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Docket No. 50-324

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 14 to Facility Operating License No. DPR-62 for the Brunswick Steam Electric Plant Unit 2. The amendment consists of changes to the Technical Specifications and is in response to your requests dated February 4 and April 27, 1976, as supplemented March 5, April 21 and 30, 1976.

This amendment (1) authorizes operation of Brunswick Steam Electric Plant Unit 2 with the lower core support bypass flow holes plugged, and (2) establishes operating limits for the four new fuel elements added during the April-May 1976 outage.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 14 to DPR-62
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

OT: RSB	OT: AD
R. Baer	[illegible]
5-10-76	5-10-76

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DATE	5/[illegible]/76	5/13/76	5/17/76	5/13/76	

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

Docket No. 50-324

May 13, 1976

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Executive Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

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2. Safety Evaluation
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cc: See next page

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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-62

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

   A. The applications for amendment by Carolina Power & Light Company (the licensee) dated February 4 and April 27, 1976, as supplemented March 5, April 21 and 30, 1976, comply 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. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(6) of Facility License No. DPR-62 is hereby deleted.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: May 13, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered revised pages:
   1.1-3 and 1.1-4
   1.1-5 and 1.1-6
   1.1-9 and 1.1-10
   1.1-11 and 1.1-12
   1.1-17 and 1.1-18
   3.1-1 and 3.1-2, 3.1-5 and 3.1-6
   3.1-13 and 3.1-14
   3.1-17 and 3.1-18
   Fig. 3.1-2A and Fig. 3.1-2B

2. Add new Fig 3.1-2C

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 Fuel Cladding Integrity (Cont'd)	2.1 Fuel Cladding Integrity (Cont'd)
C. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 inches above the top of the normal active fuel zone.	C. Reactor low water level #1 scram setting shall be ≥ 12.5" on level instruments. D. Turbine stop valve closure scram setting shall be ≤ 10 percent valve closure except that this is bypassed when power ≤ 30 percent. E. Turbine control valve 1. Fast closure - Results from low hydraulic oil pressure. 2. Loss of control oil pressure - setting shall be ≥ 850 psig. 3. For Brunswick Unit No. 2 - fast closure will initiate select rod insert and a reactor protection system trip. F. Main steam isolation scram setting shall be ≤ 10 percent valve closure. G. Main steam isolation on main steam line low pressure at inlet to turbine valves. Pressure setting shall be ≥ 850 psig. H. Reactor low water level #3 initiation of LPCI, core spray and auto blowdown shall be set at or above -147.5 inches indicated level. I. Reactor low water level #2 initiation of HPCI and RCIC shall be set at or above -38 inches indicated level.

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. 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 1.05. MCPR > 1.05 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding represents one of the physical barriers which separate radioactive materials from 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 protection system safety 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 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.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT (Cont'd)

the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented varables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The safety limit (MCPR of 1.05) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > 1.28) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit of 1.05 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1.

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.05 would not produce boiling transition.

However, if boiling transition were to occur, clad perforation would not necessarily be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Brunswick operated above the critical heat flux for a significant period of time without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit (MCPR = 1.05), operation is constrained to a maximum LHGR $\leq$ 18.5 Kw/ft. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.60. For the case of the MTPF exceeding 2.60, operation is permitted only at less than 100% of rated thermal

BASES:

1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT (Cont'd)

power and only with reduced APRM scram settings as required by Specification 2.1.A.1.

The actual power distribution in the core is established by specified control rod sequence and is monitored continuously by the incore local power range monitor (LPRM) system. However, to maintain applicability of the safety limit curves on Figure 2.1-1, the safety limits will be lowered according to the equations expressed in Specification 2.1 in the rare event of power operation with a total peaking factor in excess of 2.60.

At pressure below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 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 4.56 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 core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

Plant safety analyses have shown that if a scram occurs when a limiting safety system scram setting is exceeded, the safety limit of Specifications 1.1.A or B will not be exceeded.

During transient operation, the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel, which is eight to nine seconds. Also, the limiting safety system scram settings are at values which

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Brunswick Plant have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 2531 MWt at 100 percent recirculation flow. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 MWt is the licensed maximum power level of Brunswick, and this maximum steady-state power will never be knowingly exceeded.

Transient analyses were not performed for a power level that specifically included instrument errors. To permit appropriate conclusions from analyses which do not include instrument errors, conservatism was incorporated in the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamics performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The void reactivity coefficient utilized in the analysis is estimated to be about 25% more conservative than any value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 80% of the control rods. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect.

BASES:

2.1
LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

The time for 50 percent and 90 percent insertions are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR of 1.28 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the rated power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (2436 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal flux of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses have demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than 2.60.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.05 when the transient is initiated from MCPR > 1.28.

For operation in the startup mode while the reactor is at low pressure, APRM scram is set at ≤15 percent of rated power. This provides an adequate thermal margin between the setpoint and the safety limit, 25 percent rated power. The margin adequately accommodates anticipated maneuvers associated with 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 to be uniform by operating procedures backed up by the rod worth minimizer and the rod sequence control system.

Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable case of significant power rise. 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 percentage of rated power, the rate of power rise is slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

2. Select Rod Insert

Select rod insert is an operational aid designed to insert a predetermined group of control rods immediately following either a generator load rejection, loss of turbine control valve hydraulic pressure, or by manual operator action using a switch on the R-T-G board. The assignment of control rods to the select rod insert function is based on the startup and fuel warranty service associated with each control rod pattern, on RSCS considerations, and a dynamic function of both time and core patterns.

Approximately ten percent of the control rods in the reactor will be assigned to the select rod insert function by the operator. This selection will be accomplished by moving the rod scram test switch for those rods from the "NORMAL" position to the "SELECT ROD INSERT" position.

F & G. Main Steamline Isolation on Low Pressure and Main Steamline Isolation Scram

The low pressure isolation of the main steamlines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steamline isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not

BASES:

2.1

LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY (Cont'd)

occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steamline low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.1 Reactor Protection System

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant Operation

Plant operation at any power level shall be permitted only in accordance with Table 3.1-1.

B. System Response

The designated system response time from actuation of the sensor contact or trip output to the de-energization of the scram solenoid relay shall not exceed 100 milliseconds.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation, MCPR shall be $\geq$ 1.28 at rated power and flow. For core flows other than rated, the MCPR shall be > 1.28 times K_f, where K_f is as shown in Figure 3.1-1.

4.1 Reactor Protection System

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Plant Operation

Instrumentation systems shall be functionally tested and calibrated as indicated in Table 4.1-1.

B. System Response

The system response times will be checked prior to initial fuel loading.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Protection System (Cont'd)

D. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.1-2A, 3.1-2B, or 3.1-2C.

E. Local Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d[1 - \{(\Delta P/P)_{max}(L/L_T)\}]$$

$LHGR_d$ = Design LHGR = 18.5 KW/ft.

$(\Delta P/P)_{max}$ = Maximum power spiking penalty

= 0.026

L_T = Total core length = 12 feet

L = Axial position above bottom of core

SURVEILLANCE REQUIREMENTS

4.1 Reactor Protection System (Cont'd)

D. Average Planar Linear Heat Generation Rate (APLHGR)

The maximum ratio of the limiting value for APLHGR as a function of average planar exposure to the APLHGR value (APLHGR RATIO) for each type of fuel shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power.

E. Local Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

TABLE 3.1-1 (Cont'd)

Trip Function	Trip Settings	Modes in Which Functions Must be Operable Refuel (1)	Startup	Run	Min No. Operable Instrument Channels Per Trip System (2)	Required Conditions When When Minimum Conditions for Operations are Not Satisfied (3)
First stage turbine pressure permissive C72-PS-N003A,B,C,D	(9)			X	2	D
12. Turbine control valve fast closure EHC-PSL-1756 EHC-PSL-1757 EHC-PSL-1758 EHC-PSL-1759	≥ 850 psig (8) control oil pressure	X	X	X	2	D

NOTES:

(1) When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

A. Mode switch in SHUTDOWN

B. Manual scram

C. High flux IRM

D. Scram discharge volume high water level

It is possible during reactor operation to switch to the refuel mode and remain critical. The requirement to have all other scram functions operable in the refuel mode is therefore to assure that shifting to this mode during reactor operation does not diminish the protection afforded by the RPS.

(2) There shall be two operable, one operable and one tripped, or two tripped trip systems for each function.

NOTES (Cont'd)

(3) When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met, the appropriate actions listed below shall be taken:

A. Initiate insertion of operable rods and complete insertion of all operable rods within eight hours.

B. Reduce power level to IRM range and place mode switch in the STARTUP position within eight hours.

C. Reduce turbine load and close main steam line isolation valves within eight hours.

D. Reduce reactor power to less than 30% of rated within eight hours.

(4) "W" is the reactor driving loop flow in percent of rated (see Specification 2.1.A.1).

(5) To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 11 LPRM inputs.

(6) Twelve and one half inches on the water level instrumentation is 177 inches above the top of the active fuel.

(7) A main steam isolation valve closure bypass is permitted when the reactor mode switch is in either the SHUTDOWN, REFUEL, or STARTUP position.

(8) For Unit 2, low control oil pressure initiates select rod insert, and has the provision to delay reactor protection system trip until determination of turbine bypass valve status. The time delay for bypass valve status determination shall be set at 0.00 sec. In both units, this scram is bypassed if the first stage turbine pressure is less than 30 percent of normal rated power.

(9) A turbine stop valve closure bypass is permitted when the first stage turbine pressure is less than 30 percent of normal rated power.

(10) Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

(11) Not required to be operable when the primary containment integrity is not required.

(12) IRM's are bypassed when APRM's are on scale and the reactor mode switch is in the RUN position.

(13) The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and $\leq$ 120/125 of full scale. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

(14) The APRM high flux signal is fed through a time constant curcuit of approximately 6 seconds. The APRM fixed scram does not incorporate the time constant, but responds directly to instantaneous neutron flux.

BASES:

3.1 Limiting Condition for Operation for Reactor Protection System (Cont'd)

8. The control rod drive scram system is designed so that all of the water which is discharged from the reactor by the scram can be accommodated in the discharge piping. The scram discharge volume consists of two vertical tanks on opposite sides of the Reactor Building connected by a two-inch diameter pipe. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the discharge volume which scram the reactor when the volume of water reaches 109 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

9. The main steamline radiation monitoring system monitors for a gross release of fission products from the fuel and, upon indication of such release, initiates a reactor scram and isolation action to contain any fission products released from the fuel. The high radiation trip setting is selected so that a high radiation trip results from the fission products released in the design basis rod drop accident. An alarm setting at one half the trip setting actuates an alarm in the control room before scram and steam line isolation is effected.

BASES:

3.1 Limiting Condition for Operation for Reactor Protection System (Cont'd)

10. The main steamline isolation valve closure scram is set to scram when the isolation valves are 10 percent closed from full open in three-out-of-four lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant. This scram is bypassed when the reactor mode switch is in either the SHUTDOWN, REFUEL, or STARTUP position.

11. Turbine stop valve closure scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve. This scram is bypassed if reactor power is less than 30 percent power.

12. The turbine control valve fast closure scram is based on pressure switches sensing the electrohydraulic control (EHC) system oil pressure. The switches are set relative to the normal EHC oil pressure such that, based on the small system volume, they can rapidly detect loss of hydraulic pressure.

    A generator load rejection decreases control oil pressure to initiate turbine control valve fast closure and a reactor scram. This scram signal is automatically bypassed on both units whenever the turbine first stage pressure is below 30 percent of rated power.

    The thermal-hydraulic analysis for BSEP Unit 2 with plugged lower core plate flow bypass holes was performed without the 200 msec time delay in the generator load rejection scram logic. This was done to obtain a lower operating MCPR limit than would otherwise be necessary. Therefore, this time delay has been eliminated (set to zero).

BASES:

4.1 Surveillance Requirement for Reactor Protection System

A. The scram sensor channels listed in Table 4.1-1 are divided into three groups: A, B, and C.

Group A sensors are of the on/off-type and will be tested and calibrated at indicated intervals.

Group B devices utilize an analog sensor followed by an amplifier and bistable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bistable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during startup and inactive during full power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The system response times will be checked prior to initial fuel loading to ensure adequate reactor protection.

C. At core thermal power levels less than or equal to 25%, 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 and thermal hydraulic analysis 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 the 25% thermal power level with minimum recirculation pump speed.

BASES:

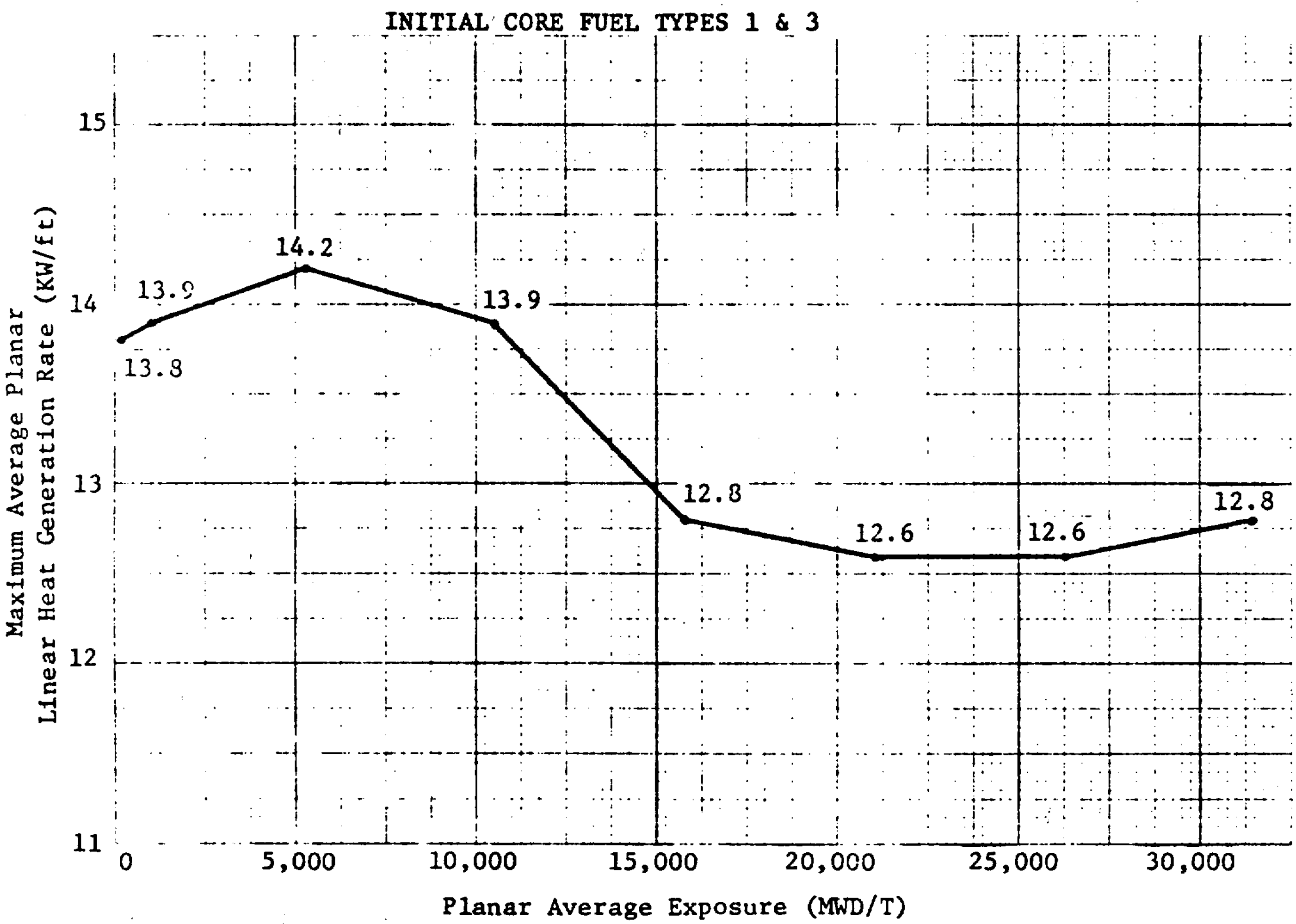
4.1.C Surveillance Requirement for Reactor Protection System (Cont'd)

The MCPR margin will thus be demonstrated such that future MCPR evaluations 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.

D. This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

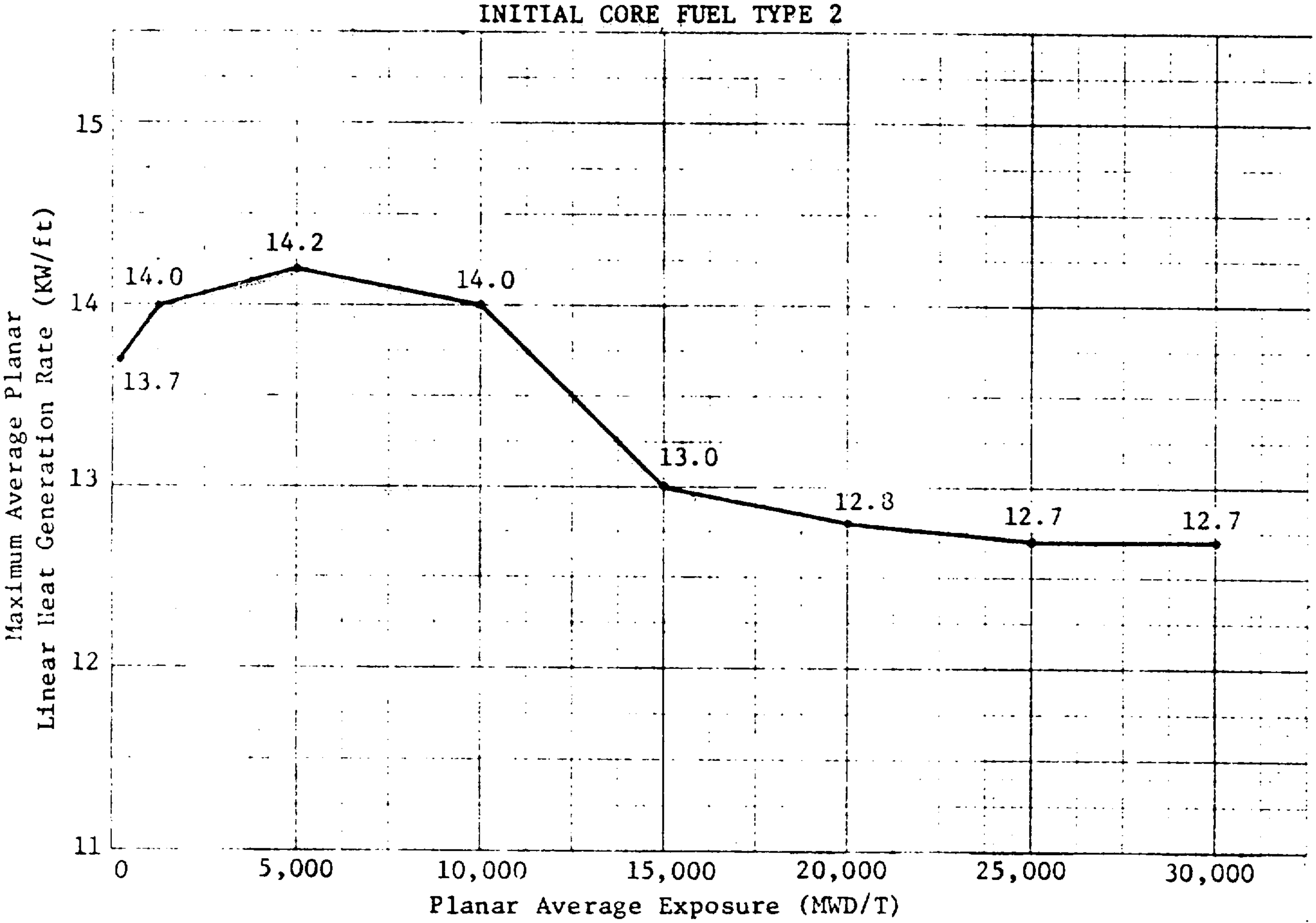
The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm 20°F$ relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.1-2A for fuel types 1 and 3, Figure 3.1-2B for fuel type 2 and Figure 3.1-2C for the interim replacement fuel assemblies.

E. This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation in



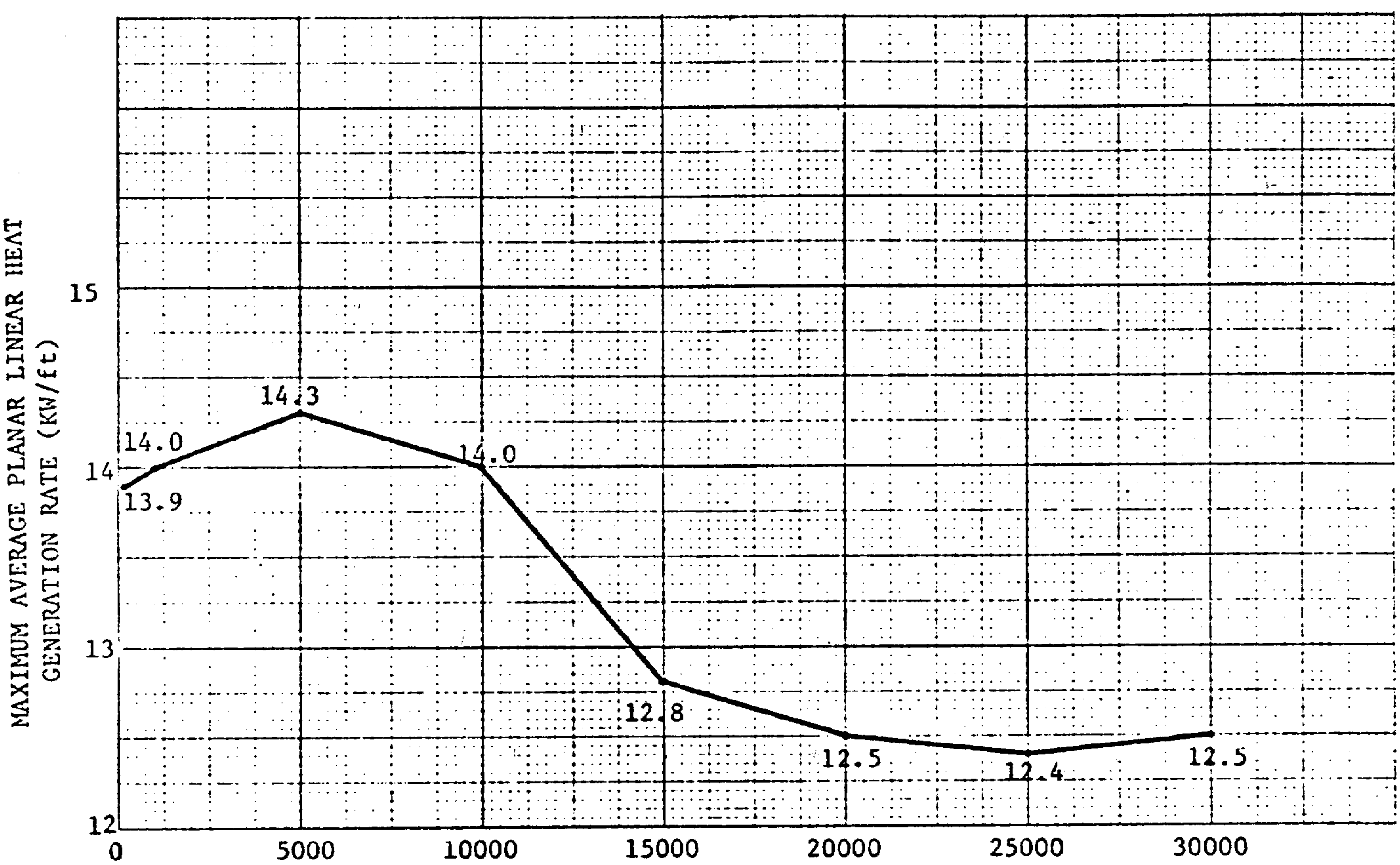
CAROLINA POWER & LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 & 2 Final Safety Analysis Report	
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE 7 x 7-Plugged Core Plate	
FIG. NO.	3.1-2A

INITIAL CORE FUEL TYPE 2



CAROLINA POWER & LIGHT COMPANY BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 & 2 Final Safety Analysis Report	
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE **7 x 7** - Plugged Core Plate	
FIG. NO.	3.1-2B

INTERIM RELOAD REPLACEMENT FUEL



PLANAR AVERAGE EXPOSURE (MWd/T)

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT
UNITS 1 & 2
Final Safety Analysis Report

(Plugged Core Plate)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) vs. PLANAR PLANAR AVERAGE EXPOSURE

FIG. NO. 3.1-2C

Amendment No. 14

SAFETY EVALUATION

OF THE

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-324

SUPPORTING AMENDMENT NO. 14

TO

FACILITY LICENSE NO. DPR-62

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2

(BSEP-2)

SAFETY EVALUATION REPORT

WITH PLUGGED BYPASS FLOW HOLES

1.0 Introduction

Carolina Power and Light Company submitted References 1 through 6 to the NRC in support of its license amendment to continue operation of the BSEP-2 plant for the remainder of cycle 1. The principal changes are the plugging of the bypass flow holes in the core support plate in order to reduce instrument tube-fuel channel interaction and the use of four replacement fuel assemblies.

2.0 Summary

The NRC staff has reviewed the proposed operation of BSEP-2 with plugged bypass holes and has concluded that BSEP-2 can be operated without undue risk to the health and safety of the public provided that the facility is operated in accordance with the operating restrictions and Technical Specification changes as presented in Appendix A of this Safety Evaluation Report. The NRC staff has concluded the following:

a. The nuclear, mechanical and thermal-hydraulic characteristics of the core are acceptable.

b. The use of plugged bypass flow holes will significantly reduce instrument tube-channel interaction that has caused excessive wear on some channels.

c. The overpressurization protection satisfies ASME code requirements for the reactor coolant system.

d. Safety analyses show that the core will not violate limiting thermal margins if the plant is operated with a steady-state MCPR equal to or greater than 1.28.

e. The MAPLHGR limits referenced herein, which are based on calculations performed with previously approved models, are acceptable.

3.0 Nuclear Design

The primary effects of plugging the bypass flow holes on the nuclear design are an increase in bypass void fraction and a reduction in the average in-channel void fraction.

At steady-state conditions, the increased bypass void fraction results in a small reduction in the maximum local peaking factor within a fuel bundle and an increase in the local bundle power calculational uncertainty. Another consequence of the reduced bypass flow is a small reduction in the infinite multiplication factor of uncontrolled fuel.

The presence of voids in the bypass region affects the relationship between the TIP signal and the local bundle power. The TIP signal is reduced by the presence of voids and could lead to an underprediction of the peak heat flux. The relationship of the power in the four bundles surrounding a TIP instrument tube and the TIP signal as a function of bypass voids was determined by GE by performing three group, two-dimensional diffusion theory calculations. A correction factor was developed and algorithms for computing the bypass void fraction and for making appropriate corrections in the local bundle power have been incorporated in the process computer.

The uncertainty in the local bundle power caused by bypass voids is taken into account in determining the MCPR safety limit. The TIP uncertainty introduced by the bypass voids is zero in the bottom half of the core and increases from 3.6% at the core midplane to 4.2% at the core exit.(1)

After the bypass flow holes are plugged, most of the fuel will be placed in its original core locations. Four replacement 7x7 fuel bundles will be placed on the periphery and 48 bundles will be moved in the reactor to maintain quadrant symmetry.(6) Such fuel shuffling is necessary to replace fuel damaged by a dropped fuel bundle. Rod patterns and withdrawal sequences will not be changed. The following observations can be made:

(1) the control rod worths are not significantly changed and, consequently, the previous results of the control rod drop analysis remain valid,

(2) the shutdown margin will remain the same as previously analyzed,

(3) the fuel storage margins are unaffected, and

(4) the standby liquid control system reactivity insertion rate and magnitude will not be adversely affected.

We have reviewed the proposed core configuration and find it to be a minor change from the original core. We conclude that the analysis of the nuclear performance of the plant with plugged bypass holes is acceptable.

4.0 Mechanical Design

4.1 Bypass Flow Hole Plugs

The only mechanical design change in the core is the use of plugs to fill the bypass flow holes. The plug primarily consists of a stainless steel body and shaft which are positioned by an Inconel spring. The shoulder of the body rests on the top of the core plate along the rim of a one-inch bypass hole and is secured by the spring. An equal and opposite force is applied on the shaft. A stainless steel latch is connected to the bottom of the shaft by means of a pin. This latch is free to rotate about the pin and latches the shaft to the core plate. Upon installation the spring exerts a minimum of 35 pounds on the body and latch and a maximum of 46 pounds (with the worst tolerance combination).(1)

Removal of a plug can be accomplished by applying about 500 pounds of force and deforming the latch plastically. More than 10 plugs were removed in tests performed at the GE test facility with consistent latch deformations without damaging other parts.

Plugs identical to those to be used in the BSEP-2 reactor have been installed in the Vermont Yankee reactor and eleven other reactors. The plugs installed in Vermont Yankee reactor were removed during a refueling operation after 10 months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be on the order of 27 to 45 psi. These loads together with the spring preload will produce yielding of the latch in bending but will be significantly below the 500 pounds of force necessary for removing the plug. The 1973 GE full scale flow mockup test shows that leakage flow through the plugged holes remains at less than 2% of the unplugged configuration. No plug vibration was observed during the test and no apparent deformation on the latch was evident after the test.

Stainless steel and Inconel are compatible with other reactor internals and are not expected to introduce any unusual oxidation and stress corrosion problems. The flux level at the core plate elevation is quite low and an insignificant reduction in ductility due to irradiation is anticipated. GE has performed creep tests with both Inconel springs and stainless steel latches and found that stress relaxation or creep deformation were insignificant. The tests were performed at 550^{O}F.

Carolina Power & Light Company presented to the NRC staff a summary of channel inspections on BWR-2s and BWR-3s. These older plants have instrument tubes similar to BSEP-2, but no bypass flow holes in the core support plate. The bypass flow for these plants enters through clearances in the assembly end fittings, which is similar to the proposed BSEP-2 configuration with plugged bypass holes. One hundred sixty-four channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in 7 plants. No significant channel wear was observed at the corners adjacent to the instrument tubes.

The Duane Arnold reactor has the same 1-inch bypass holes in the lower core support plate which are being plugged in BSEP-2. The Duane Arnold bypass holes were plugged during a mid-1975 shutdown. Subsequently, Duane Arnold operated about seven months before the current shutdown during which the channel boxes were inspected. The condition of the channel box corners was observed to be equivalent to the corner conditions observed in BWR/3 reactors having no 1-inch bypass holes in the lower core support plate.

Based on a review of the design, the test rig, the installation methods and primarily the previously successful operating experience at Vermont Yankee, Pilgrim, and Duane Arnold reactors, we conclude that the plugs will not fail so as to result in loose parts in the core or result in unplugging of the bypass flow holes. Also, we conclude that the installed plugs will substantially reduce the instrument tube vibration (due to flow through the bypass holes) to preclude any unacceptable wear for at least the proposed fuel cycle.

4.2 Instrument Tube-Channel Box Interaction Surveillance

Excessive instrument tube-channel interaction previously has been determined from the noise level in the incore instrumentation. The noise content in the 1.4 to 3 Hz frequency range caused by vibration of the LPRM instrument tube should be reduced, by plugging the 1-inch bypass holes, relative to the power dependent noise content. Some increase in the boiling noise, 5 to 50 Hz range, is expected because of boiling in the bypass water region.

Before the plant shutdown, TIP traces were obtained for several combinations of power and flow. These data will provide a basis for evaluating the efficiency of plugging the bypass flow holes. After reactor startup, comparison of similar measurements with pre-shutdown data will be made at BSEP-2 to confirm that the vibration of the instrument tubes has been substantially reduced.

Carolina Power & Light Company has committed to conduct a post-plugging surveillance program to monitor instrument tube - channel box interaction. This program consists of instrument tube vibration monitoring using TIP traces, and is acceptable.

5.0 Stability of the Core

The plugged bypass flow holes increase the core hydraulic resistance which reduces the recirculation flow rate by 2 percent. However, the assembly flow rates are increased while the total bypass flow is decreased.

The stability of the core was analyzed based on the most limiting conditions of natural circulation and 51.5% power. The analysis, which is similar to that reported in the FSAR, showed that the decay ratios for both the channel and the core decreased from the values presented in the FSAR. Based on the analyses presented, operation with plugged bypass holes results in improved stability for the channel performance and core performance.

6.0 GETAB

6.1 Safety Limit MCPR

The fuel cladding integrity safety limit MCPR for the 7x7 fuel, including the effects of greater bypass region voiding and uncertainty in the effect of that voiding on the TIP readings, is 1.05. This phenomenon is taken into account by inclusion of an additional uncertainty factor of 3.63% to 5.24% in the GETAB analysis to account for the bypass void effect on the TIP readings[1]. We find this additional uncertainty and the resulting 1.05 MCPR safety limit acceptable.

6.2 Operating Limit MCPR

The licensee re-analyzed three abnormal transients - turbine trip, loss of feedwater heater, and rod withdrawal error - as the most limiting events to be considered. The main factors affecting the plant transient analyses are the moderator void coefficient of reactivity, the Doppler coefficient of reactivity, and the full power scram reactivity function.

- The Doppler coefficient of reactivity is affected by the changes in the moderator density in the fuel channel and bypass region primarily through changes in the Dancoff Ginsburg rod shadowing effect. This effect is small and insignificantly affects the Doppler coefficient of reactivity.

- The full power scram reactivity function for the end-of-cycle with plugged bypass flow holes, including an acceptable conservatism, is shown in figure 7-1 of reference (1).

- The moderator void coefficient of reactivity used in the safety analysis of BSEP-2 with plugged bypass flow holes is more negative than used in the FSAR for two reasons. The first cause is a renormalization of the void coefficient calculations based on analyses of operating BWR data. This effect, of the order of 15 to 20 percent, is unrelated to the plugging of the bypass flow holes. The second cause is the increase in the amount of voids present in the bypass region after the bypass flow holes are plugged.

The limiting transient is a turbine trip with failure of bypass valves to open. The analysis was initiated from 104 percent design power and the scram was initiated by the position switch on the turbine stop valves. A peak pressure of 1209 psig was calculated at the bottom of the vessel. The decrease in MCPR is 0.23 which is the limiting change in thermal margin. This was calculated assuming zero scram delay following stop valve closure; consequently, the plant will be allowed to operate only with the previous built-in delay of 200 msec eliminated, i.e. with a Technical Specification allowing only zero delay (see Appendix A). This represents an increase over the unplugged case (Δ MCPR = 0.22) due largely to changes in the void distribution (more voids in the bypass region, less in the active fuel) which results in a more negative overall core void coefficient which makes overpressure (void collapse) transients more severe. The ΔMCPR change would have been more severe than 0.01 (0.23 compared to 0.22) except for the compensating effect of elimination of the scram delay.[4] The loss of feedwater heater transient was shown to be less severe, with a ΔMCPR of 0.13 which is acceptable.

The rod withdrawal error was analyzed for a limiting control rod pattern. The results of the analysis indicate that a Rod Block Monitor (RBM) setpoint of 107% of full power will provide, for the worst case failure of Local Power Range Monitor (LPRM) detectors, a rod block at approximately 6 feet of rod withdrawal for the withdrawing rod. The MCPR at this point will be about 1.10 and the cladding strain will be less than 1.0%.

The staff has concluded that the dynamic events re-analysis has correctly identified a conservative operating limit MCPR of 1.28 for the turbine trip without bypass. This conclusion is based on analyses which correctly included "neutron effective void" coefficient and required no additional correction as were required in previous unplugged core analyses.

At less than rated power or flow, the previous conclusions[7] apply, except the required MCPR is now 1.28 (not 1.27) times the k_f factor in order to assure that the safety limit MCPR of 1.05 is not exceeded.

We find the transient analyses for the plugged core to be acceptable, and continued operation of the plant for the remainder of cycle 1 is acceptable with the plant not to operate below a MCPR of 1.28 as stated in Appendix A.

7.0 Overpressure Protection

The licensee referenced a generic BWR-4 overpressure analysis and justified its conservative applicability to BSEP-2 in order to demonstrate that an adequate margin exists below the ASME code allowable pressure of 110% of vessel design pressure.(4) The limiting transient was identified as the closure of all main steam isolation valves with high neutron flux scram. The results presented were stated to be conservative for 105% power with the end of cycle scram reactivity insertion rate, scram initiated by high neutron flux, void reactivity applicable to the initial (current) fuel cycle, no credit for relief valve operation, and all safety valves operable. The peak pressure at the bottom of the vessel was calculated to be 1285 psig yielding a margin of 90 psig below the allowable 1375 psig ASME code limit (110% of the 1250 psig design pressure). In addition, the licensee provided results of a sensitivity study performed for BWR-4 reactors indicating that for one failed safety valve the results would increase less than 20 psi which would still leave a margin of 70 psi for the required analysis with one faiied valve.

We find the overpressure analysis acceptable on the basis that the sensitivity study with one failed valve shows considerable margin below the allowable limit.

8.0 ECCS Appendix K Analysis

The re-analyses of BSEP-2 for the plugged core case has referenced the unplugged BSEP-2 analyses as the "lead" plant for break spectrum, location, and single failure analyses. Justification for this lead plant reference was provided by a review of the LOCA analysis performed by the applicant to determine the effects of plugging the bypass flow holes using the LAMB and SCAT codes [1] on a BWR/4 similar to BSEP-2. The difference in peak cladding temperature (PCT) versus time for both the plugged and unplugged configurations at the same MAPLHGR showed that the difference in PCT at any time during the transient was less than 30°F. The difference in PCT at its maximum value was less than 10°F. Also, it was shown that the duration of nucleate boiling was essentially unchanged; lower plenum flashing occurred only about 0.2 seconds later with the bypass flow holes plugged; heat transfer coefficients during lower plenum flashing were approximately the same (the unplugged case was slightly higher, but the difference never exceeded about 10 Btu/hr-ft^2-°F); and time to uncovery was essentially unchanged. These results demonstrate the acceptability of using the BSEP-2 unplugged analysis as the lead plant analysis for the plugged BSEP-2 plant.

The major effect of plugging the bypass flow holes in BSEP-2 is to retard the calculated reflood time for large breaks. This occurs because the limiting break configuration (discharge line break with LPCI injection valve failure) is strongly influenced by counter-current flow (CCFL) phenomena. There is no LPCI flow, since one loop is assumed lost through the break and the other is lost due to injection valve (assumed single) failure, leaving only spray flow which must flow downward through the core. This downward flow through the fuel assemblies is limited by the CCFL phenomena for the plugged or unplugged cases, but the flow that passed through the bypass region into the lower plenum in the unplugged case is now severely reduced by the plugged bypass holes. This delayed reflood time is properly taken into account in the new plugged core analysis. A reflood delay of approximately 76 seconds is predicted due to the plugged holes. This causes a reduction in the MAPLHGR values of approximately 7%.

For small breaks, the limiting single failure is the HPCI. For this case at least one LPCI is available. However, since the LPCI flow is unaffected by CCFL the reflooding time is not significantly affected for small breaks.

The applicant submitted acceptable additional LOCA analyses for the four replacement fuel bundles used to replace the four damaged fuel bundles. MAPLHGR limits for those bundles are given in Figure 3.1-2C of reference 6.

We have reviewed the evaluation of the ECCS performance submitted by Carolina Power and Light Co for BSEP-2 and conclude that the analysis has been performed wholly in conformance with the requirements of 10 CFR 50.46. Therefore, operation of the reactor would meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) specified in Appendix A to this SER.

9.0 Surveillance

As noted above in Section 4.2, CP&L has committed to conducting a post-plugging surveillance program to monitor instrumentation tube-channel box interaction to confirm that the mechanical vibration of the instrument tubes has been substantially reduced.

10.0 Conclusion

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR § 51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have 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 the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 13, 1976

References

1) CP&L letter dated March 5, 1976, with Attachment A (NEDO-21200, "Brunswick Steam Electric Plant Unit 2 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged, February 1976) and Attachment B (Tech. Spec. Revisions).

2) NEDE-21156, "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration," January 1976.

3) NEDC-21118, "Brunswick Steam Electric Plant Unit 2 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations," November, 1975.

4) NEDO-21200 Additional Information, letter to B. C. Rusche, Director, NRR, from J. A. Jones, Executive Vice President, CP&L Co., April 21, 1976.

5) Letter to B. C. Rusche, Director, NRR, from E. E. Utley, CP&L, Response to Request for Additional Information, April 30, 1976.

6) Letter to B. C. Rusche, Director, NRR, from E. E. Utley, CP&L, Request for License Amendment-Revision to Technical Specifications, April, 1976.

7) Letter to R. C. DeYoung from Victor Stello, Jr., AD for Reactor Safety, Review of Brunswick 2, Appendix K and GETAB Calculations: (TAR-1633), August 22, 1975.

APPENDIX A

BRUNSWICK STEAM ELECTRIC PLANT UNIT NO. 2

TECHNICAL SPECIFICATIONS CHANGES

Limitations on the continued operation of the reactor for the remainder of Cycle 1 are presented below. Operation shall conform to a MCPR value of 1.28, as proposed by the licensee.[1] The limiting values of MAPLHGR included with the proposed Technical Specifications submitted (attachment B of reference 1) properly account for the proposed operation with plugged bypass holes. The revised values are given in Figures 3.1-2A and 3.1-2B.[1] The limiting MAPLHGR for the four replacement fuel assemblies is given by Figure 3.1-2C of reference 6.

Additionally, since the transient analyses presented in support of an operating limit MCPR of 1.28 assumed zero scram delay following a turbine trip without bypass event, the Technical Specifications have been modified to preclude operation with non-zero scram delay (CP&L has the capability of including a time delay for the purpose of preventing a scram following turbine trip if the 100% bypass capacity valves open properly) - CP&L has agreed with this change.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-62 issued to the Carolina Power and Light Company, which revised Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit 2, located in Brunswick County, North Carolina. The amendment is effective as of the date of issuance.

This amendment (1) authorizes operation of Brunswick Steam Electric Plant Unit 2 with the lower core support bypass flow holes plugged, and (2) establishes operating limits for the four new fuel elements added during the April-May 1976 outage.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with item (1) above was published in the Federal Register on February 19, 1976 (41 FR 7594). No request for a hearing or petition for leave to intervene was filed following notice of this proposed action. Prior public notice of item (2) above is not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated February 4 and April 27, 1976, as supplemented March 5, April 21 and 30, 1976, (2) Amendment No. 14 to License No. DPR-62, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. 20555, and at the Southport-Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

A single copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 13th day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors