Docket Nos. 50-325
and 50-324

DISTRIBUTION
See attached list

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2, REGARDING CHANGES ON ELIMINATION OF CYCLE DEPENDENT PARAMETER VALUES AND DELETION OF THE LINEAR HEAT GENERATION RATE LIMIT (TAC NOS. 66153 AND 66154)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 131 to Facility Operating License No. DPR-71 and Amendment No. 161 to Facility Operating License No. DPR-62, for Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your submittals dated September 4, 1987, as amended and supplemented by letters dated April 5, 1988, February 20, 1989 and March 20, 1989.

The amendments change the Technical Specifications to: (1) modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report for the values of those limits and (2) delete the redundant linear heat generation rate limit from the specifications.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original Signed By:

E. G. Tourigny, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 131 to License No. DPR-71
2. Amendment No. 161 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

DFol
1/1

[BSEP12 AMEND 66153/54]

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:	:
NAME	: PAnderson	:ETourigny:jw	:EAdensam	:	:	:	:
DATE	:4/11/89	:4/11/89	:4/12/89	:	:	:	:

Mr. L. W. Eury
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

Mr. Russell B. Starkey, Jr.
Project Manager
Brunswick Nuclear Project
P. O. Box 10429
Southport, North Carolina 28461

Mr. R. E. Jones, General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Mr. Mark S. Calvert
Associate General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Ms. Grace Beasley
Board of Commissioners
P. O. Box 249
Bolivia, North Carolina 28422

Mrs. Chrys Baggett
State Clearinghouse
Budget and Management
116 West Jones Street
Raleigh, North Carolina 27603

Resident Inspector
U. S. Nuclear Regulatory Commission
Star Route 1
P. O. Box 208
Southport, North Carolina 28461

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georgia 30323

Mr. Dayne H. Brown, Chief
Radiation Protection Branch
Division of Facility Services
N. C. Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

Mr. J. L. Harness
Plant General Manager
Brunswick Steam Electric Plant
P. O. Box 10429
Southport, North Carolina 28461

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
P. O. Box 29520
Raleigh, North Carolina 27626-0520

AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
E. Tourigny
N. Le
L.Spessard (MNBB 3701)
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meeks (4) (P1-137)
W. Jones (P-130A)
E. Butcher (11F23)
D. Fieno
J. Stefano
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. DPR-71

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated September 4, 1987, as amended and supplemented by letter dated April 5, 1988, February 20, 1989 and March 20, 1989, 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 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 25, 1989

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR		
NAME	PAnderson	ETourigny	MYoung	EAdensam		
DATE	4/11/89	4/1/89	4/26/89	5/25/89		

OFFICIAL RECORD COPY

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
I	I
II	II
IV	IV
X	X
XV	XV
XVI	XVI
1-1	1-1
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
1-8	1-8
1-9	1-9
1-10	1-10
1-11	1-11
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-4	B 2-4
B 2-5	B 2-5
B 2-6	B 2-6
B 2-7	B 2-7
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 3-47	3/4 3-47
B 3/4 1-2	B 3/4 1-2
B 3/4 1-17	B 3/4 1-17
B 3/4 2-1	B 3/4 2-1
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ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS

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

ACTION

ACTIONS are those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

AVERAGE PLANAR EXPOSURE

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 at that height.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.

CHANNEL CALIBRATION

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

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

A CHANNEL FUNCTIONAL TEST shall be:

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

CHANNEL FUNCTIONAL TEST (Continued)

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

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls in the reactor core with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative location.

CORE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO

The CORE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO (CMAPRAT) shall be the largest value of the MAPRAT that exists in the core.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.3.1, 6.9.3.2, 6.9.3.3, and 6.9.3.4. Plant operation within these core operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μCi/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 following is defined equivalent to 1 μCi of I-131 as determined from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites": I-132, 28 μCi; I-133, 3.7 μCi; I-134, 59 μCi; I-135, 12 μCi.

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

DEFINITIONS

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint 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.

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

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 be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

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.

LIMITING CONTROL ROD PATTERN

A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a limiting value for APLHGR or MCPR.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor output to activated device, to ensure that components are OPERABLE.

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO

The MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO (MAPRAT) for a bundle shall be the largest value in the bundle of the ratio of the APLHGR at a specific height in the bundle divided by the exposure dependent APLHGR limit for that specific height.

MEMBER(S) OF THE PUBLIC

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

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

ODYN OPTION A

ODYN OPTION A shall be analyses which refer to minimum critical power ratio limits which are determined using a transient analysis plus an analysis uncertainty penalty.

ODYN OPTION B

ODYN OPTION B shall be analyses which refer to minimum critical power ratio limits determined using a transient analysis which includes a requirement for 20% scram insertion times to reduce the analysis uncertainty penalty.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) is a manual which contains the current methodology and parameters to be used to calculate offsite doses resulting from the release of radioactive gaseous and liquid effluents; the methodology to calculate gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints; and, the requirements of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s).

