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SUBJECT: Forwards amend to license DPR-23,requesting rev to TS for HB Robinson Steam Electric Plant Unit 2,per generic ltr 90-06.

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R. B. STARKEY, JR.
Vice President
Nuclear Services Department

JUN 18 1992

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United States Nuclear Regulatory Commission
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT
TECHNICAL SPECIFICATIONS REQUIRED BY GENERIC LETTER 90-06

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light Company (CP&L) hereby requests a revision to the Technical Specifications (TS) for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2).

Generic Letter 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 10CFR50.54(f)," requires PWR plants with an operating license to make requests for TS changes of Power-Operated Relief Valve (PORV) and Low-Temperature Overpressure Protection (LTOP) TSs. This letter provides the required specifications. The proposed TSs will add LCO and Surveillance Requirements for the Pressurizer PORVs and their associated Block Valves whenever T_{avg} is above 350 degrees F or the Reactor is critical. Specifications will also be added for the LTOP whenever T_{avg} is less than 350 degrees F and the RCS is not vented to the containment.

Enclosure 1 provides a detailed description of the proposed changes and the basis for the changes.

Enclosure 2 details, in accordance with 10CFR50.91(a), the basis for the Company's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 3 provides an environmental evaluation which demonstrates that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental assessment needs to be prepared in connection with the issuance of the amendment.

Enclosure 4 provides page change instructions for incorporating the proposed revisions.

A019
1/1

Enclosure 5 provides the proposed TS pages.

In accordance with 10CFR50.91(b), CP&L is providing the State of South Carolina with a copy of the proposed license amendment.

In order to allow time for procedure revision and orderly incorporation into copies of the TSs, CP&L requests that the proposed amendments, once approved by the NRC, be issued such that implementation will occur within 60 days of issuance of the amendment.

Please refer any questions regarding this submittal to Mr. R. W. Prunty at (919) 546-7318.

Yours very truly,

R. B. Starkey, Jr.

JSK/jbw

Enclosures:
1. Basis for Change Request
2. 10CFR50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Technical Specification Pages

cc: Mr. S. D. Ebneter
Mr. L. W. Garner
Ms. B. L. Mozafari
Mr. Heyward G. Shealy (SC)
Attorney General (SC)

R. B. Starkey, Jr., having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Eleanor C. Chappell
Notary (Seal)

ELEANOR C. CHAPPELL NOTARY PUBLIC WAKE COUNTY, N.C.

My commission expires: 2/6/96

(1669RNP)

ENCLOSURE 1

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

BASIS FOR CHANGE REQUEST

Background

Generic Letter 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 10CFR50.54(f)," requires PWR plants with an operating license to make requests for TS changes of PORV and LTOP TSs. This letter provides the required specifications. Guidance for these changes was provided in enclosures A and B of the subject Generic Letter.

Proposed Change

The proposed TSs will add LCO and Surveillance Requirements for the Pressurizer Power-Operated Relief Valves (PORVs) and their associated Block Valves whenever T_{avg} is above 350 degrees F or the Reactor is critical. Specifications will also be added for the LTOP whenever T_{avg} is less than 350 degrees F and the RCS is not vented to the containment.

Basis

In the resolution of Generic Issues 70 and 94, as documented in Generic Letter 90-06 (GL 90-06), the NRC requested that licensees, in order to enhance safety, take certain actions identified in the enclosures to GL 90-06. Among these actions was a requirement to modify TSs using the guidance provided.

Traditionally, the PORV and its Block Valve are provided for plant operational flexibility and for limiting the number of challenges to the safety-related pressurizer safety valves. The operation of the PORVs has not been classified as a safety-related function, i.e., one on which the results and conclusions of the safety analysis are based and that invokes the highest level of quality and construction. For overpressure protection of the reactor coolant pressure boundary (RCPB) at normal operating temperature and pressure, the operation of PORVs has not been explicitly considered as a safety-related function. Also, an inadvertent opening of a PORV or Safety Valve has been analyzed in the Updated Final Safety Analysis Reports (UFSARs) as an anticipated operational occurrence with acceptable consequences. For these reasons, most PWRs, particularly those licensed prior to 1979, do not classify PORVs as safety-related components.

The TMI-2 accident focused attention on the reliability of PORVs and Block Valves since the malfunction of the PORV at TMI-2 contributed to the severity of the accident. On other occasions, PORVs have stuck open when called upon to function. Also, there are PORVs in many operating plants that have leakage problems so that the plants must be operated with the Block Valves in the closed position. The TSs governing PORVs on most operating PWRs, which deal with closing the Block Valve and removing power, were developed to allow continued

plant operation with degraded PORVs, but did not consider the need for the PORVs to perform the safety functions discussed below.

Following the TMI-2 accident, the NRC began to examine transient and accident events in more detail, particularly with respect to required operator actions and equipment availability and performance. As a result, the NRC initiated an evaluation of the role of PORVs to perform certain safety-related functions.

It has been determined that over a period of time the role of PORVs and their Block Valves have changed such that PORVs are relied upon by many plants to perform one or more safety-related functions. For HBR2, these safety-related functions are as follows:

1. Maintaining the RCS pressure boundary,
2. Manual control of PORVs to control RCS pressure as required for steam generator tube rupture 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.

Where PORVs are used to perform these functions, or any other safety-related functions that might be identified in the future, it is appropriate to reconsider the safety classification of PORVs and the associated Block Valves if they perform safety-related functions. For operating PWR plants, the NRC concluded that it was not cost effective to replace/backfit existing non-safety-grade PORVs and Block Valves (and associated control systems) with PORVs and Block Valves that are safety grade even when they have been determined to perform any safety-related functions. Subsequent to the TMI-2 accident, a number of improvements were required, which for operating plants the greatest immediate benefit was considered to be implementation of the recommendations of Generic Letter 90-06. These recommendations were considered to increase the reliability of the PORVs, Block Valves and associated components and provide assurance that they would function as required.

Recommended improvements made at Robinson were as follows:

1. Inclusion of the PORVs and Block Valves in the plant operational quality assurance list, even though they remained non-safety-related.
2. Implementation by trained maintenance personnel of a maintenance/refurbishment program for PORVs and Block Valves based on manufacturer's recommendations or guidelines.
3. Procurement of replacement parts and spares for the existing non-safety-related PORVs and Block Valves in accordance with the original construction codes and standards.
4. Inclusion of the PORVs and Block Valves within the HBR2 ASME Section XI IST program.
5. Inclusion of the PORV Block Valves in the MOV Test Program.

The changes being proposed by this request are essentially administrative in nature. The changes apply new restrictions where none previously existed regarding limiting conditions for operation for the PORVs and their Block Valves when T_{avg} is greater than 350 degrees F. The LTOP system changes proposed reduce the previous allowed out of service time for the PORVs from seven days to 72 hours. More explicit Surveillance Requirements are also provided. The ultimate effect of these changes are expected to be more reliable systems with improved availability. There are no system modifications required to implement these changes; therefore, there would not be the possibility of any new or different kinds of accidents from those previously evaluated. The enhanced availability and reliability would tend to decrease the probability of any accidents or consequences of any accidents previously evaluated. The improved availability due to the reduced allowed out of service times, the enhanced reliability due to improved/additional surveillance requirements and programs, and the enhanced attention to the safety-related aspects of PORV and associated Block Valve operations would tend to enhance the margin of safety.

In preparing these specifications, some deviations from the guidance provided by the NRC were required to convert the Standard TS guidance to an HBR2 custom specification format and accommodate HBR2 plant design features:

I. Since the HBR2 specifications do not define a HOT STANDBY condition, the specification wording is proposed in a format consistent with the HBR2 operating condition definitions for the range of conditions as defined in the Standard TSs. The Standard TSs define HOT STANDBY as a K_{eff} less than 0.99, 0 percent rated thermal power and T_{avg} greater than 350 degrees F. The Standard TSs' definition for HOT SHUTDOWN is the same as for HOT STANDBY, except that T_{avg} is less than 350 degrees F and greater than 200 degrees F. For the HBR2 custom specifications, HOT SHUTDOWN is defined as when the reactor is subcritical and T_{avg} is greater than 200 degrees F. Based on the HBR2 definitions, the NRC's HOT STANDBY Condition would be represented by the reactor being subcritical and T_{avg} greater than or equal to 350 degrees F. The NRC's HOT SHUTDOWN condition as represented by the HBR2 definitions would be that the reactor is subcritical with a T_{avg} less than 350 degrees F. The plant/reactor conditions noted for HBR2 are the same conditions as represented by the NRC.

II. Deviation from the guidance is also required in the area of PORV block valve inoperability, specifically with respect to inoperability as a result of the normal or emergency power source being inoperable (see the definition of "operable" in Section 1.3 of TSs). Because the PORVs and block valves were not originally designed as safety-related components at HBR2, the power for both PORV Block Valves is supplied from the same power source, Emergency Bus E-2. E-2 is provided normal power from 4 KV Bus 3 and with emergency power from "B" Emergency Diesel Generator. The existing TSs allow power operation to continue for up to seven days while "B" Emergency Diesel Generator is inoperable to accommodate corrective or preventive maintenance on the diesel, (see TS 3.7.2.d). The existing TSs also provide LCOs 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. (See TS 3.7.2.a, b, and c). If the Generic Letter model TS 3.4.4.d was applied at HBR2 and both Block Valves were determined not to meet the definition of operability (TS 1.3) due to

their common normal or emergency power source being inoperable, the seven-day (or longer) LCO would be effectively shortened to 72 hours. Since the stated intent of this NRC initiative is not to require any backfit, a deviation to exempt inoperability due to the normal or emergency power source being inoperable is clearly warranted. Providing an exception in this case (footnote 1 to TS 3.1.1.5) for loss of normal or emergency power would eliminate an unnecessarily restrictive thermal cycle on the plant. This deviation is considered acceptable for the following reasons: the normal or emergency power source for both valves would be operable; the block valves would remain in the "as is" position on a loss of all power; automatic operation of the PORVs could be maintained.

III. Deviation is also warranted with regard to actions to be taken to eliminate leakage through a PORV. Since the HBR2 TSs do not provide an acceptable value for leakage from the PORV, it is more appropriate to relate actions to be taken to the specification for RCS leakage contained in Section 3.1.5.2. Since a block valve can be closed to assist in the identification of a leak location, PORV seat leakage by definition would be identified. Continued operation would be safe since the block valve could be closed to isolate the leak, therefore continued operation with leakage ≤ 10 gpm would be permitted by TS 3.1.5.2. For any nonisolable leakage situation exceeding 10 gpm, TS 3.1.5.2 requires shutdown to hot shutdown within 12 hours using normal operating procedures. To provide for treatment of RCS leakage through a PORV consistent with other nonisolable RCS leakage, the new specification 3.1.1.5.a should also provide 12 hours to achieve hot shutdown. To provide for consistent treatment of PORV inoperability, specifications T.S. 3.1.1.5.b. and c. should also allow 12 hours to hot shutdown.

IV. It has been the experience at HBR2 that when minor leakage begins to occur at the PORV seat, the continuation of this leakage tends to erode away or cut the valve seat area even more over time. For this reason, HBR2 practice has been to isolate a leaking PORV before reaching the limits of TS 3.1.5.2. This practice has shown that the PORV damage from leakage can be limited thereby precluding the need for major valve rework or replacement at the appropriate time. As published, the guidance would allow power operation to continue only if the block valves were shut after the excessive leakage threshold had been exceeded. Clearly, discretionary isolation of a leaking PORV at the earliest indication of leakage can save dollars and radiological dose due to rework/replacement and should be allowed under 3.1.1.5.a without incurring the shutdown requirement of 3.1.1.5.b or 3.1.1.5.c. For this reason, HBR2 has added this flexibility to specification 3.1.1.5.b and c. Also, a deviation from the requirement to consider the overpressure protection system inoperable solely due to inoperability of the emergency power source has been included in the proposed Section 3.1.2.1 for completeness and is analogous to the deviation in Section 3.1.1.5.

V. The time frame NRC guidance provides is 6 hours for achieving HOT SHUTDOWN subsequent to being in HOT STANDBY. For proposed specification 3.1.1.5, HBR2 needs 12 hours to achieve that condition. Should a PORV or Block Valve inoperability require the RCS to be opened, the system would be required to be degassed, i.e.. hydrogen and other volatile gases be

removed. This gas removal process is more efficient when performed at higher temperatures and pressures. It is generally accepted at HBR2 that performing degassing above a T_{avg} of 350 degrees F will reduce the overall outage time of the equipment. 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 on shutdown; maintenance is allowed to thereby commence more quickly, hence reducing the time required to put the system back in service.

VI. For specification 3.1.2.1, HBR2 needs 12 hours to depressurize and vent the RCS versus 8 hours proposed by the modified TSs, Section 3.4.9.3. In order that the reactor cooldown rate not exceed that allowed by the TS cooldown curves, 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, it can be seen that at the maximum cooldown rate allowed and assuming no hold points due to problems, a cool down from 350°F to 200°F requires a minimum of 6 hours. Based on the need to warm up the RHR system to take the plant to cold shutdown and potential for delays in adjusting cooldown rates, an additional 4 hours is being requested for this LCO.

VII. The Standard TSs provide a specification which precludes entry into an operational condition until all limiting conditions for operation are met without reliance on an associated action statement. Exceptions to this specification are allowed as stated in the individual specifications. 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 Limiting Condition for Operation 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 HBR2 has no statement comparable to 3.0.4, the following statement is proposed in TS 3.1.1.5.e:

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

VIII. Additionally, proposed HBR2 specification 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 the short duration of the surveillance tests performed and the plant conditions under which the tests are performed, the probability of an event occurring is minimal. 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.

IX. Finally, under the surveillance requirement of proposed HBR2 TS 4.2.4.2, testing of the Block Valves should be avoided when they are closed to isolate a PORV with excessive seat leakage, even though the Block Valves may be used to manually mitigate an overpressure event. As stated by the NRC, maintenance of the reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event. 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 excess leakage concern. For this reason the Block Valve should not be cycled when isolating a PORV with excess seat leakage.

Conclusions

The changes being proposed by this request are essentially administrative in nature. The changes apply new restrictions where none previously existed regarding limiting conditions for operation for the PORVs and their Block Valves when T_{avg} is greater than 350 degrees F. The LTOP system changes proposed reduce the previous allowed out of service time for the PORVs from seven days to 72 hours. More explicit Surveillance Requirements are also provided. The ultimate effect of these changes are expected to be more reliable systems with improved availability. There are no system modifications required to implement these changes; therefore, there would not be the possibility of any new or different kinds of accidents from those previously evaluated. The enhanced availability and reliability would tend to decrease the probability of any accidents or consequences of any accidents previously evaluated. The improved availability due to the reduced allowed out of service times, the enhanced reliability due to improved/additional surveillance requirements and programs, and the enhanced attention to the safety-related aspects of PORV and associated Block Valve operations would tend to enhance the margin of safety.

ENCLOSURE 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

10CFR50.92 EVALUATION

The Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination. The bases for this determination are as follows:

Proposed Change

The proposed Technical Specifications (TSs) will add Limiting Conditions for Operation (LCO) and Surveillance Requirements for the Pressurizer Power-Operated Relief Valves (PORVs) and their associated Block Valves whenever T_{avg} is above 350 degrees F or the Reactor is critical. Specifications will also be added for the Low-Temperature Overpressure Protection System (LTOP) whenever T_{avg} is less than 350 degrees F and the RCS is not vented to the containment.

Basis

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. No equipment modifications are required for implementation of these changes. The proposed TSs increase the availability and reliability of the PORVs and Block Valves for their intended function. This enhanced availability and reliability would not create a significant increase in the probability or consequences of accidents previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. As noted above, the requested changes do not involve any physical changes to the plant. The proposed changes would tend to increase the availability and reliability of the PORVs and Block Valves. With no physical changes being made to the PORV and Block Valve equipment and enhanced surveillance and maintenance requirements being employed, the proposed amendment would not create the possibility of a new or different accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The proposed amendment will result in improved availability due to reduced allowed out of service times, enhanced reliability due to improved/additional surveillance requirements and programs, and enhanced attention to the safety-related aspects of PORV and associated Block Valve operations which would not involve a significant reduction in the margin of safety.

ENCLOSURE 3

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

ENVIRONMENTAL CONSIDERATIONS

10CFR51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site; (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

The proposed TSs will add LCO and Surveillance Requirements for the Pressurizer Power-Operated Relief Valves (PORVs) and their associated Block Valves whenever T_{avg} is above 350 degrees F or the Reactor is critical. Specifications will also be added for the Low-Temperature Overpressure Protection System (LTOP) whenever T_{avg} is less than 350 degrees F and the RCS is not vented to the containment.

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9) for the following reasons:

1. As demonstrated in Enclosure 2, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site. The changes being proposed by this request are essentially administrative in nature. The changes apply new restrictions where none previously existed regarding limiting conditions for operation for the PORVs and their Block Valves when T_{avg} is greater than 350 degrees F. The LTOP system changes proposed reduce the previous allowed out of service time for the PORVs from seven days to 72 hours. More explicit Surveillance Requirements are also provided. The ultimate effect of these changes are expected to be more reliable systems with improved availability. There are no system modifications required to implement these changes. Therefore, the amendment will not result in a significant change in the types or

significant increase in the amounts of any effluents that may be released off-site.

3. The proposed amendment does not result in an increase in individual or cumulative occupational radiation exposure since the changes are essentially administrative in nature as discussed in item 2. above.

ENCLOSURE 4

H.B.ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

PAGE CHANGE INSTRUCTIONS

Removed Page	Inserted Page
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-	3.1-3d
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-	3.1-3f
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-	3.1-3h
3.1-4	3.1-4
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3.1-10	-
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ENCLOSURE 5

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

TECHNICAL SPECIFICATION PAGES

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.

INSERT A (NEXT PAGE)
3.1.1.5 RELIEF VALVES

INSERT A

3.1.1.5 Relief Valves

Whenever T_{avg} is above 350 degrees 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 degrees F within the following 12 hours[2].

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 degrees F within the following 12 hours.

---

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.

INSERT A - Con't

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 hour 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 degrees 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 degrees 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 degrees 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.

---

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

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Basis

At the conditions of the RCS temperature (T_{avg}) greater than 350 degrees 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 a 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.

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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.

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 3.1-1a and Figure 3.1-2a (for vessel exposure up to 12.5 EFPY) or Figure 3.1-1b and Figure 3.1-2b (for vessel exposure up to 15 EFPY). The 15 EFPY curves may be used for operation prior to the end of 12.5 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 3.1-2a or 3.1-2b (as appropriate). This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2a or Figure 3.1-2b may be obtained by interpolation.

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

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d. ~~The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:~~

~~1. Cooldown and depressurize the RCS or~~

INSERT B

3.1.2.1 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 cold leg temperature is less than or equal to 350 degrees 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 degrees and any RCS cold leg temperature less than 350 degrees:

   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 degrees 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.

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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.

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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 degrees F may continue so long as the limits of the associated action statements are met.

~~2. Heatup the RCS to above 350°F.~~

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

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 3.1-1b and 3.1-2b 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 3.1-1b and 3.1-2b 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

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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 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 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 60F°, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure.

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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

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V-notch 30 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 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 temperture, 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

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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.

(INSERT C - SEE NEXT PAGE)

Pages 3.1-8 through 3.1-10 have been 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.

INSERT C

Basis

The TS requirements for LTOP apply when T_{avg} is less than 350 degrees 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 degrees F RCS temperature.

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Capsule No. 3 leads the vessel maximum exposure by a factor of 2.2 and is scheduled to be removed after twenty years. Thus, sample No. 3 will provide data for an exposure to the vessel of approximately forty years.

Capsule Nos. 4 and 5 lag the maximum vessel exposure by factors of 0.7 and 0.5, respectively. Thus, Capsule No. 4, which is scheduled to be removed after thirty years, provides data for a vessel exposure of twenty-one years and Capsule No. 5, which is scheduled to be removed at forty years, provides data for a vessel exposure of twenty years.

In addition to the capsules discussed above, there are three spares. Two are located at the same location as Capsule No. 5 and one is located at the same location as Capsule No. 4.

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 tear inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods.

INSERT D - SEE NEXT PAGE

References

(1) FSAR, Section 4.4
(2) FSAR, Volume 4, Tab VII, Question VI.C

INSERT D

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 degrees 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.a., 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.

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Basis

The OPERABILITY of two PORVs for low temperature overpressure protection 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 Low Temperature Overpressure Protection system (LTOP) 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 degrees 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 degrees 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.

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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.a.,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.

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].

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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 hour 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.

---

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 a 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.

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 3.1-1a and Figure 3.1-2a (for vessel exposure up to 12.5 EFPY) or Figure 3.1-1b and Figure 3.1-2b (for vessel exposure up to 15 EFPY). The 15 EFPY curves may be used for operation prior to the end of 12.5 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 3.1-2a or 3.1-2b (as appropriate). This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2a or Figure 3.1-2b may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1a or Figure 3.1-1b (as appropriate) 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.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

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 3.1-1b and 3.1-2b 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 3.1-1b and 3.1-2b 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 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 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 that 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 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 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 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.

Capsule No. 3 leads the vessel maximum exposure by a factor of 2.2 and is scheduled to be removed after twenty years. Thus, sample No. 3 will provide data for an exposure to the vessel of approximately forty years.

Capsule Nos. 4 and 5 lag the maximum vessel exposure by factors of 0.7 and 0.5, respectively. Thus, Capsule No. 4, which is scheduled to be removed after thirty years, provides data for a vessel exposure of twenty-one years and Capsule No. 5, which is scheduled to be removed at forty years, provides data for a vessel exposure of twenty years.

In addition to the capsules discussed above, there are three spares. Two are located at the same location as Capsule No. 5 and one is located at the same location as Capsule No. 4.

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.a., 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 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 Low Temperature Overpressure Protection system (LTOP) 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.a.,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