ENCLOSURE 8

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
REQUEST FOR LICENSE AMENDMENT
FUEL CYCLE 12 RELOAD LICENSING

TYPED TECHNICAL SPECIFICATION PAGES - UNIT NO. 1

2.1 SAFETY LIMITS

THERMAL POWER (Low Pressure or Low Flow)

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER (High Pressure and High Flow)

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 with the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 and the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure ≤ 1325 psig within 2 hours.

CORE OPERATING LIMITS REPORT (Continued)

b. The core flow and core power adjustments for Specification 3.2.2.1.

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.2.1 and 3.2.2.2.

d. The rod block monitor upscale trip setpoint and allowable value for Specification 3.3.4.

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.

c. Deleted.

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.

ENCLOSURE 9

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
REQUEST FOR LICENSE AMENDMENT
FUEL CYCLE 12 RELOAD LICENSING

MARKED-UP TECHNICAL SPECIFICATION PAGES - UNIT NO. 1

2.1 SAFETY LIMITS

THERMAL POWER (Low Pressure or Low Flow)

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 800 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER (High Pressure and High Flow)

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 ~~[illegible]*~~ with the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 ~~[illegible]~~ and the reactor vessel steam dome pressure greater than 800 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure ≤ 1325 psig within 2 hours.

~~*MCPR values in Technical Specification 2.1.2 are applicable only for Cycle 11 operation.~~

CORE OPERATING LIMITS REPORT (Continued)

b. The core flow and core power adjustments for Specification 3.2.2.1.

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.2.1 and 3.2.2.2.

d. The rod block monitor upscale trip setpoint and allowable value for Specification 3.3.4.

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.

c. Deleted ~~The NRC Safety Evaluation for Brunswick Unit 1 Amendment No. 182.~~

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.

ENCLOSURE 10

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
REQUEST FOR LICENSE AMENDMENT
FUEL CYCLE 12 RELOAD LICENSING

TYPED PAGE REVISION TO PREVIOUSLY SUBMITTED
IMPROVED TECHNICAL SPECIFICATION (ITS) CONVERSION — UNIT NO. 1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. In conjunction with LCOs, the limiting safety system settings, defined in LCO 3.3.1.1 as the Allowable Values, establish the threshold for protective system action to prevent exceeding acceptable limits, including this reactor vessel water level SL, during Design Basis Accidents. 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 irradiated 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 becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 2). Therefore, it is required

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. 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:

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

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

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ENCLOSURE 11

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
REQUEST FOR LICENSE AMENDMENT
FUEL CYCLE 12 RELOAD LICENSING

<u>MARK-UP FOR REVISION TO PREVIOUSLY SUBMITTED</u>
<u>IMPROVED TECHNICAL SPECIFICATION (ITS) CONVERSION — UNIT NO. 1</u>

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 ~~NOTE~~
~~MCPR SL values are only applicable for Cycle 11 operation.~~

With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ ~~1.10~~ 1.09 for two recirculation loop operation or ≥ ~~1.12~~ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

BASES

APPLICABLE SAFETY ANALYSES (continued)	2.1.1.3 Reactor Vessel Water Level During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. In conjunction with LCOs, the limiting safety system settings, defined in LCO 3.3.1.1 as the Allowable Values, establish the threshold for protective system action to prevent exceeding acceptable limits, including this reactor vessel water level SL, during Design Basis Accidents. 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 irradiated 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 becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.
SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations. ~~The MCPR SL values are based on an NRC approved methodology that uses cycle specific input parameters. As a result, SL 2.1.1.2 is modified by a Note which restricts use of the MCPR values in SL 2.1.1.2 to Cycle 11 operation only.~~
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 2). Therefore, it is required

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. 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:

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

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

~~5. NRC Safety Evaluation for Brunswick Unit 1 Amendment No. 182.~~

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