



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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 34  
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:
  - A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated May 18, 1989 and supplemented on August 21, 1989 (NLR-N89160) (LCR 89-12, Rev. 1) 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 set forth in 10 CFR Chapter I;
  - 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 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 34, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within sixty days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

*Walter R. Butler*

Walter R. Butler, Director  
Project Directorate 1-2  
Division of Reactor Projects I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: October 20, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 34

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages provided to maintain document completeness.\*

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## 1.0 DEFINITIONS

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The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

### ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

### AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

### AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

### CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

### CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

### CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.
- b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

## DEFINITIONS

### CORE ALTERATION

- 1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

### CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

- 1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be highest value of the FLPD which exists in the core.

### CORE OPERATING LIMITS REPORT

- 1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

### CRITICAL POWER RATIO

- 1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

### DOSE EQUIVALENT I-131

- 1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per 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."

### E-AVERAGE DISINTEGRATION ENERGY

- 1.12  $\bar{E}$  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, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

### EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

- 1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation set-point at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.



## DEFINITIONS

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### END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

- a. Turbine stop valves, and
- b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

### FRACTION OF LIMITING POWER DENSITY

1.15 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

### FRACTION OF RATED THERMAL POWER

1.16 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

### FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

### IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

- a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

### ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

### LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

## DEFINITIONS

### LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

### LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

### MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.23 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be highest value of the FLPD which exists in the core.

### MEMBER(S) OF THE PUBLIC

1.24 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

### MINIMUM CRITICAL POWER RATIO

1.25 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

### OFF-GAS RADWASTE TREATMENT SYSTEM

1.26 An OFF-GAS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting reactor coolant system offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

### OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the radiological environmental monitoring program.

## DEFINITIONS

### OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

### OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

### PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

### PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

### PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

- a. All primary containment penetrations required to be closed during accident conditions are either:
  1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or
  2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.
- b. All primary containment equipment hatches are closed and sealed.
- c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.
- e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.
- f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.



## DEFINITIONS

### PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With SOLIDIFICATION, the PCP shall identify the process parameters influencing SOLIDIFICATION such as pH, oil content, H<sub>2</sub>O content, solids content ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. With dewatering, the PCP shall include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of the low-level radioactive waste disposal site.

### PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is required to purify the confinement.

### RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWT.

### REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

### REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

### ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

## DEFINITIONS

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### SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

- a. All secondary containment penetrations required to be closed during accident conditions are either:
  1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or
  2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.
- b. All secondary containment hatches and blowout panels are closed and sealed.
- c. The filtration, recirculation and ventilation system is in compliance with the requirements of Specification 3.6.5.3.
- d. For double door arrangements, at least one door in each access to the secondary containment is closed.
- e. For single door arrangements, the door in each access to the secondary containment is closed, except for normal entry and exit.
- f. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.
- g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

### SHUTDOWN MARGIN

1.40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

### SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled, by the licensee.

## DEFINITIONS

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### SOLIDIFICATION

1.42 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the water type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP).

### SOURCE CHECK

1.43 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

### SPIRAL RELOAD

1.44 A SPIRAL RELOAD is a core loading methodology employed to refuel the core after a complete core unload. During a SPIRAL RELOAD the fuel is to be loaded into individual control cells (four bundles surrounding a control blade) in a spiral fashion centered on an SRM moving outward. Before initiating a SPIRAL RELOAD, up to four bundles may be loaded in the four bundle locations immediately surrounding each of the four SRMs to obtain the required channel count rate.

1.45 A SPIRAL UNLOAD is a core unloading methodology employed to defuel when the complete core is to be unloaded. The core unload is performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for the four bundles around each of the four SRMs. When sixteen or less fuel bundles are in the core, four around each of the four SRMs, there is no need to maintain the required channel count rate.

### STAGGERED TEST BASIS

1.46 A STAGGERED TEST BASIS shall consist of:

- a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.
- b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

### THERMAL POWER

1.47 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.



## DEFINITIONS

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### TURBINE BYPASS SYSTEM RESPONSE TIME

- 1.48 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two separate time intervals: a) time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established, and b) the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. Either response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

### UNIDENTIFIED LEAKAGE

- 1.49 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

### UNRESTRICTED AREA

- 1.50 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

### VENTILATION EXHAUST TREATMENT SYSTEM

- 1.51 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

### VENTING

- 1.52 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1  
SURVEILLANCE FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
Z	During startup, prior to exceeding 30% of RATED THERMAL POWER, if not performed within the previous 7 days
N.A.	Not applicable.