OPERABLE - OPERABILITY (Continued)

Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, 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

An OPERATIONAL CONDITION shall be any one inclusive combination of mode switch position and average reactor coolant temperature as indicated in Table 1.2.

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve 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 3.6.3.1, or

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

PURGE or PURGING is 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 required to purify the containment.

RATED THERMAL POWER

RATED THERMAL POWER shall be total reactor core heat transfer rate to the reactor coolant of 2436 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

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.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE EVENT

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

ROD DENSITY

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

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All automatic reactor building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position.

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the reactor building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon free.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee, as defined by Figure 5.1.3-1.

SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first. Up to four fuel bundles may be loaded around each of the four SRMs.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload 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 fuel bundles around each of the four SRMs. Up to four fuel bundles may be left around each SRM to maintain adequate count rate.

STAGGERED TEST BASIS

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

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is 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

VENTING is 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. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1
FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
SM	At least once per 16 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 release.
NA	Not applicable.

TABLE 1.2
OPERATIONAL CONDITIONS

OPERATIONAL CONDITIONS	MODE SWITCH POSITIONS	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown#,***	> 212°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 212°F
5. REFUELING*	Shutdown or Refuel**,#	≤ 212°F

The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

* Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** See Special Test Exceptions 3.10.1 and 3.10.3.

*** The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is OPERABLE.

2.1 SAFETY LIMITS

BASES

2.0 The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MINIMUM CRITICAL POWER RATIO (MCPR) is no less than the Safety Limit CPR of Specification 2.1.2. The Safety Limit MCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER (Low Pressure or Low FLow)

The use of the NRC approved CPR correlation is not valid for all critical power calculations at pressures below 800 psia or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 800 psia is conservative.

2.1.2 THERMAL POWER (High Pressure and High Flow)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power, result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an approved critical power correlation. Details of the fuel cladding integrity safety limit calculation are provided in Reference 1.

Uncertainties used in the determination of the fuel cladding integrity safety limit and the bases of these uncertainties are presented in Reference 1.

The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Brunswick Unit 1 during any fuel cycle would not be as severe as the distribution used in the analysis. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

Reference

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. However, the pressure safety limit is set high enough such that no foreseeable circumstances can cause the system pressure to rise to this limit. The pressure safety limit is also selected to be the lowest transient overpressure allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III and USAS Piping Code, Section B 31.1.

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5-decade 10-range instrument. The trip setpoint of 120 divisions is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. Range 10 allows the IRM instruments to remain on scale at higher power levels to provide for additional overlap and also permits calibration at these higher powers.

The most significant source of reactivity change during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed, Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRMs are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shut down and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above the Safety Limit MCPR of Specification 2.1.2. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. This margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained by the RSCS and RWM. Of all the possible sources of reactivity input, uniform control rod

2. Average Power Range Monitor (Continued)

withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% APRM trip remains active until the mode switch is placed in the Run position.

The APRM flow-biased trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and, therefore, the monitors respond directly and quickly to changes due to transient operation; i.e., the thermal power of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer. Analyses demonstrate that with only the 120% trip setting, none of the abnormal operational transients analyzed violates the fuel safety limit and there is substantial margin from fuel damage. Therefore, the use of the flow-referenced trip setpoint, with the 120% fixed setpoint as backup, provides adequate margins of safety.

The APRM trip setpoint was selected to provide adequate margin for Safety Limits and yet allows operating margin that reduces the possibility of unnecessary shutdowns. The flow-referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when the combination of thermal power and CMAPRAT indicates a highly peaked power distribution.

3. Reactor Vessel Steam Dome Pressure-High

High Pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating, will also tend to increase the power of the reactor by compressing voids, thus adding reactivity. The trip will quickly reduce the neutron flux counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the

3. Reactor Vessel Steam Dome Pressure-High (Continued)

pressure measurement compared to the highest pressure that occurs in the system during a transient. This setpoint is effective at low power/flow conditions when the turbine stop valve closure is bypassed. For a turbine trip under these conditions, the transient analysis indicates a considerable margin to the thermal hydraulic limit.

4. Reactor Vessel Water Level-Low, Level #1

The reactor water level trip point was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate water to account for evaporation losses and displacement of cooling following the most severe transients. This setting was also used to develop the thermal-hydraulic limits of power versus flow.

5. Main Steam Line Isolation Valve-Closure

The low-pressure isolation of the main steamline trip was provided to give protection against rapid depressurization and resulting cooldown of the reactor vessel. Advantage was taken of the shutdown feature in the run mode which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low pressures does not occur. Thus, the combination of the low-pressure isolation and isolation valve closure reactor trip with the mode switch in the Run position assures the availability of neutron flux protection over the entire range of the Safety Limits. In addition, the isolation valve closure trip with the mode switch in the Run position anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure.

6. Main Steam Line Radiation - High

The Main Steam Line Radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a scram is initiated to reduce the continued failure of fuel cladding. At the same time, the Main Steam Line Isolation Valves are closed to limit the release of fission products. The trip setting is high enough above background radiation level to prevent spurious scrams, yet low enough to promptly detect gross failures in the fuel cladding.

7. Drywell Pressure, High

High pressure in the drywell could indicate a break in the nuclear process systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge tank receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this tank fill up to a point where there is insufficient volume to accept the displaced water, control rod movement would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods when they are tripped.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed.

10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low

The reactor protection initiates a scram signal after the control valve hydraulic oil pressure decreases due to a load rejection exceeding the capacity of the bypass valves or due to hydraulic oil system rupture. The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where upon a loss of oil pressure, control valves closure time is approximately twice as long as that for the stop valves, which means that resulting transients, while similar, are less severe than for stop valve closure. No fuel damage occurs, and reactor system pressure does not exceed the safety relief valve setpoint. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first-stage pressure.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that either:

   1. The inoperable RBM channel is restored to OPERABLE status within 24 hours, or

   2. The redundant RBM is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable RBM is restored to OPERABLE status, and the inoperable RBM is restored to OPERABLE status within 7 days, or

   3. THERMAL POWER is limited such that MCPR will remain above the Safety Limit MCPR of Specification 2.1.2, assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   Otherwise, trip at least one rod block monitor channel.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and during the OPERATIONAL CONDITIONS specified in Table 4.3.4-1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 During power operation, the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall not exceed limits based on applicable APLHGR limit values that have been approved for the respective fuel and lattice type and determined by the approved methodology described in GESTAR-II. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) of each type of fuel shown in the applicable figures of 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 an APLHGR exceeding the limits specified in Technical Specification 3.2.1, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in Specification 3.2.1:

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.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The flow-biased APRM scram trip setpoint (S) and rod block trip set point (S_{RB}) shall be established according to the following relationship:

$$S \leq (0.66W + 54\%)\ T$$

$$S_{RB} \leq (0.66W + 42\%)\ T$$

where: S and S_{RB} are in percent of RATED THERMAL POWER.
W = Loop recirculation flow in percent of rated flow.
T = FRACTION OF RATED THERMAL POWER (FRTP) divided by CORE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO (CMAPRAT) (T is applied only if less than 1.0.)

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

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and CMAPRAT shall be determined, the value of T calculated, and the flow-biased APRM scram trip and rod block trip setpoints verified to be within the above limits or adjusted, as required:

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 CMAPRAT greater than or equal to FRTP.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow and cycle average exposure, shall be equal to or greater than the MCPR limit times the K_f specified in the CORE OPERATING LIMITS REPORT. The MCPR limits for ODYN OPTION A and ODYN OPTION B analyses, used in the above determination, shall be specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With MCPR, as a function of core flow and cycle average exposure, 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 applicable limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR, as a function of core flow and cycle average exposure, shall be determined to be equal to or greater than the applicable MCPR limit of Specification 3.2.3.1:

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 in a LIMITING CONTROL ROD PATTERN for MCPR.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (ODYN OPTION B)

LIMITING CONDITION FOR OPERATION

3.2.3.2 For the OPTION B MCPR limits provided in the CORE OPERATING LIMITS REPORT to be used, the cycle average 20% (Notch 36) scram time (τ_{ave}) shall be less than or equal to the OPTION B scram time limit (τ_B), where τ_{ave} and τ_B are determined as follows:

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

i = Surveillance test number,
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test, and
τ_i = Average scram time to notch 36 for surveillance test i

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i}\right)^{1/2} (\sigma), \text{ where:}$$

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.813 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36),
σ = 0.018 seconds
(standard deviation of the above statistical distribution).

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

Within twelve hours after determining that τ_{ave} is greater than τ_B, the operating limit MCPRs shall be either:

a. Adjusted for each fuel type such that the operating limit MCPR is the maximum of the non-pressurization transient MCPR operating limit specified in the CORE OPERATING LIMITS REPORT or the adjusted pressurization transient MCPR operating limits, where the adjustment is made by:

$$MCPR_{adjusted} = MCPR_{option\ B} + \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B} (MCPR_{option\ A} - MCPR_{option\ B})$$

where: τ_A = 1.05 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,

$MCPR_{option\ A}$ = Specified in the CORE OPERATING LIMITS REPORT,
$MCPR_{option\ B}$ = Specified in the CORE OPERATING LIMITS REPORT, or

b. The OPTION A MCPR limits specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3.2 The values of τ_{ave} and τ_B shall be determined and compared each time a scram test is performed. The requirement for the frequency of scram time testing shall be identical to Specification 4.1.3.2.

3/4.3.4 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The control rod withdrawal block instrumentation shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

APPLICABILITY: As shown in Table 3.3.4-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4-2, declare the channel inoperable until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, POWER OPERATION may continue provided that either:

   1. The inoperable channel(s) is restored to OPERABLE status within 24 hours, or

   2. The redundant trip system is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable channel is restored to OPERABLE status, and the inoperable channel is restored to OPERABLE status within 7 days, or

   3. For the Rod Block Monitor only, THERMAL POWER is limited such that MCPR will remain above the Safety Limit MCPR of Specification 2.1.2, assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   4. Otherwise, place at least one trip system in the tripped condition within the next hour.

c. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one trip system in the tripped condition within one hour.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.4 Each of the above required control rod withdrawal block instrumentation channels shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK, CHANNEL CALIBRATION, and a CHANNEL FUNCTIONAL TEST during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.4-1.

CONTROL RODS (Continued)

potential effects of the rod ejection accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem; therefore, with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the non-fully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shut down for investigation and resolution of the problem.

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent the MPCR from becoming less than the Safety Limit MCPR of Specification 2.1.2 during the limiting power transient analyzed in Section 14.3 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the Safety Limit MCPR of Specification 2.1.2. The occurrence of scram times longer than those specified should be viewed as an indication of a systemic problem with the rod drives and, therefore, the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion

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 the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The limiting values for APLHGR when conformance to the operating limit is performed by hand calculation are provided in the CORE OPERATING LIMITS REPORT for each fuel type and, when required for the most limiting lattice for multiple lattice fuel bundle types.

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 1) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 1. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis APLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the permissible planar power (APLHGR) limits comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

The Technical Specification APLHGR limit is the most limiting composite of the fuel mechanical design analysis APLHGR and the ECCS APLHGR limit.

3/4.2.2 APRM SETPOINTS

The scram setting and rod block functions of the APRM instruments are adjusted to ensure that fuel design and safety limits are not exceeded. The scram settings and rod block settings are adjusted in accordance with the relationship provided in Specification 3.2.2. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.2.3 are derived from an established fuel cladding integrity Safety Limit MCPR approved by the NRC 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 as given in Specification 2.2.1.

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).

Details on how evaluations are performed, on the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in Reference 1 and the CORE OPERATING LIMITS REPORT.

At core THERMAL POWER levels less than or equal to 25% RATED THERMAL POWER, the reactor will be operating at a 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 indicated that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, an MCPR evaluation will be made at 25% 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 above 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 power or power shape, regardless of magnitude that could place operation at a thermal limit.

References:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved version.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Accident Monitoring Instrumentation, Specification 3.3.5.3.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Reactor coolant specific activity analysis, Specification 3.4.5.

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

h. Fire barrier penetration, Specification 3.7.8.

i. Liquid Effluents Dose, Specification 3.11.1.2.

j. Liquid Radwaste Treatment, Specification 3.11.1.3.

k. Dose - Noble Gases, Specification 3.11.2.2.

l. Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

m. Gaseous Radwaste Treatment, Specification 3.11.2.4.

n. Ventilation Exhaust Treatment, Specification 3.11.2.5.

o. Total Dose, Specification 3.11.4.

p. Monitoring Program, Specification 3.12.1.b.

q. Primary Containment Structural Integrity, Specification 4.6.1.4.2

CORE OPERATING LIMITS REPORT

6.9.3.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specification 3.2.1.

CORE OPERATING LIMITS REPORT (Continued)

b. The K_f core flow adjustment factor for Specification 3.2.3.1

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.3.1 and 3.2.3.2.

and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.3.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

b. The May 18, 1984 and October 22, 1984 NRC Safety Evaluation Reports for the Brunswick Reload Methodologies described in:

   1. Topical Report NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," February 1983.

   2. Topical Report NF-1583.02, "Methods of RECORD," February 1983.

   3. Topical Report NF-1583.03, "Methods of PRESTO-B," February 1983.

   4. Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," May 1983.

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

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

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

RECORD RETENTION (Continued)

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.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

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

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility 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 facility radiation and contamination surveys.

d. Records or radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.1-1.

g. Records of reactor tests and experiments.

h. Records of training and qualification for current members of the plant staff.

i. Records of inservice inspections performed pursuant to these Technical Specifications.

j. Records of Quality Assurance activities required by the QA Manual.

RECORDS RETENTION (Continued)

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

l. Records of the service lives of all hydraulic and mechanical snubbers referenced in Section 3.7.5 including the data at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

n. Records of (1) meetings of the PNSC, (2) meetings of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC), (3) the independent reviews performed by the Corporate Nuclear Safety Section, and (4) the independent reviews performed by the Corporate Quality Assurance Audit Program, Performance Evaluation Unit.

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 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less 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.

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 above shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Operations Shift Foreman on duty and/or the Radiation Control Supervisor.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1. Sufficiently detailed information to totally support rationale 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);

   2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and,

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

PROCESS CONTROL PROGRAM (PCP) (Continued)

a. 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:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2. A determination that the change did not reduce the overall conformance of the solidification waste product to existing criteria for solid wastes; and

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS 7/

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

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

   2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

   4. An evaluation of the change that shows the predicted release of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   5. An evaluation of the change that shows the expected maximum exposure to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

7/ Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable to the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-62

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

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated September 4, 1987, as amended and supplemented by letters dated April 5, 1988, February 20, 1989 and March 20, 1989, 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 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 25, 1989

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	ETourigny	MYoung	EAdensam			
DATE	4/11/89	4/11/89	4/26/89	5/23/89			

OFFICIAL RECORD COPY

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
I	I
II	II
IV	IV
X	X
XV	XV
XVI	XVI
1-1	1-1
1-2	1-2
1-3	1-3
1-4	1-4
1-8	1-8
1-9	1-9
B 2-1	B 2-1
B 2-2	B 2-2
B 2-4	B 2-4
B 2-5	B 2-5
3/4 1-17	3/4 1-17
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
4/5 3-47	3/4 3-47
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
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1.0 DEFINITIONS

DEFINITIONS

SECTION

1.0 DEFINITIONS (Continued)

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ADMINISTRATIVE CONTROLS

SECTION PAGE

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ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS

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

ACTION

ACTIONS are those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

AVERAGE PLANAR EXPOSURE

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

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.

CHANNEL CALIBRATION

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

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 indication and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

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

CHANNEL FUNCTIONAL TEST (Continued)

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

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls in the reactor core with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative location.

CORE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO

The CORE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO (CMAPRAT) shall be the largest value of the MAPRAT that exists in the core.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.3.1, 6.9.3.2, 6.9.3.3, and 6.9.3.4. Plant operation within these core operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μCi/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 following is defined equivalent to 1 μCi of I-131 as determined from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites": I-132, 28 μCi; I-133, 3.7 μCi; I-134, 59 μCi; I-135, 12 μCi.

$\bar{E}$ -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 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

The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint 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.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to recirculation pump breaker trip from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

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 be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

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.

LIMITING CONTROL ROD PATTERN

A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a limiting value for APLHGR or MCPR.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor output to activated device, to ensure that components are OPERABLE.

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO

The MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO (MAPRAT) for a bundle shall be the largest value in the bundle of the ratio of the APLHGR at a specific height in the bundle divided by the exposure dependent APLHGR limit for that specific height.

MEMBER(S) OF THE PUBLIC

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 not include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

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

ODYN OPTION A

ODYN OPTION A shall be analyses which refer to minimum critical power ratio limits which are determined using a transient analysis plus an analysis uncertainty penalty.

ODYN OPTION B

ODYN OPTION B shall be analyses which refer to minimum critical power ratio limits determined using a transient analysis which includes a requirement for 20% scram insertion times to reduce the analysis uncertainty penalty.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first. Up to four fuel bundles may be loaded around each of the four SRMs.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload 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 fuel bundles around each of the four SRMs. Up to four fuel bundles may be left around each SRM to maintain adequate count rate.

STAGGERED TEST BASIS

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

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is 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

VENTING is 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. Vent, used in system names, does not imply a VENTING process.

2.1 SAFETY LIMITS

BASES

2.0 The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MINIMUM CRITICAL POWER RATIO (MCPR) is no less than the Safety Limit CPR of Specification 2.1.2. The Safety Limit MCPR represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER (Low Pressure or Low Flow)

The use of the NRC approved CPR correlation is not valid for all critical power calculations at pressures below 800 psia or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 800 psia is conservative.

SAFETY LIMITS

BASES (Continued)

2.1.2 THERMAL POWER (High Pressure and High Flow)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power, result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an approved critical power correlation. Details of the fuel cladding integrity safety limit calculation are given in Reference 1.

Uncertainties used in the determination of the fuel cladding integrity safety limit and the bases of these uncertainties are presented in Reference 1.

The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Brunswick Unit 2 during any fuel cycle could not be as severe as the distribution used in the analysis. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

Reference

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Setpoints specified in Table 2.2.1-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5-decade, 10-range instrument. The trip setpoint of 120 divisions is active in each of the 10 ranges. Thus, as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. Range 10 allows the IRM instruments to remain on scale at higher power levels to provide for additional overlap and also permits calibration at these higher powers.

The most significant source of reactivity change during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed in Section 7.5 of the FSAR. The most severe case involves an initial condition in which the reactor is just subcritical and the IRMs are not yet on scale. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the rod being withdrawn is bypassed. The results of this analysis show that the reactor is shut down and peak power is limited to 1% of RATED THERMAL POWER, thus maintaining MCPR above the Safety Limit MCPR of Specification 2.1.2. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides an adequate thermal margin between the setpoint and the Safety Limits. This margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor; cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained by the RSCS and RWM. Of all

2. Average Power Range Monitor (Continued)

the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally, the heat flux is in near-equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THEMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% APRM trip remains active until the mode switch is placed in the Run position.

The APRM flow-biased trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and, therefore, the monitors respond directly and quickly to changes due to transient operation; i.e., the thermal power of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer. Analyses demonstrate that with only the 120% trip setting, none of the abnormal operational transients analyzed violates the fuel safety limit and there is substantial margin from fuel damage. Therefore, the use of the flow-referenced trip setpoint, with the 120% fixed setpoint as backup, provides adequate margins of safety.

The APRM trip setpoint was selected to provide an adequate margin for the Safety Limits and yet allows an operating margin that reduces the possiblility of unnecessary shutdowns. The flow-referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when the combination of thermal power and CMAPRAT indicates a highly peaked power distribution.

3. Reactor Vessel Steam Dome Pressure-High

High Pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids, thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that either:

   1. The inoperable RBM channel is restored to OPERABLE status within 24 hours, or

   2. The redundant RBM is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable RBM is restored to OPERABLE status within 7 days, or

   3. THERMAL POWER is limited such that MCPR will remain above the Safety Limit MCPR of Specification 2.1.2, assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   Otherwise, trip at least one rod block monitor channel.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and during the OPERATIONAL CONDITIONS specified in Table 4.3.4-1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 During power operation, the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall not exceed limits based on applicable APLHGR limit values that have been approved for the respective fuel and lattice type and determined by the approved methodology described in GESTAR-II. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) of each type of fuel shown in the applicable figures 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 an APLHGR exceeding the limits specified in Technical Specification 3.2.1, initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in Specification 3.2.1:

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.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The flow-biased APRM scram trip setpoint (S) and rod block trip set point (S_{RB}) shall be established according to the following relationship:

$$S \leq (0.66W + 54\%)\ T$$

$$S_{RB} \leq (0.66W + 42\%)\ T$$

where: S and S_{RB} are in percent of RATED THERMAL POWER.
W = Loop recirculation flow in percent of rated flow,
T = FRACTION OF RATED THERMAL POWER (FRTP) divided by CORE MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE RATIO (CMAPRAT) (T is applied only if less than 1.0.)

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

ACTION:

With S or S_{RB} exceeding the allowable value, initiate corrective action within 15 minutes and continue corrective action so that S and S_{RB} are within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and CMAPRAT shall be determined, the value of T calculated, and the flow-biased APRM scram trip and rod block trip setpoints verified to be within the above limits or adjusted, as required:

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 CMAPRAT greater than or equal to FRTP.

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow and cycle average exposure, shall be equal to or greater than the MCPR limit times the K_f specified in the CORE OPERATING LIMITS REPORT. The MCPR limits for ODYN OPTION A and ODYN OPTION B analyses, used in the above determination, shall be specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With MCPR, as a function of core flow and cycle average exposure, 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 applicable limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR, as a function of core flow and cycle average exposure, shall be determined to be equal to or greater than the applicable limit determined of Specification 3.2.3.1:

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 in a LIMITING CONTROL ROD PATTERN for MCPR.

3/4.2.3 MINIMUM CRITICAL POWER-RATIO (ODYN OPTION B)

LIMITING CONDITION FOR OPERATION

3.2.3.2 For the OPTION B MCPR limits provided in the CORE OPERATING LIMITS REPORT to be used, the cycle average 20% (notch 36) scram time (τ_{ave}) shall be less than or equal to the Option B scram time limit (τ_B), where τ_{ave} and τ_B are determined as follows:

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

i = Surveillance test number,
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test, and
τ_i = Average scram time to notch 36 for surveillance test i

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i}\right)^{1/2} (\sigma), \text{ where:}$$

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.813 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36),
σ = 0.018 seconds
(standard deviation of the above statistical distribution)

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

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION:

Within twelve hours after determining that τ_{ave} is greater than τ_B, the operating limit MCPRs shall be either:

a. Adjusted for each fuel type such that the operating limit MCPR is the maximum of the non-pressurization transient MCPR operating limit specified in the CORE OPERATING LIMITS REPORT or the adjusted pressurization transient MCPR operating limits, where the adjustment is made by:

$$MCPR_{adjusted} = MCPR_{option\ B} + \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B} (MCPR_{option\ A} - MCPR_{option\ B})$$

where: τ_A = 1.05 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,

$MCPR_{option\ A}$ = Specified in the CORE OPERATING LIMITS REPORT,
$MCPR_{option\ B}$ = Specified in the CORE OPERATING LIMITS REPORT, or,

b. The OPTION A MCPR limits specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3.2 The values of τ_{ave} and τ_B shall be determined and compared each time a scram time test is performed. The requirement for the frequency of scram time testing shall be identical to Specification 4.1.3.2.

3/4.3.4 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The control rod withdrawal block instrumentation shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

APPLICABILITY: As shown in Table 3.3.4-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4-2, declare the channel inoperable until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, POWER OPERATION may continue provided that either:

   1. The inoperable channel(s) is restored to OPERABLE status within 24 hours, or

   2. The redundant trip system is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable channel is restored to OPERABLE status, and the inoperable channel is restored to OPERABLE status within 7 days, or

   3. For the Rod Block Monitor only, THERMAL POWER is limited such that the MCPR will remain above the Safety Limit MCPR of Specification 2.1.2, assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   4. Otherwise, place at least one trip system in the tripped condition within the next hour.

c. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one trip system in the tripped condition within one hour.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.4 Each of the above required control rod withdrawal block instrumentation channels shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK, CHANNEL CALIBRATION, and a CHANNEL FUNCTIONAL TEST during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.4-1.

CONTROL RODS (Continued)

potential effects of the rod ejection accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem; therefore, with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the non-fully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shut down for investigation and resolution of the problem.

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent the MPCR from becoming less than the Safety Limit MCPR of Specification 2.1.2 during the limiting power transient analyzed in Section 14.3 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the Safety Limit MCPR of Specification 2.1.2 The occurrence of scram times longer than those specified should be viewed as an indication of a systemic problem with the rod drives and, therefore, the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion

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 the Final Acceptance Criteria (FAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The limiting values for APLHGR when conformance to the operating limit is performed by hand calculation are provided in the CORE OPERATING LIMITS REPORT for each fuel type and, when required, for the most limiting lattice for multiple lattice fuel bundle types.

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 1) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 1. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis APLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the permissible planar power (APLHGR) limits comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

The Technical Specification APLHGR limit is the most limiting composite of the fuel mechanical design analysis APLHGR and the ECCS APLHGR limit.

3/4.2.2 APRM SETPOINTS

The scram setting and rod block functions of the APRM instruments are adjusted to ensure that fuel design and safety limits are not exceeded. The scram settings and rod block settings are adjusted in accordance with the relationship provided in Specification 3.2.2. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change.

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 an established fuel cladding integrity Safety Limit MCPR approved by the NRC 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 an instrument trip setting as given in Specification 2.2.1.

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).

Details on how evaluations are performed, on the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in Reference 1 and the CORE OPERATING LIMITS REPORT.

At core thermal power levels less than or equal to 25% 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 indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% 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 above 25% 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 power or power shape, regardless of magnitude that could place operation at a thermal limit.

POWER DISTRIBUTION LIMITS

BASES

References:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel", latest approved version.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Accident Monitoring Instrumentation, Specification 3.3.5.3.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Reactor coolant specific activity analysis, Specification 3.4.5.

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

h. Fire barrier penetration, Specification 3.7.8.

i. Liquid Effluents Dose, Specification 3.11.1.2.

j. Liquid Radwaste Treatment, Specification 3.11.1.3.

k. Dose - Noble Gases, Specification 3.11.2.2.

l. Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

m. Gaseous Radwaste Treatment, Specification 3.11.2.4.

n. Ventilation Exhaust Treatment, Specification 3.11.2.5.

o. Total Dose, Specification 3.11.4.

p. Monitoring Program, Specification 3.12.1.b.

q. Primary Containment Structural Integrity, Specification 4.6.1.4.2

CORE OPERATING LIMITS REPORT

6.9.3.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specification 3.2.1.

CORE OPERATING LIMITS REPORT (Continued)

b. The K_f core flow adjustment factor for Specification 3.2.3.1

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.3.1 and 3.2.3.2.

and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.3.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

b. The May 18, 1984 and October 22, 1984 NRC Safety Evaluation Reports for the Brunswick Reload Methodologies described in:

1. Topical Report NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," February 1983.

2. Topical Report NF-1583.02, "Methods of RECORD," February 1983.

3. Topical Report NF-1583.03, "Methods of PRESTO-B," February 1983.

4. Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," May 1983.

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

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

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

RECORD RETENTION (Continued)

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.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

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

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility 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 facility radiation and contamination surveys.

d. Records or radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7.1-1.

g. Records of reactor tests and experiments.

h. Records of training and qualification for current members of the plant staff.

i. Records of inservice inspections performed pursuant to these Technical Specifications.

j. Records of Quality Assurance activities required by the QA Manual.

RECORDS RETENTION (Continued)

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

l. Records of the service lives of all hydraulic and mechanical snubbers referenced in Section 3.7.5 including the data at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

n. Records of (1) meetings of the PNSC, (2) meetings of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC), (3) the independent reviews performed by the Corporate Nuclear Safety Section, and (4) the independent reviews performed by the Corporate Quality Assurance Audit Program, Performance Evaluation Unit.

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 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less 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.

* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1 above shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Operations Shift Foreman on duty and/or the Radiation Control Supervisor.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1. Sufficiently detailed information to totally support rationale 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);

   2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and,

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

PROCESS CONTROL PROGRAM (PCP) (Continued)

a. 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:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2. A determination that the change did not reduce the overall conformance of the solidification waste product to existing criteria for solid wastes; and

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS[7/]

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

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

   2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

   4. An evaluation of the change that shows the predicted release of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   5. An evaluation of the change that shows the expected maximum exposure to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

7/ Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable to the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 131 TO LICENSE NO. DPR-71

AND AMENDMENT NO. 161 TO LICENSE NO. DPR-62

CAROLINA POWER AND LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated September 4, 1987 (Ref.1), as amended and supplemented by letter dated April 5, 1988 (Ref. 2), the Carolina Power & Light Company (the licensee) proposed changes to the Technical Specifications (TS) for Brunswick Units 1 and 2. These two letters (Refs. 1 and 2) were superseded by a letter dated February 20, 1989 (Ref. 3) and amended by letter dated March 20, 1989 (Ref. 6). The proposed changes include two distinct areas which have been reviewed and approved by the NRC in the context of TS improvement. The first set of proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the "Definitions" section and to the reporting requirements of the "Administrative Controls" section of TS. Guidance on the modification of TS that have cycle specific parameters was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 4).

The other set of proposed changes would modify specifications by deleting redundant limits from the TS. These redundant limits include the limits on the linear heat generation rate (LHGR) and associated total peaking factor (TPF). Guidance on the proposed changes was developed by NRC on the basis of its review of Amendment 19 to NEDE-24011-P-A (Ref. 5). All of the Definitions, Specifications, and Bases of the TS that are affected by the proposed change are provided in Reference 5.

The staff review of the licensee's initial submittals (Refs. 1 and 2) led to clarifying telephone calls. The licensee provided Reference 3 which supersedes References 1 and 2. Our evaluation of the licensee's proposed changes to the TS is based on this latest submittal (Ref. 3), as amended by Reference 6.

2.0 EVALUATION

2.1 Changes Related to Cycle-Specific Parameter Values (Generic Letter 88-16)

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

a. Specification 3.2.1 - Average Planar Linear Heat Generation Rate

The average planar linear heat generation limits are provided in the COLR.

b. Specification 3.2.3.1 - Minimum Critical Power Ratio

The K_f factors and MCPRs are provided in the COLR.

c. Specification 3.2.3.2 - Minimum Critical Power Ratio

The MCPRs are provided in the COLR.

(3) Specification 6.9.3.1, 6.9.3.2, 6.9.3.3, and 6.9.3.4 were added to the reporting requirements of the administrative controls section of the TS. These specifications require that the COLR be submitted to the NRC Document Control Desk, upon issuance, with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC-approved methodology and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

b. Brunswick reload methodologies approved by the NRC in Safety Evaluation of May 18, 1984 and October 22, 1984 for the following topical reports:

1. Topical Report NF-1583.01, "Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," February 1983.

2. Topical Report NF-1583.02, "Method of RECORD," February 1983.
3. Topical Report NF-1583.03, "Methods of PRESTO-B," February 1983.

4. Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," May 1983.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted to NRC, upon issuance, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC approved methodology, the NRC staff concludes that there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

2.2 Changes Related to Deleting the Linear Heat Generation Rate Limit from the TS

The licensee has followed the guidance provided by the staff in its Safety Evaluation (Ref. 5) on Amendment 19 to GESTAR-II with respect to deleting the linear heat generation rate (LHGR) limit from the TS. All affected Definitions, Specifications, and Bases have been appropriately changed. All applicable APLHGRs (average planar LHGRs) will include (as a function of fuel burnup and, when necessary, fuel bundle lattice type) the more limiting of either APLHGR based on ECCS analysis requirements or APLHGR based on fuel mechanical design analysis requirements. The proposed TS changes will result in the same operating power distribution limits and safety margins as the current TS. In addition, the proposed TS will reduce TS complexity by removing specifications which are in effect redundant and will preclude the need for inclusion of numerous lattice specific maximum average planar linear heat generation rate (MAPLHGR) limit curves. Because plant operation continues to be limited in accordance with appropriate thermal limits, the NRC staff concludes that the proposed TS changes are acceptable and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

2.3 Other TS Changes

The licensee has also made a number of editorial changes to the TS. The most important is the removal of the specific value of the safety limit Minimum Critical Power Ratio (MCPR) from a number of Specifications and Bases and referring to the safety limit MCPR of Specification 2.1.2. The staff concludes that these editorial changes are acceptable.

3.0 SUMMARY

We have reviewed the request by the Carolina Power & Light Company to modify the TS of the Brunswick Units 1 and 2 plants that would (1) remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specification, and (2) delete the redundant linear heat generation rate limit from the specifications. Based on this review we conclude that these TS modifications are acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

These amendments change a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site; and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 15822) on April 19, 1989, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Fieno

Dated: May 25, 1989

6.0 REFERENCES

1. Letter (NLS-87-185) from L.W. Eury (CP&L) to NRC, dated September 4, 1987.

2. Letter (NLS-88-062) from L.W. Eury (CP&L) to NRC, dated April 5, 1988.

3. Letter (NLS-89-038) from M.A. McDuffie (CP&L) to NRC, dated February 20, 1989.

4. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988.

5. Letter from Ashok C. Thadani (NRC) to J.S. Charnley (GE), "Acceptance for Referencing of Amendment 19 to General Electric Licensing Topical Report NEDE-24011-P-A (GESTAR-II), 'General Electric Standard Application for Reactor Fuel,' April 7, 1987," dated November 17, 1987.

6. Letter (NLS-89-055) from M. A. McDuffie (CP&L) to NRC, dated March 20, 1989.