Docket No. 50-261

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. DPR-23 for the H. B. Robinson Unit 2 facility. The amendment includes Change No. 37 to the Technical Specifications and is in response to your request dated January 24, 1975.

The amendment revises the Technical Specifications pertaining to the (1) heatup and cooldown rates, (2) pressure temperature limits and (3) requirements for reporting results of the irradiation specimen measurement program.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 12
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

SEE PREVIOUS YELLOW FOR CONCURRENCE*

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Docket No. 50-261

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KRG

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Carolina Power & Light Company

JUL 2 2 1975

cc: w/enclosures

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910 17th Street, N. W.
Washington, D. C. 20006

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Raleigh, North Carolina 27603

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29532

Mr. Dave Hopkins
Environmental Protection Agency
Region IV Office
1421 Peachtree Street, N. E.
Atlanta, Georgia 30309

Hartsville Memorial Library
Home and Fifth Avenues
Hartsville, South Carolina 29550

U. S. NUCLEAR REGULATORY COMMISSION

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
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 January 24, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

   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.

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

   "B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 37."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 37 to the
Technical Specifications

Date of Issuance: 7/22/75

ATTACHMENT TO AMENDMENT NO. 12

CHANGE NO. 37 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace pages 3.1-4 through 3.1-12 with the attached revised pages.

Replace Figures 3.1-1 and 3.1-2 with the attached revised Figures.

Delete Figure 3.1-3.

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, and are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed $60^{o}F/hr$ 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 $100^{o}F/hr$ in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 37
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.

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 $70^{o}F$.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed $200^{o}F/hr$. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than $320^{o}F$.

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 exposure 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, Summer 1972 Addenda, Non-Mandatory Appendix G. 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. 37

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 under certain conditions of irradiation. In pressure vessel material, the most serious mechanical property change is the reduction in the upper shelf impact strength. Accompanying the 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 temperatue $60^{\circ}F$ less than the 50 ft-lb (and 35 mils lateral 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 the ASME Code. The K_{IR} curve is a lower bound 37
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 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. The increase in the Charpy V-notch 50 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + RT_{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 Non-Mandatory 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 explicit safety factors of 2.0 and 1.25* on stress intensity factors induced by pressure and thermal gradients, respectively. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\, K_{IM} + 1.25\, K_{It} \leq K_{IR} \qquad (1)$$

37

where:

- K_{IM} is the pressure intensity factor caused by the thermal (pressure) stresses.
- K_{It} is the stress intensity factor caused by the thermal gradients.
- K_{IR} is the reference stress intensity factor provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

---

* The 1.25 safety factor on K_{It} represents additional conservatism above Code Requirements.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4T location is considered. The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4T location becomes the controlling factor. Unlike the situation at the 1/4T locations, at the 3/4T position (i.e., the tip of the 1/4T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceeding paragraph cannot be defined.
Rather, each rate of interest must be analyzed on an individual basis. 37

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 use of the composite curve in the cooldown analysis is necessary because
system control is based on a measurement of reactor coolant temperature, 37
whereas the limiting pressure is calculated using the material temperature at the tip of the assumed 1/4T reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the coolant which is at the indicated temperature. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} which is less limiting at a given indicated temperature for finite cooldown rates than for steady state under certain conditions. Hydrostatic (leak) test temperatures are defined by ASME Code Appendix G. For H. B. Robinson Unit No. 2 which has a reactor vessel shell thickness of 9.3125 inches and a vessel inner radius of 77.75 inches, a hydrostatic test at ~2350 psi produces membrane stresses of 19,619 psi. Since bending and secondary stresses due to thermal gradients are negligible during hydrostatic test conditions, the governing equation becomes:

$$1.5\ K_{Im} < K_{IR}$$

Using the methods of ASME code Appendix G for determining stress intensity factors for the 1/4T assumed flaw, K_{Im} is 58.13 Ksi $\sqrt{in}$ and thus, 1.5 K_{Im} is 87.2 Ksi $\sqrt{in}$. In order for K_{IR} to be 87.2 Ksi $\sqrt{in}$. or greater, a temperature of RT_{NDT} + $109^{o}F$ must be attained as determined from the K_{IR} curve of ASME Code Appendix G. This results in the limit shown on Figure 3.1-1 for the applicable integrated power period.