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

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3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT. The limits specified in the CORE OPERATING LIMITS REPORT shall be reduced to a value of 0.86 times the two recirculation loop operation limit when in single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

#### ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

#### SURVEILLANCE REQUIREMENTS

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4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
- d. The provisions of Specification 4.0.4 are not applicable.



## POWER DISTRIBUTION LIMITS

### 3/4.2.2 APRM SETPOINTS

#### LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint ( $S_{RB}$ ) shall be established according to the following relationships:

<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
$S \leq (0.66(w-\Delta w)^{**} + 51\%)T$	$S \leq (0.66(w-\Delta w)^{**} + 54\%)T$
$S_{RB} \leq (0.66(w-\Delta w)^{**} + 42\%)T$	$S_{RB} \leq (0.66(w-\Delta w)^{**} + 45\%)T$

where: S and  $S_{RB}$  are in percent of RATED THERMAL POWER,  
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,  
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (F RTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

#### ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or  $S_{RB}$ , as above determined, initiate corrective action within 15 minutes and adjust S and/or  $S_{RB}$  to be consistent with the Trip Setpoint values\* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

#### SURVEILLANCE REQUIREMENTS

4.2.2 The F RTP and the CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

- At least once per 24 hours,
- Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to F RTP.
- The provisions of Specification 4.0.4 are not applicable.

\*With CMFLPD greater than the F RTP, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

\*\*The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w).  $\Delta w$  is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow.  $\Delta w = 0$  for two recirculation loop operation.  $\Delta w =$  "To be determined at a later date" for single recirculation loop operation.

## POWER DISTRIBUTION LIMITS

### 3/4.2.3 MINIMUM CRITICAL POWER RATIO

#### LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

$\tau_A = 0.86$  seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.688 + 1.65 \left[ \frac{N_1}{n} \right]^{1/2} (0.052),$$
$$\sum_{i=1}^n N_i$$

$$\tau_{ave} = \frac{\sum_{i=1}^n N_i \tau_i}{\sum_{i=1}^n N_i}$$

$n$  = number of surveillance tests performed to date in cycle,

$N_i$  = number of active control rods measured in the  $i^{th}$  surveillance test,

$\tau_i$  = average scram time to notch 39 of all rods measured in the  $i^{th}$  surveillance test, and

$N_1$  = total number of active rods measured in Specification 4.1.3.2.a.

#### APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

#### ACTION:

- a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be greater than or equal to the EOC-RPT inoperable limit specified in the CORE OPERATING LIMITS REPORT.

## POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION (Continued)

#### ACTION: (Continued)

- b. With MCPR less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

#### 4.2.3 MCPR, with:

- a.  $\tau = 1.0$  prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or
- b.  $\tau$  as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT:

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
- d. The provisions of Specification 4.0.4 are not applicable.



## POWER DISTRIBUTION LIMITS

### 3/4.2.4 LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

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3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

#### ACTION:

With the LHGR of any fuel rod exceeding the limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

#### SURVEILLANCE REQUIREMENTS

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4.2.4 LHGR's shall be determined to be equal to or less than the limit specified in the CORE OPERATING LIMITS REPORT:

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

### 3/4.2 POWER DISTRIBUTION LIMITS

#### BASES

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The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

#### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is specified in the CORE OPERATING LIMITS REPORT.

The calculational procedure used to establish the APLHGR specified in the CORE OPERATING LIMITS REPORT is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1.

## POWER DISTRIBUTION LIMITS

### BASES

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#### AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

For plant operation with single recirculation loop, the MAPLHGR limits specified in the CORE OPERATING LIMITS REPORT are multiplied by 0.86. The constant factor 0.86 is derived from LOCA analysis initiated from single loop operation to account for earlier transition at the limiting fuel node compared to the standard LOCA evaluations.

#### 3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and the flow biased neutron flux-upscale control rod block trip setpoints must be adjusted to ensure that the MCPR does not become less than the fuel cladding Safety Limit or that > 1% plastic strain does not occur in the degraded situation. The scram setpoints and rod block setpoints are adjusted in accordance with the formula in Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.



## POWER DISTRIBUTION LIMITS

### BASES

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#### 3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154<sup>(3)</sup> and the program used in non-pressurization events is described in NEDO-10802<sup>(2)</sup>. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149<sup>(4)</sup>. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the  $K_f$  factor specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the  $K_f$  factor. The  $K_f$  factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The  $K_f$  factors may be applied to both manual and automatic flow control modes.

The  $K_f$  factors values specified in the CORE OPERATING LIMITS REPORT were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The  $K_f$  factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the  $K_f$  factors were calculated such that for the maximum flow rate, as limited by the pump-scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the  $K_f$ .

## POWER DISTRIBUTION LIMITS

### BASES

#### MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated thermal flow.

The  $K_f$  factors specified in the CORE OPERATING LIMITS REPORT are conservative for the General Electric plant operation because the operating limit MCPRs of Specification 3.2.3 is the same as the original 1.20 operating limit MCPR used for the generic derivation of  $K_f$ .

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

#### 3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

#### References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.
2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.
3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.
4. TASC 01-A Computer Program for the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

## ADMINISTRATIVE CONTROLS

### ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

### SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 Routine radioactive release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The historical annual average meteorology or the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM). The Semiannual Radioactive Effluent Release Report shall identify those radiological environmental sample parameters and locations where it is not possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In addition, the cause of the unavailability of samples for the pathway and the new location(s) for obtaining replacement samples should be identified. The report should also include a revised figure(s) and table(s) for the ODCM reflecting the new location(s).



## ADMINISTRATIVE CONTROLS

### SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive effluent release report to be submitted within 60 days after January 1, of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each class of solid waste (as defined by 10 CFR 61) shipped offsite during the report period:

- a. Container volume,
- b. Total curie quantity (specify whether determined by measurement or estimate),
- c. Principal radionuclide (specify whether determined by measurement or estimate),
- d. Type of waste (e.g., spent resin, compact dry waste, evaporator bottoms),
- e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
- f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to the UNRESTRICTED AREA of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP), OFFSITE DOSE CALCULATION MANUAL (ODCM) or radioactive waste systems made during the reporting period.

### MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

### CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the PSE&G generated CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following Technical Specifications:

- 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE
- 3/4.2.3 MINIMUM CRITICAL POWER RATIO
- 3/4.2.4 LINEAR HEAT GENERATION RATE

## ADMINISTRATIVE CONTROLS

### CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A (the latest approved revision), General Electric Standard Application for Reactor Fuel (GESTAR II).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

### SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of NRC via the Licensee Event Report System within 30 days.

### 6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

### SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

- a. Records and logs of unit operation covering time interval at each power level.
- b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.
- c. All REPORTABLE EVENTS submitted to the Commission.
- d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.
- e. Records of changes made to the procedures required by Specification 6.8.1.
- f. Records of radioactive shipments.
- g. Records of sealed source and fission detector leak tests and results.

## ADMINISTRATIVE CONTROLS

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### RECORD RETENTION (Continued)

- h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

- a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.
- b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.
- c. Records of radiation exposure for all individuals entering radiation control areas.
- d. Records of gaseous and liquid radioactive material released to the environs.
- e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.
- f. Records of reactor tests and experiments.
- g. Records of training and qualification for current members of the unit staff.
- h. Records of inservice inspections performed pursuant to these Technical Specifications.
- i. Records of quality assurance activities required by the Quality Assurance Program.
- j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- k. Records of meetings of the SORC and the NSR activities.
- l. Records of the snubber service life monitoring pursuant to Technical Specification 4.7.5.
- m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

### 6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

### 6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity



## ADMINISTRATIVE CONTROLS

### HIGH RADIATION AREA (Continued)

of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr\* shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)\*\*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.
- c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem\* shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Senior Nuclear Shift Supervisor on duty and/or the radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem\* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

### 6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

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\*Measurement made at 18 inches from source of radioactivity.

\*\*Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

ADMINISTRATIVE CONTROLS

PROCESS CONTROL PROGRAM (PCP) (Continued)

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:
  - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;
  - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;
  - b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
  - c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.
2. Shall become effective upon review and acceptance by the SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:
  - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);
  - b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determination; and
  - c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.
2. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the FSAR for the period in which the evaluation was reviewed by SORC. The discussion of each changes shall contain: