April 14, 199[illegible]

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: ISSUANCE OF AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING RESOLUTION OF GENERIC LETTER 90-06, "RESOLUTION OF GENERIC ISSUE 70, 'POWER-OPERATED RELIEF VALVE AND VALVE RELIABILITY,' AND GENERIC ISSUE 94,' 'ADDITIONAL LOW-TEMPERATURE OVER-PRESSURE PROTECTION FOR LIGHT-WATER REACTORS' PURSUANT TO 10 CFR 50.54(f)," FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, (TAC NO. M83963)

Dear Mr. Hinnant:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 162 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated June 18, 1992, as supplemented December 8, 1992, and February 3, 1995.

The amendment adds limiting conditions of operation and surveillance requirements for the pressurizer power-operated relief valves and their associated block valves whenever average temperature (Tavg) is above 350 degrees F or the reactor is critical. Technical Specifications have also been added for low-temperature overpressure protection whenever Tavg is less than 350 degrees F and the reactor coolant system is not vented to the containment.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

(Original Signed By)
Brenda Mozafari, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures: 1. Amendment No. 162 to DPR-23
2. Safety Evaluation

cc w/enclosures: See next page

CP-1

DOCUMENT NAME: P:\ROB83963.AMD *See previous concurrence

OFFICE	LA:PDII-1	PM:PDII-1	D:PDII-1	OGC*	SRXB*	EMEB*
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cc:

Mr. R. E. Jones
General Counsel - Legal Department
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Karen E. Long
Assistant Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
2112 Old Camden Road
Hartsville, South Carolina 29550

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., N.W., Ste. 2900
Atlanta, Georgia 30323

Mr. Dale E. Young
Plant General Manager
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
3581 West Entrance Road
Hartsville, South Carolina 29550

Public Service Commission
State of South Carolina
Post Office Drawer 11649
Columbia, South Carolina 29211

Mr. R. M. Krich
Manager - Regulatory Affairs
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

Mr. Dayne H. Brown, Director
Department of Environmental,
Health and Natural Resources
Division of Radiation Protection
Post Office Box 27687
Raleigh, North Carolina 27611-7687

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

Mr. Max Batavia, Chief
South Carolina Department of Health
Bureau of Radiological Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. H. W. Habermeyer, Jr.
Vice President
Nuclear Services Department
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. DPR-23

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated June 18, 1992, as supplemented December 8, 1992, and February 3, 1995, 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 for the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 162 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

for

David B. Matthews, 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: April 14, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

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
3.1 - 3c	3.1 - 3c
-	3.1 - 3d
-	3.1 - 3e
-	3.1 - 3f
-	3.1 - 3g
-	3.1 - 3h
3.1 - 4	3.1 - 4
-	3.1 - 4a
-	3.1 - 4b
3.1 - 5	3.1 - 5
3.1 - 5a	3.1 - 5a
3.1 - 6	3.1 - 6
3.1 - 6a	3.1 - 6a
3.1 - 7	3.1 - 7
3.1 - 8	-
3.1 - 9	-
3.1 - 10	-
3.1 - 11	3.1 - 11
4.2 - 6	4.2 - 6
-	4.2 - 7
-	4.2 - 7a
-	4.2 - 7b
-	4.2 - 7c
-	4.2 - 7d

3.1.1.4 Reactor Coolant System (RCS) Vent Path

A. When the RCS temperature is greater than 200°F, the RCS vent paths consisting of at least two valves in series powered from emergency buses, shall be operable (except that valves RC-567, 568, 569, and 570 shall be closed with power removed from the valve actuators) from each of the following locations:

   1. Reactor Vessel Head

   2. Pressurizer Steam Space

B. When the RCS temperature is greater than 200°F, RCS vent path valves RC-571 and 572 shall be closed, except that they may be periodically cycled to depressurize the RCS vent system should leakage past RC-567, 568, 569, or 570 occur.

C. With less than the above required equipment operable, perform the following as applicable:

   1. With the Reactor Vessel Head vent path inoperable, restore the vent path to operable status within 30 days or be in Hot Shutdown within 6 hours and Cold Shutdown within the following 30 hours.

   2. With the Pressurizer Steam Space vent path inoperable, restore the vent path to operable status within 30 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause of the inoperable vent path, the action being taken to restore the vent path to operable status, the estimated date for completion of repairs, and any compensatory action being taken while the vent path is inoperable.

3. With both the Reactor Vessel Head and Pressurizer Steam Space vent paths inoperable, restore at least one vent path to operable status within 7 days or be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.1.1.5 Relief Valves

Whenever T_{avg} is above 350°F or the reactor is critical both power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.[1]

a. With one or both PORVs inoperable because of leakage through the PORV resulting in excessive RCS leakage, i.e., not in accordance with the leakage criteria in Technical Specification 3.1.5.2:

1. Within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); or

2. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.[2]

---

1 PORV block valves shall not be considered inoperable solely because either their normal or emergency power source is inoperable.

2 Power operation may continue pursuant to the requirements of this specification with the associated block valve closed, as a precautionary measure, to isolate minor leakage prior to the RCS leakage exceeding the leakage criteria in Technical Specification 3.1.5.2, with power maintained to the block valve during the period of the discretionary isolation.

b. With one PORV inoperable due to causes other than (1) leakage through the PORV resulting in excessive RCS leakage or (2) discretionary isolation to prevent minor leakage from becoming excessive:

1. Within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; and

2. Restore the PORV to OPERABLE status within the following 72 hours; or

3. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.

c. With both PORVs inoperable due to causes other than (1) leakage through the PORV resulting in excessive RCS leakage or (2) discretionary isolation to prevent minor leakage from becoming excessive:

1. Within 1 our either restore at least one PORV to OPERABLE status; or close its associated block valve and remove power from the block valve; and

2. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.

d. With one or both block valves inoperable[1]:

1. Within 1 hour restore the block valve(s) to OPERABLE status or place the associated PORV(s) in manual control; and

2. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; and

3. Restore any remaining inoperable block valve to operable status within 72 hours; or

4. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.

e. For this specification, reactor startup, heatup and entry into operational conditions with T_{avg} greater than or equal to 350°F may continue so long as the limits of the associated action statements are met.

f. During performance of the required surveillance testing of the PORVs and their associated block valves, the respective valve train need not be declared inoperable nor the associated action statements performed unless the associated valves are determined to be inoperable via this testing. Testing of no more than one train at a time may be performed and the time in the out of normal test configuration shall not exceed 24 hours.

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1 PORV block valves shall not be considered inoperable solely because either their normal or emergency power source is inoperable.

Basis

At the conditions of the RCS temperature (T_{avg}) greater than 350°F or the reactor critical, the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs at the conditions noted above. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORVs to perform manual RCS pressure control for mitigation of SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function at the conditions noted above. The automatic pressure control function limits the number of challenges to the safety valves, while the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Operation with the block valves open is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed to isolate PORV leakage is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The ability to operate with the block valve(s) closed with power maintained to the block valve(s) is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage so that maintenance can be performed on the PORVs to eliminate the leakage condition. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor coolant system pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has leakage resulting in excessive RCS leakage. (Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.) The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to exceeding cold shutdown following the associated refueling outage.

The OPERABILITY of the PORVs and block valves at the conditions noted above is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,

2. Manual control of PORVs to control RCS pressure as required for SGTR mitigation,

3. Manual closing of a block valve to isolate a stuck open PORV,

4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and

5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for SGTR mitigation.

A PORV is defined as leaking with up to and including one (1) gpm of seat leakage, but is not inoperable and is not experiencing "excessive" seat leakage as identified within Specification 3.1.1.5.a. With leakage up to and including ten (10) gpm, the PORV would be considered to have "excessive" seat leakage and would be subject to the compensatory actions described within Specification 3.1.1.5.a. This condition would continue to require block valve testing on a 92 day interval as required by Surveillance Requirement 4.2.4.2. Finally, with PORV leakage exceeding ten (10) gpm, the PORV is considered inoperable in accordance with Specifications 3.1.1.5.b. and c., and block valve testing is not required.

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 (for vessel exposure up to 24 EFPY). These limitations are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be OPERABLE[1], with both power operated relief valves OPERABLE with a lift setting of less than or equal to 420 psi whenever any RCS

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1 The overpressure protection system shall not be considered inoperable solely because either the normal or emergency power source for the PORV block valves is inoperable.

cold leg temperature is less than or equal to 350°F and when the head is on the reactor vessel and the RCS is not vented to the containment.

1. With one PORV inoperable and T_{avg} greater than 200°F and any RCS cold leg temperature less than 350°F:

   A. Restore the inoperable PORV to OPERABLE status within 7 days; or

   B. Depressurize and vent the RCS to the CV within the next 12 hours.

2. With one PORV inoperable and T_{avg} less than or equal to 200°F:

   A. Restore the inoperable PORV to OPERABLE status within 24 hours; or

   B. Complete depressurization and venting of the RCS to the CV within an additional 12 hours.

3. With both PORVs inoperable, complete depressurization and venting of the RCS to the CV within 12 hours.

4. With the RCS vented per 1, 2, or 3, verify the vent pathway:

   A. At least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; or

   B. At least once per shift.

5. In the event the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.3 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

6. For this specification, reactor startup, heatup and entry into operational conditions with T_{avg} greater than or equal to 350°F may continue so long as the limits of the associated action statements are met.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 120°F.

3.1.2.3 The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

3.1.2.4 Figures 3.1-1 and 3.1-2 shall be updated periodically in accordance with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposures for which the figures apply.

a. At least 60 days before the end of the integrated power period for which Figures 3.1-1 and 3.1-2 apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and in accordance with applicable appendices of 10CFR50. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

b. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

Basis

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility and impact toughness under certain conditions of irradiation. Accompanying a decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperature 60°F less than the 50 ft-lb (and 35 mils laterial expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of Section III of the ASME Boiler and Pressure Vessel Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The Certified Material Test Reports (CMTRs) for the original steam generators provided records of Charpy V-notch tests performed at +10°F. Acceptable Charpy V-notch tests of +10°F indicate RT_{NDT} is at or below this temperature. The steam generator lower assemblies were replaced in 1984 and the material tests results indicate the highest RT_{NDT} is 60°F or below. The ASME code recommends that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60°F, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure. The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program[1] where a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. These data are compared to data from pertinent radiation effects studies and an increase in the Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original ΔRT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The overpressure protection system consists of two operable pressurizer power-operated relief valves (PORVs) connected to the station instrument air system, a backup nitrogen supply, and the associated electronics.

The TS requirements for low-temperature overpressure protection (LTOP) apply when T_{avg} is less than 350°F and the RCS is not vented to the containment. During these conditions, one train (or channel) of the LTOP system is capable of mitigating an LTOP event that is bounded by the largest mass addition to the RCS or by the largest increase in RCS temperature that can occur. The largest mass addition to the RCS is limited based upon the assumption that no more than a fixed number of pumps are capable of providing makeup or injection into the RCS. Hence, this is a matter important to safety that pumps in excess of this design basis assumption for LTOP not be capable of providing makeup or injection to the

RCS. In this regard the SI Pump breakers are required to be racked out at less than 350°F RCS temperature.

Pages 3.1-8 through 3.1-10 deleted.

References

1. S. E. Yanichko, "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy Systems - WCAP-7373 (January 1970).

2. E. B. Norris, "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit No. 2, Analysis of Capsule V," Southwest Research Institute - Final Report SWRI Project No. 02-4397.

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limits shall be less than or equal to:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is

Class 2 and Class 3 components were chosen based on Regulatory Guide 1.26 and ANSI N18.2 and N18.2a "Nuclear Safety Criteria for the Design of stationary Pressurized Water Reactor Plants."

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes for evidence of mechanical damage or progressive degradation. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Wastage-type defects will be minimized with proper chemistry treatment of the secondary coolant. If defects or significant degradations should develop in service, this condition is expected to be detected during inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit. Steam generator tube inspections by means of eddy current testing have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications.

4.2.2 Materials Irradiation Surveillance Specimens

The reactor vessel material surveillance specimens shall be removed and examined to determine changes in their material properties, as required by Appendix H to 10CFR50.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling after each ten year inspection. At the fourth refueling after each ten year inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods.

4.2.4 Relief Valves

4.2.4.1 In addition to the requirements of Specification 4.0.1, each PORV shall be demonstrated OPERABLE at each refueling by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the PORV through one complete cycle of full travel while T_{avg} is greater than 350°F and the reactor is subcritical.

c. Operating the solenoid air control valves and check valves for their associated accumulators in PORV control systems through one complete cycle of full travel or function testing of individual components.

4.2.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of Specification 3.1.1.5.b. or c.

4.2.4.3 The accumulator for the PORVs shall be demonstrated OPERABLE at each refueling by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.

4.2.5 Low-Temperature Overpressure Protection

4.2.5.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION at each refueling shutdown; and

c. Verifying the PORV block valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

Basis

The OPERABILITY of two PORVs for low-temperature overpressure protection (LTOP) or an RCS vent ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of three charging pumps with injection into a water-solid RCS.

The maximum allowed PORV setpoint for the LTOP system is derived by analyses which model the performance of the LTOP assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure over-shoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require the power supply breakers of all three safety injection pumps be racked out while in hot shutdown and below 350°F with the reactor vessel head installed and the RCS is not vented to containment and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOP will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50 Appendix H.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their required functions. Specification 4.2.4.1 addresses PORVs, 4.2.4.2 the block valves, and 4.2.4.3 the independent pneumatic power source. Specification 4.2.5.1 addresses the PORV overpressure protection functions and 4.2.5.2 addresses RCS vent pathways.

Surveillance Requirement 4.2.4.1.a. provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for the PORV to be OPERABLE in the power operating and hot shutdown conditions greater than 350°F.

Surveillance Requirement 4.2.4.1.b. provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS. This test will be done at hot shutdown with T_{avg} greater than 350°F before the PORV is required for overpressure protection in Technical Specification 3.1.2.1.d.

Surveillance Requirement 4.2.4.1.c. provides assurance that the mechanical and electrical aspects of the control system are functional.

Surveillance Requirement 4.2.4.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with Technical Specification 3.1.1.5.b. or c. This precludes the need to cycle the valves with a full system differential pressure or when maintenance is being performed to restore an inoperable PORV to OPERABLE status. Also, this limits the challenges to the primary function of the block valve which is to provide an RCS pressure boundary for a degraded PORV.

Surveillance Requirement 4.2.4.3 provides assurance of operability of the accumulators and that the accumulators are capable of supplying sufficient Nitrogen to operate the PORV(s) if they are needed for RCS pressure control and normal Nitrogen and the backup Instrument Air systems are not available. Backup Instrument Air is supplied when the accumulator reaches its low pressure setpoint.

Surveillance Requirement 4.2.5.1 provides assurance that the instrumentation for the actuation of the LTOP function of PORVs is calibrated to provide automatic actuation of the PORVs for low temperature conditions. Also, the flow path to the PORV is assured to be open.

References

(1) FSAR, Section 4.4

(2) FSAR, Volume 4, Tab VII, Question VI.C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

On June 25, 1990, the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." The GL represented the technical resolution of the above-mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in pressurized water reactor (PWR) plants. The GL discussed how PORVs are increasingly being relied upon to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed NRC positions and improvements to the plant's Technical Specifications (TS) were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The GL discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letter dated June 18, 1992, as supplemented December 8, 1992, and February 3, 1995, Carolina Power & Light Company (the licensee) requested changes to the TS for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR). The requested changes would add limiting conditions for operation (LCOs) and surveillance requirements for the pressurizer PORVs and their associated block valves whenever average temperature is above 350 degrees F or the reactor is critical. Proposed TS will also be added for low-temperature overpressure protection (LTOP) whenever average temperature is less than 350 degrees F and the reactor coolant system (RCS) is not vented to the containment. The December 8, 1992, letter corrected a typographical error and did not affect the no significant hazards consideration determination. The licensee's letter dated February 3, 1995, proposed a revision to the TS regarding block valve testing in accordance with the GL recommendations.

2.0 EVALUATION

2.1 Generic Issue 70

The actions proposed by the NRC to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

With the exception of the following variations, the proposed changes to the HBR TS are consistent with the guidance of GL 90-06.

2.1.1 Hot Standby

The licensee proposed specification wording in a format consistent with the HBR operating condition definitions for the range of conditions as defined in the Standard Technical Specifications (STS), because the HBR specifications do not define a hot standby condition. The STS defines hot standby as a reactivity less than 0.99, 0 percent rated thermal power and Tavg greater than or equal to 350 degrees F. The STS definition for hot shutdown is the same as for hot standby, except that Tavg is less than 350 degrees F and greater than 200 degrees F. In the HBR TS, hot shutdown is defined as when the reactor is subcritical and Tavg is greater than 200 degrees F. Based on the HBR definitions, the STS hot standby condition would be represented by the reactor being subcritical and Tavg greater than or equal to 350 degrees F. The STS hot shutdown condition, as represented by the HBR definitions, would be that the reactor is subcritical with a Tavg less than 350 degrees F. The HBR plant/reactor conditions are the same conditions as represented by the STS, and are, thus, acceptable to the NRC staff.

2.1.2 PORV Block Valve Inoperability

The licensee has deviated from the GL with respect to PORV block valve inoperability due to normal or emergency power source inoperability. Because the PORVs and block valves were not originally designed as safety-related components at HBR, the power for both PORV block valves is supplied from the same power source, emergency bus E-2. The 4-kV bus 3 powers emergency bus E-2, and its emergency power is from the "B" emergency diesel generator (EDG). The existing TS 3.7.2.d allows power operation to continue for up to 7 days while the "B" EDG is inoperable, to accommodate corrective or preventive maintenance on the EDG. The existing TS 3.7.2.a, b and c also provide LCO for the loss of all normal power sources which vary from 24 hours to indefinite operation, depending on the nature of the loss of the power source. If the GL model TS 3.4.4.d were applied at HBR and both block valves were determined inoperable by TS 1.3 due to an inoperable common normal or emergency power source, the 7-day (or longer) LCO would be effectively shortened to 72 hours. The NRC staff agrees that an exception for loss of normal or emergency power in this case (footnote 1 to TS 3.1.1.5) eliminates an unnecessary thermal cycle on the plant, and, therefore, finds the change acceptable.

2.1.3 PORV Through-Leakage

The current HBR TS do not provide an acceptable value for leakage from a PORV. However, actions to be taken can be related to TS 3.1.5.2 for RCS leakage. A block valve can be closed to assist in the identification of a leak location, and, thereby, PORV seat leakage would be identified. Continued operation would be safe because the block valve could be closed to isolate the leak, and continued operation with leakage less than or equal to 10 gallons per minute (gpm) would be permitted by TS 3.1.5.2. For any nonisolable leakage exceeding 10 gpm, TS 3.1.5.2 requires the plant to go to hot shutdown within 12 hours using normal operating procedures. To address RCS leakage through a PORV consistently with other nonisolable RCS leakage, the new specification 3.1.1.5.a should also provide 12 hours to achieve hot shutdown. For consistency relative to PORV inoperability, TS 3.1.1.5.b and c should also allow 12 hours to reach hot shutdown. The NRC agrees with this change, because it meets the intent of the GL.

2.1.4 Isolation of PORVs

The practice at HBR has been to isolate a leaking PORV before reaching the limits of TS 3.1.5.2 to preclude degradation of the valve seat. This practice has shown that the PORV damage from leakage can be limited thereby avoiding the need for major valve rework or replacement. The NRC guidance allows power operation to continue only if the block valves were shut after the excessive leakage threshold had been exceeded. The licensee has included the discretionary isolation of a leaking PORV in TS 3.1.1.5.b and c, consistent with TS 3.1.1.5.a. This change is acceptable as it is consistent with the GL.

2.1.5 Degassing prior to Hot Shutdown

NRC guidance provides 6 hours for achieving hot shutdown subsequent to being in hot standby. For proposed TS 3.1.1.5, the licensee states that HBR needs 12 hours to achieve hot shutdown. Should a PORV or block valve inoperability require the RCS to be opened, the system must be degassed (i.e., hydrogen and other volatile gases must be removed). This gas removal process is performed more efficiently at higher temperatures (Tavg greater than 350 degrees F) and pressures. The degassing process requires more than 12 hours to effectively achieve a hydrogen concentration of less than the 5 cc/kg limit. The 6-hour period specified in the NRC guidance for cooling down to 350 degrees F would relegate a considerable portion of the degassing process to those lower-temperature, lower-pressure conditions and could effectively delay the start of valve maintenance. By completing degassing at higher temperatures, the system is ready for entry more quickly upon shutdown; maintenance can be commenced more quickly, hence, reducing the time required to put the system back in service. This additional time for degassing is acceptable to the NRC staff, because it allows for safer operation.

2.1.6 Entry into an Operational Condition

The STS provide a specification which precludes entry into an operational condition until all LCO are met without reliance on an associated action statement. Exceptions to this specification are allowed as stated in the individual TS. For PORVs, the NRC allows that exception. The standard specification for precluding entry into an operating condition is STS 3.0.4 and is stated as follows:

> Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the LCO are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

Since the HBR TS has no statement comparable to STS 3.0.4, the NRC accepts the wording the licensee has proposed in TS 3.1.1.5.e:

> For this specification, reactor start-up, heatup and entry into operational conditions with Tavg greater than or equal to 350 degrees F may continue so long as the limits of associated action statements are met.

2.1.7 Proposed Surveillance Testing

Proposed TS 3.1.1.5.f would allow performance of certain surveillance testing of the PORVs and block valves without declaring the associated valve train inoperable. Due to the short duration of the surveillance tests performed and the plant conditions under which the tests are performed, the probability of an event occurring during the test is very low. Further, with only one valve train allowed out of service at a time and its redundant train available, overpressure protection remains available during the testing. The NRC finds this acceptable as a plant specific alternative.

2.1.8 Testing of Block Valves

In the initial amendment request dated June 18, 1992, the licensee indicated that testing of the block valve, when isolated for pressure boundary protection control, could challenge the plant protective systems by causing a decrease in system pressure and could exacerbate the excessive leakage. Following a discussion to clarify the NRC position, the licensee submitted a revised amendment request in a letter dated February 3, 1995. The licensee agreed with the GL that assurance of the block valve operability outweighs potential risks associated with isolating a pressurizer PORV with excessive seat leakage during surveillance.

The capability to cycle block valves is needed for several accident mitigation strategies, such as "feed-and-bleed" of the RCS in case of a total loss of main and auxiliary feedwater. The licensee, however, in adopting the NRC's position with respect to testing block valves associated with isolated leaking PORVs has requested that additional wording be added to the TS to characterize the nature of the leakage. A pressurizer PORV would be defined as "leaking" with up to and including 1 gallon per minute (gpm) of seat leakage, but would not be inoperable. "Excessive" leakage is defined by the licensee as greater than 1 gpm up to and including 10 gpm. In the case of excessive seat leakage, as proposed in TS 3.1.1.5.a, power operation may continue with excessive leakage through the pressurizer PORV with the associated block valve closed provided block valve surveillance testing continues on a 92-day interval as proposed in TS 4.2.4.2. With pressurizer PORV leakage exceeding 10 gpm, the pressurizer PORV is considered inoperable and block valve surveillance testing would not be required. The NRC agrees with the licensee's proposed TS changes; the block valve is not required to be cycled when isolating an inoperable PORV.

The NRC has reviewed the licensee's proposed changes to the HBR TS as described above. The proposed changes will ensure that HBR satisfies the intent of GL 90-06, which is to enhance the reliability of PORVs and block valves. Since the proposed modifications are consistent with the NRC's position previously stated in GL 90-06 or found to be acceptably justified, the NRC finds the proposed modifications to be acceptable.

2.2 Generic Issue 94

The actions proposed by the NRC to improve the availability of the LTOP system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

The proposed changes to the HBR TS reduce the allowed out-of-service time for the PORVs from 7 days to 72 hours. More explicit surveillance requirements are also provided. With the exception of the following variations, the proposed changes to the HBR TS are consistent with GL 90-06.

2.2.1 Cooldown Rates

For TS 3.1.2.1, the licensee states that 12 hours are needed to depressurize and vent the RCS versus 8 hours proposed by the modified TS, Section 3.4.9.3 recommended in the GL. In order that the reactor cooldown rate not exceed that allowed by the TS cooldown curves, the general procedure GP-007, "PLANT COOLDOWN FROM HOT SHUTDOWN TO COLD SHUTDOWN," provides the following guidance in a note prior to the steps initiating cooldown:

Do not exceed the cooldown limitations set below:

350°F to 300°F	60°F/hour maximum
300°F to 250°F	30°F/hour maximum
250°F to 200°F	15°F/hour maximum
200°F to 170°F	10°F/hour maximum
less than 170°F	3°F/hour maximum

Using this guidance, a cooldown from 350 degrees F to 200 degrees F requires a minimum of 6 hours. Based on the need to warm up the residual heat removal system to take the plant to cold shutdown and potential for delays in adjusting cooldown rates, the licensee requested an additional 4 hours for this LCO. The NRC has reviewed the request and finds the additional 4 hours is justified and the change is, therefore, acceptable.

2.2.2 Overpressure Protection System Inoperability

A deviation from the GL guidance to consider the overpressure protection system inoperable due to inoperability of the emergency power source has been included in the proposed TS 3.1.2.1. The NRC finds this deviation acceptable and consistent with TS 3.1.1.5, as discussed above in section 2.1.2 relative to Generic Issue 70. The change satisfies the intent of the GL with respect to LTOP, and is, therefore, acceptable to the NRC staff.

The NRC has reviewed the licensee's proposed modifications to the HBR TS. The proposed changes satisfy the intent of GL 90-06, which is to enhance the reliability of PORVs and block valves and provide additional LTOP. The proposed modifications are consistent with GL 90-06, and are otherwise acceptably justified. Therefore, the NRC finds the proposed modifications to the TS to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the Surveillance Requirements. The NRC has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60FR 11127). Accordingly, the amendment meets 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 the amendment.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Gary Hammer
Lambros Lois
Brenda L. Mozafari

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