | Set Point | Each refueling shutdown | NA |
| 8. Containment Isolation Trip | Functioning | Each refueling shutdown | NA |
| 9. Refueling System Interlocks | Functioning | Prior to each refueling | NA |
| 10. Accumulator | Boron Concentration | At least once per 31 days and within 6 hours after each solution volume increase of $\geq 1\%$ of tank volume. ÷ | |
| 11. Reactor Coolant System Leakage | Evaluate | Daily | NA |
| 12. Diesel Fuel Supply | Fuel inventory | Weekly | 10 |
| 13. Spent Fuel Pit | Boron Concentration | Prior to refueling | NA |
| 14. Secondary Coolant | I-131 Concentration | Weekly* ÷ | 10 |
| 15. Vent Gas and Particulates | I-131 and Particulate Activity | Weekly* | 10 |
| 16. Fire Protection Pump and Power Supply | Operable | Monthly | 45 |
| 17. Turbine Stop and Control Valves, Reheater Stop and Intercept Valves | Closure | Monthly*** | 45 |
| 18. LP Turbine Rotor Inspector (w/o rotor disassembly) | V, MT, PT | Every 5 years | 6 years |
| 19. Spent Fuel Cask Crane Interlocks | Functioning | Within 7 days | 7 days when crane is being used to maneuver spe fuel cask. |

## B3.1 BASES FOR LIMITING CONDITIONS FOR OPERATION, REACTOR COOLANT SYSTEM

### 1. Operational Components

The specification requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling in the event that a loss of flow occurs. The flow provided will keep DNBR well above 1.30. When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the reactor coolant system volume in approximately one half hour.

Each of the pressurizer safety valves is designed to relieve 283,300 lbs. per hour of saturated steam at the valve set point. Below 350 F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lifting pressure would be less than the capacity of a single valve. Also, two safety valves have capacity greater than the maximum surge rate resulting from comples loss of load.[(2)]

The 50 F limit on maximum differential between steam generator secondary water temperature and reactor coolant temperature assures that the pressure transient caused by starting a reactor coolant pump when cold leg temperature is $\leq$ 275 F can be relieved by operation of one Power Operated Relief Valve (PORV). The 50 F limit includes instrument error.

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. In power operation with one reactor coolant loop not in operation, this specification requires that the plant be in at least Hot Shutdown within 1 hour.

In Hot Shutdown, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat, however, single failure considerations require that two loops be operable.

In Cold Shutdown, a single reactor coolant loop or RHR coolant loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The requirement that at least one residual heat removal (RHR) loop be in operation during Refueling Shutdown ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 160 F as required during Refueling Shutdown and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution stratification.

The requirement to have two RHR loops operable when there is less than 23 feet of water above the core ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

2. Pressure/Temperature Limits

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in

Section 4.1.5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for prevention of brittle fracture.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile streses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curves are composite curves prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100 F per hour. The cooldown limit curves are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$. Adjusted reference temperatures, based upon the fluence and copper content of the material in question, are then determined. The heatup and cooldown limit curves include the shift in $RT_{NDT}$ at the end of the service period shown on the heatup and cooldown curves.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples has a definite relationship to the spectra at the vessel inside radius, the measured transition shift for a sample can be related with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.2-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3. Leakage

Any leakage from the reactor coolant system, or from any other system containing potentially radioactive material, is considered to be of major importance as it may indicate a condition is developing that would lead to gross leakage. Gross leakage must be prevented to minimize any remote possibility of release of activity to and from the site. Leakage prevention first of all protects the public and also it prevents potential contamination of the equipment. Prompt maintenance and repair leads to improved reliability, which is an overall operating objective.

Thus any indication of leakage; for example: unbalanced water inventories, radiation monitor reading increases, boric acid crystal deposits, insulation dampness; shall be considered to be the result of a leak and shall require immediate attention with prompt evaluation required.

Action shall be prompt as it is possible that a small leak may propagate and become a major leak. The fact that a leak of 5 gpm, at the maximum allowed reactor coolant activity, released as airborne material without holdup or cleanup, would not exceed 10 CFR 20 limits shall not permit relaxation of the requirement that action be prompt and positive.

When a real or imagined leak is detected, the Plant Supervisor will immediately initiate a detailed investigation as to source and cause after first notifying the Plant Superintendent or his designated alternate. Evaluation will be made by the Plant Superintendent, who will call upon Production Department supervisors, such as the Regional Superintendent and the Superintendent of Generating Stations, as necessary for consultation. This procedure is an established and proven one in operation of fossil fuel fired units when leaks develop, as it brings to bear the judgement of experienced persons.

When the leak has been identified, the plant management will determine by a safety evaluation whether operation may continue. Leakage source (ex. valve stem, pump shaft seal) shall be considered. Make up capability and potential increased demand shall also be one of the evaluation factors.

Leakage in the containment will be detected by one or more of the following:

1. The air particulate monitor
2. The gas monitor
3. The sump level recorder
4. Changes in make up water requirements
5. Visual inspection
6. Audible detection

Leakage to other systems will be detected by: activity changes (ex. within the component cooling system), water inventory changes (ex. tank levels).

4. Maximum Reactor Coolant Activity

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits 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 values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Turkey Point Units 3 & 4 site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.1-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.1-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.1-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Reducing average reactor coolant temperature to less than 500 F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary 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 primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

5. Maximum Reactor Coolant Oxygen and Chloride Concentration

By maintaining the reactor coolant chemistry within the limits specified, the integrity of the Reactor Coolant System is protected. (3)

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank, and further because of the time dependent nature of any adverse effects arising from concentrations in excess of the limits, it is unnecessary to shutdown immediately since the condition can be corrected. Thus the period of 24 hours for corrective action to restore the concentrations within the limts has been established. If the corrective action has not been effective at the end of the 24 hour period, then the reactor will be brought to the cold shutdown condition and the corrective action will continue.

6. DNB Parameters

Reactor Coolant Flow Measurements (4)

Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap readout has been well established by the following equation:

$\frac{\Delta P}{\Delta P_o} = \left(\frac{W}{W_o}\right)^2$, where $\Delta P_o$ is the referenced pressure differential with the corresponding references flow rate $W_o$, and $\Delta P$ is the pressure differential with

the corresponding flow rate W. The full flow reference point is established during initial startup. The low flow trip point is then established by extrapolating along the correlation curve.

References


(1) FSAR Table 4.1-3

(2) FSAR Section 14.1.10

(3) FSAR Section 4.2.8

(4) FSAR Section 4.2.9