<u>References:</u> 37

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. S. E. Yanichko etal, "Analysis of Capsule S from Carolina Power & Light Company, H. B. Robinson Unit No. 2, Reactor Vessel Radiation Surveillance Program", Westinghouse Nuclear Energy Systems - FP-RA-2 .(December 18, 1973).

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, above which the moderator temperature coefficient is positive.

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. 37

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be 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 the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.[1] [2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the initial fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the initial fuel cycle and during subsequent reload fuel cycles, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[1] [2] The maximum teperature at which the moderator coefficient is positive, at the beginning of life of the initial fuel cycle, with all control rods withdrawn, will be determined during preoperational physics tests. When control rods are inserted, the temperature at which the moderator coefficient becomes negative is lower so that at the temperature determined during the physics tests with the operational control rod program, the temperature coefficient is expected to be negative. The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50 Appendix G paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods, are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as 37
calculated by ASME Code Appendix G methods, and an additional safety margin of $40^{\circ}F$ must be maintained above the applicable heatup curve at all times.

References:

(1) FSAR Table 3.2.1-1
(2) FSAR Figure 3.2.1-9
(3) FSAR Figure 3.2.1-10

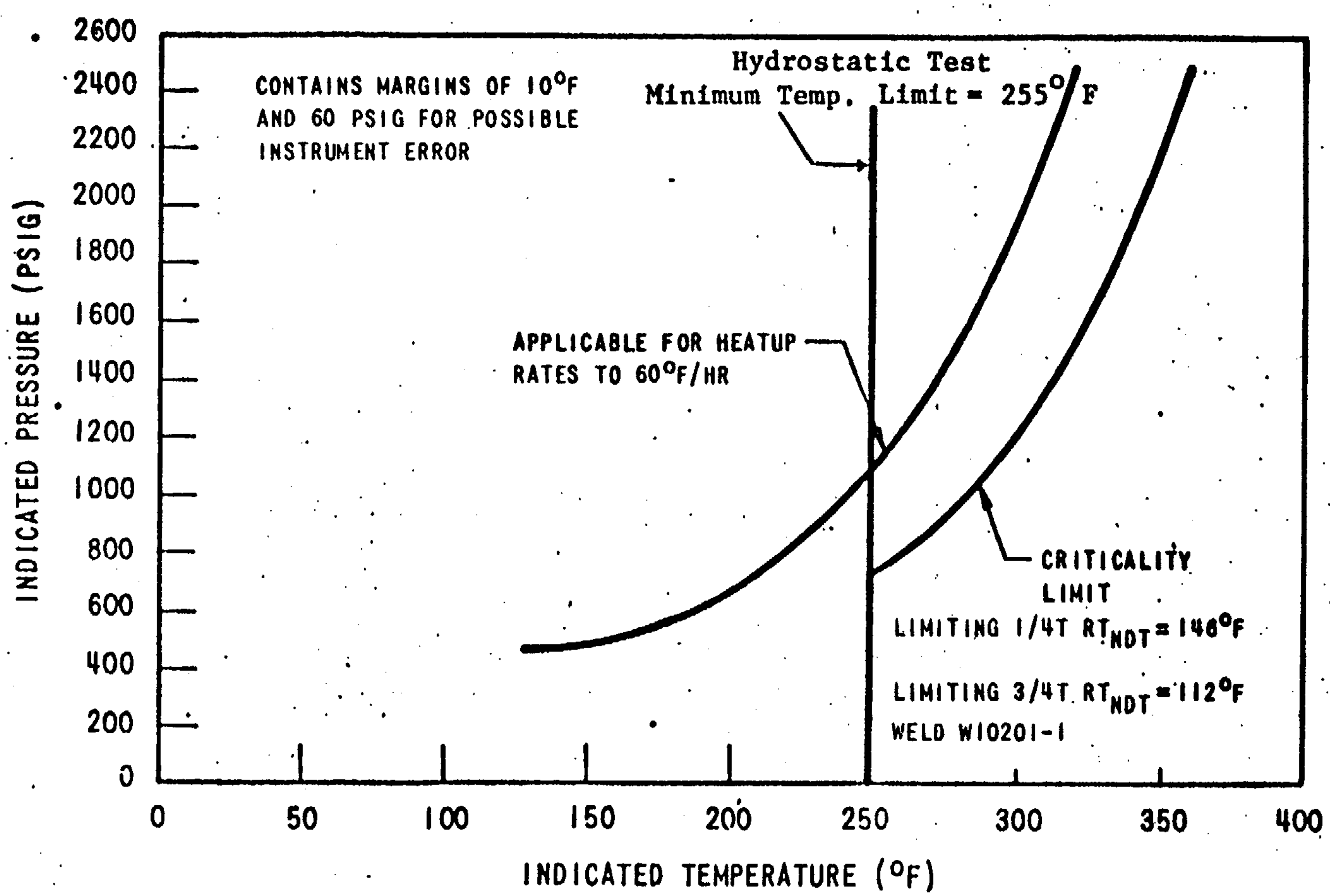
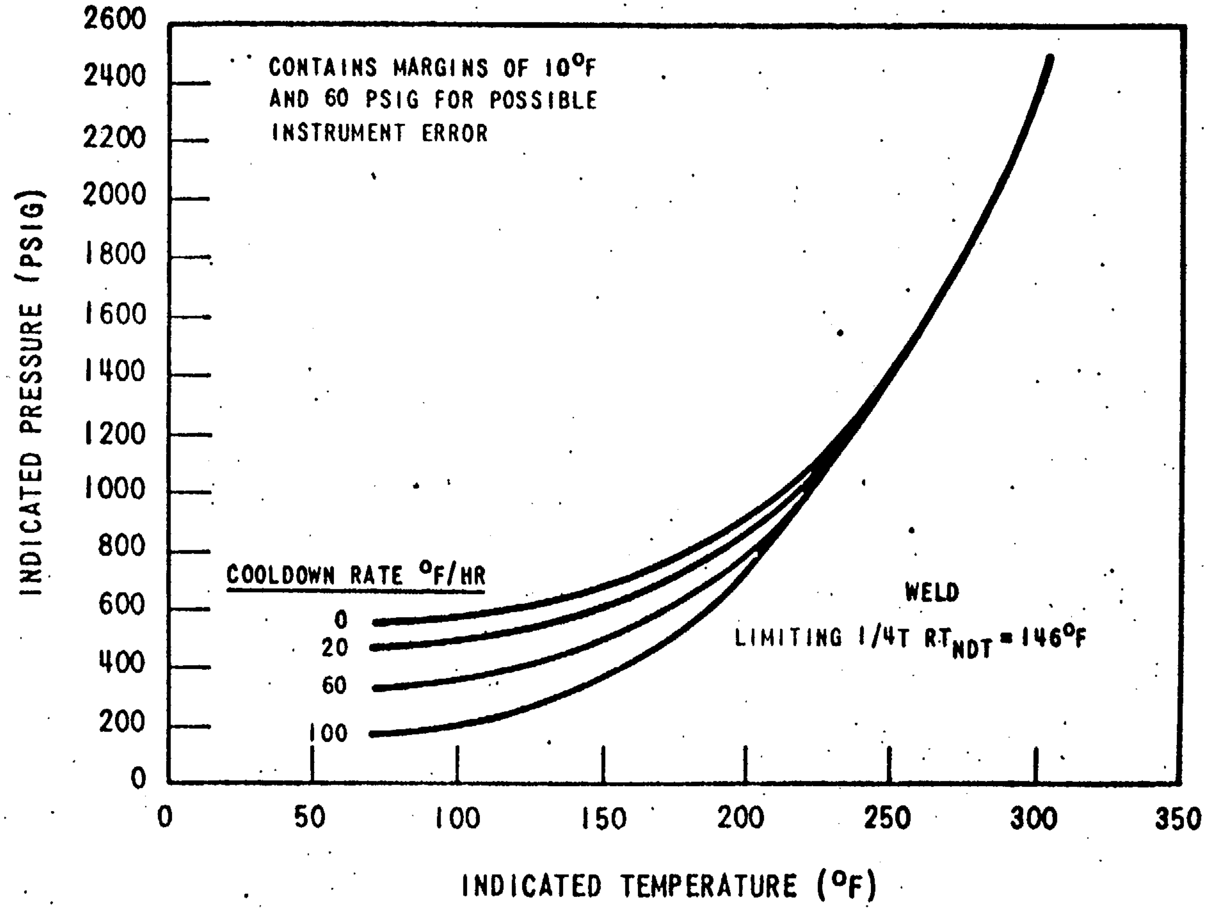


Figure 3.1-1

H. B. Robinson Unit No. 2 Reactor Coolant System Heatup Limitations Applicable For the First 4.25 Effective Full Power Years (Fluence $\sim$ 3.7 x 10 ^{18}N/CM^2)

Figure 3.1-2



H. B. Robinson Unit No. 2 Reactor Coolant System Cooldown Limitations Applicable for the First 4.25 Effective Full Power Years (Fluence $\sim$ 3.7 x 10^{18} N/CM^2)

Figure 3.1-3 Deleted 37

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 TO LICENSE NO DPR-25

(CHANGE NO. 37 TO THE TECHNICAL SPECIFICATIONS)

CAROLINA POWER & LIGHT COMPANY

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

Introduction

By letter dated January 24, 1975, Carolina Power and Light Company (CP&L) requested a change to the Technical Specifications appended to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Unit No. 2 (Robinson-2). These changes involve pressure-temperature operating limits and are based on the requirements of Appendix G to 10 CFR Part 50 and the results of tests on reactor vessel surveillance samples removed during the 1973 refueling outage.

Discussion

In accordance with Appendix H to 10 CFR Part 50 the licensee has an irradiation surveillance program to monitor changes in the fracture toughness properties of reactor vessel materials as a result of neutron irradiation and the thermal enviroment due to plant operation. Eight surveillance capsules to monitor these effects on the Robinson-2 reactor pressure vessel core region material over the life of the vessel were inserted in the reactor vessel prior to initial plant startup. The first of these capsules, Capsule S, was removed during the first refueling shutdown and the samples subsequently tested. The capsule contained specimens for mechanical property testing (tensile tests, Charpy impact tests, and wedge opening loading tests), dosimeters for fast neutron fluence measurements, and thermal monitors to indicate maximum sample temperature. The data from this testing program and the requirements of Appendix G to 10 CFR Part 50 served as the basis for the proposed Technical Specification changes described herein.

Specifically, the proposed Technical Specification changes include the following modifications and limits:

(1) Revised reactor system heatup and cooldown rates and accompanying pressure-temperature limits.

(2) Definition of pressure-temperature limits for reactor system hydrostatic leak tests.

(3) Requirements for updating the heatup and cooldown rates and operating limits for time periods beyond the applicability of the proposed heatup and cooldown rates and pressure-temperature limits.

(4) Requirement for reporting the results of the irradiation specimen measurement program.

(5) Revised pressure-temperature limits for attaining criticality.

The proposed pressure-temperature limits would replace interim operating limits and would be effective through 4.25 effective full power years of operation (Robinson-2 presently has accumulated approximately 3 effective full power years of operation). The service period of 4.25 effective full power years was chosen for the first heatup and cooldown calculations since this operational period corresponds to a neutron fluence of 3.69 x 10^{18} n/cm^2 (the fluence received by Capsule S) at the one-fourth wall thickness depth location in reactor vessel. Since the capsule containing test specimens of vessel metal was located such that it received a higher neutron flux than the reactor vessel wall, test results obtained from the capsule samples can be conservatively used to predict irradiation effects on the reactor vessel material.

The most useful and fundamental data attained from the 1973 test measurements were the experimental determination of the fast neutron fluence exposure in the capsule specimens (3.7 x 10^{18} n/cm^2) and the shift in the reference nil ductility temperature (RT_{NDT}) of the pressure vessel core region shell material (maximum shift of 35°F). The shift in the RT_{NDT} (a direct effect of the neutron irradiation) is an increase in the metal temperature associated with the transition from brittle to ductile fracture mode of metal failure. The practical implication of RT_{NDT} shift is that during the heatup and cooldown process for a given reactor pressure, the required temperature of the vessel metal becomes increasingly larger with reactor exposure. This property is one of the primary factors monitored in the surveillance program.

Evaluation

The proposed pressure-temperature operating limits involve limits for heatup, cooldown, hydrostatic testing, and criticality. Although all of these operating conditions represent a wide range of activity the limit structure is derived using the same basic data and analytical approach. The proposed limits are based primarily on experimental data (and conservative theoretical values when experimental data was not available) and upgraded analytical techniques. Specific comments on each aspect of the proposed Technical Specification changes will be discussed in the following paragraphs.

(1) Revised heatup-cooldown rates and pressure-temperature limits.

Present Technical Specifications limit maximum heatup and cooldown rates to a maximum of 100°F/hr with lower rates imposed for certain temperature ranges. The proposed Technical Specifications limit heatup rates to 60°F/hr and cooldown rates to 100°F/hr. The proposed heatup and cooldown rates and accompanying pressure-temperature limits are based on updated techniques which are spelled out in Summer 1972 Addenda to the ASME Boiler and Pressure Vessel Cold, Section III and presently required now in Appendix G to 10 CFR Part 50. The technique requires the use of the RT_{NDT} which is in turn used to index the material to a reference stress intensity factor curve which appears in Appendix G of the ASME Code. The curve in the Code provides the stress intensity factor for the material (considering radiation effects) and this stress intensity factor serves as an allowable upper limit in the analysis. Actual stress intensity factors are then determined by combining the effects of pressure stress and thermal gradient stress. The allowable stress intensity factor must then be larger than the actual stress intensity factors with appropriate safety margins included. The analysis then consists of determining limiting conditions within these guidelines for various heatup and cooldown rates. Some additional constraints are also required by the Code such as the assumption of certain flaws.

We have reviewed the sample test data, the analytical techniques, and the resulting pressure-temperature curves and conclude that the proposed Technical Specifications meet the requirements of Appendix G to 10 CFR Part 50.

(2) Pressure-temperature limits for hydrostatic leak tests.

The pressure-temperature limits defined for hydrostatic leak tests were determined in a similar manner as the limits discussed above and in accordance with the ASME Code. In this case, however, there was no need for consideration of the thermal gradient stress intensity factor (no metal heatup or cooldown occurs during the hydrostatic test). We have reviewed the proposed limits and conclude that they meet the requirements of Appendix G to 10 CFR Part 50.

(3) Requirements for updating the heatup and cooldown rates and operating limits for time periods beyond the applicability of operating limits in existence.

This proposed Technical Specification change requires that operating limit curves be developed at least 60 days prior to the end of the period for which curves in existence apply. This requirement is in accordance with the requirements of Appendix G to 10 CFR Part 50.

(4) Requirements for reporting the results of the irradiation specimen measurement program.

This proposed Technical Specification change requires that results of the measurement program be reported to the Nuclear Regulatory Commission within 90 days of completion of the test but no later than 60 days prior to expiration of the time period for curves applicable to unit operation. The latter constraint was not part of the original Technical Specification change; it has been discussed and found acceptable to the NRC staff and licensee. We conclude, therefore, that the recommended reporting time is reasonable and timely.

(5) Revised pressure-temperature limits for attaining criticality.

The proposed limits were derived using methods described earlier with additional safety factors required by Appendix G to 10 CFR Part 50. These additional factors include the temperature for criticality (equal to that of hydrostatic test) and an additional 40°F temperature increase beyond the operating pressure-temperature limits discussed earlier. We have reviewed the basic test data, the analytical techniques and the resultant limits and conclude that they meet the requirements of Appendix G to 10 CFR Part 50.

Summary

CP&L has conducted tests on their first capsule irradiation specimens from Robinson-2. Based on the results from the test program, CP&L has proposed revised reactor vessel heatup and cooldown rates and accompanying pressure-temperature limits. We have reviewed the test results, analytical techniques, and proposed limits and conclude that the proposed changes meet the requirements of Appendix G to 10 CFR Part 50.

The operating limit restriction on pressure and temperature discussed above not only meet the requirements of Appendix G to 10 CFR Part 50 but also apply conservatisms, beyond the requirements of Appendix G, to further assure conservative restrictions.

Conclusion

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.

Dated: JUL 2 2 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & 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. 12 to Facility Operating License No. DPR-23 issued to Carolina Power & Light Company which revised Technical Specifications for operation of the H. B. Robinson Unit 2, located in Darlington County, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment permits revisions to the Technical Specifications pertaining to the (1) heatup and cooldown rates, (2) pressure temperature limits, and (3) requirements for reporting results of the irradiation specimen measurement program.

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 this action was published in the FEDERAL REGISTER on April 21, 1975 (40 F.R. 17647). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated January 24, 1975, (2) Amendment No. 12 to License No. DPR-23, with Change No. 37 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, N. W., Washington, D. C., and at the Hartsville Memorial Library, Home & Fifth Avenues, Hartsville, South Carolina.

A 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 Reactor Licensing.

Dated at Bethesda, Maryland, this JUL 2 2 1975

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing