Docket file

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U.S. NUCLEAR REGULATORY COMMISSION

Docket No. 50-261

Mr. J. A. Jones
Senior Executive Vice President
Carolina Power and Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

SUBJECT: ORDER FOR MODIFICATION OF LICENSE CONCERNING PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

This letter transmits an Order for Modification of License which revises the Technical Specifications for Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. The change is a result of the information you provided in response to our 10 CFR 50.54(f) letter of February 23, 1980, regarding primary coolant system pressure isolation valves. Based upon our review of your response, as well as other previously docketed information, we have concluded that a WASH-1400 Event V valve configuration exists at your facility and that corrective action as defined in the attached Order is necessary.

Attached to the Order for Modification of License is the Technical Evaluation Report (TER) which supports the Order; and the plant Technical Specifications which will ensure public health and safety over the operating life of your facility. We are aware that there may be editorial corrections to the attached TER. Please note that the Technical Specifications correctly delineate the requirements for your facility.

In addition to Event V valve configurations, we are continuing our efforts to review other configurations located at high pressure/low pressure system boundaries for their potential risk contribution to an intersystem LOCA. Therefore, further activity regarding the broader topic of intersystem LOCA's may be expected in the future.

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OFFICE							
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A copy of the enclosed Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

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Docket file

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

April 20, 1981

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Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

Mr. J. A. Jones
Carolina Power and Light Company

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

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

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29535

State Clearinghouse
Division of Policy Development
116 West Jones Street
Raleigh, North Carolina 27603

Attorney General
Department of Justice
Justice Building
Raleigh, North Carolina 27602

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
Route 5, Box 266-1A
Hartsville, South Carolina 29550

Michael C. Farrar, Chairman
Atomic Safety and Licensing
Appeal Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Richard S. Salzman
Atomic Safety and Licensing
Appeal Board Panel
U. S. Nuclear Regulatory Commission
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Dr. W. Reed Johnson
Atomic Safety and Licensing
Appeal Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of Carolina Power and Light Co. (H. B. Robinson Steam Electric Plant, Unit No. 2)	)))))))	Docket No. 50-261

ORDER FOR MODIFICATION OF LICENSE

I

The Carolina Power and Light company (the licensee) holds Facility Operating License No. DPR-23, which authorizes the licensee to operate the H. B. Robinson Steam Electric Plant Unit No. 2 at power levels not in excess of 2300 megawatts thermal rated power . The license was originally issued on July 31, 1970 and will expire on April 13, 2007. The facility, which is located at the licensee's site in Darlington County, South Carolina, is a pressurized water reactor (PWR) used for the commercial generation of electricity.

II

The Reactor Safety Study (RSS), WASH-1400, identified in a PWR an inter-system loss of coolant accident (LOCA) which is a significant contributor to risk of core melt accidents (Event V). The design examined in the RSS . contained in-series check valves isolating the high pressure Primary Coolant System (PCS) from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the Event V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.

In order to better define the Event V concern, all light water reactor licensees were requested by letter dated February 23, 1980, to provide the following in accordance with 10 CFR 50.54(f):

1. Describe the valve configurations and indicate if an Event V isolation valve configuration exists within the Class I boundary of the high pressure piping connecting PCS piping to low pressure system piping; e.g., (1) two check valves in series, or (2) two check valves in series with a motor operated valve (MOV);
2. If either of the above Event V configurations exist, indicate whether continuous surveillance or periodic tests are being performed on such valves to ensure integrity. Also indicate whether valves have been known, or found, to lack integrity; and
3. If either of the above Event V configurations exist, indicate whether plant procedures should be revised or if plant modifications should be made to increase reliability.

In addition to the above, licensees were asked to perform individual check valve leak testing prior to plant startup after the next scheduled outage.

By letter dated March 14, 1980, the licensee responded to our February letter. Based upon the NRC review of this response as well as the review of previously docketed information for the facility, I have concluded in consonance with the attached Safety Evaluation (Attachment 1) that one or more valve configuration(s) of concern exist at the facility. The attached Technical Evaluation Report (TER) (Attachment 2) provides, in Section 4.0, a tabulation of the subject valves.

The staff's concern has been exacerbated due not only to the large number of plants which have an Event V configuration(s) but also because of recent unsatisfactory operating experience. Specifically, two plants have leak tested check valves with unsatisfactory results. At Davis-Besse, a pressure isolation check valve in the LPIS failed and the ensuing investigation found that valve internals had become disassembled. At the Sequoyah Nuclear Plant, two Residual Heat Removal (RHR) injection check valves and one RHR recirculation check valve failed because valves jammed open against valve over-travel limiters.

It is, therefore, apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important to safety, they should be tested periodically to ensure low probability of gross failure. As a result, I have determined that periodic examination of check valves must be undertaken by the licensee as provided in Section III below to verify that each valve is seated properly and functioning as a pressure isolation device. Such testing will reduce the overall risk of an inter-system LOCA. The testing mandated by this Order may be accomplished by direct volumetric leakage measurement or by other equivalent means capable of demonstrating that leakage limits are not exceeded in accordance with Section 2.2 of the attached TER.

In view of the operating experiences described above and the potential consequences of check valve failure, I have determined that prompt action is necessary to increase the level of assurance that multiple pressure isolation barriers are in place and will remain intact. Therefore, the public health, safety and interest require that this modification of Facility Operating License No. DPR-23 be immediately effective.

III

Accordingly, pursuant to Section 161i of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT EFFECTIVE IMMEDIATELY, Facility Operating License No. DPR-23 is modified by the addition of the following requirements:

1. Implement Technical Specifications (Attachment 3) which require periodic surveillance over the life of the plant and which specify limiting conditions for operation for PCS pressure isolation valves.
2. If check valves have not been (a) individually tested within 12 months preceding the date of the Order, and (b) found to comply with the leakage rate criteria set forth in the Technical Specifications described in Attachment 3, the MOV in each line shall be closed within 30 days of the effective date of this Order and quarterly Inservice Inspection (ISI) MOV cycling ceased until the check valve tests have been satisfactorily accomplished. (Prior to closing the MOV, procedures shall be implemented and operators trained to assure

that the MOV remains closed. Once closed, the MOV shall be tagged closed to further preclude inadvertent valve opening).

3. The MOV shall not be closed as indicated in paragraph 2 above unless a supporting safety evaluation has been prepared. If the MOV is in an emergency core cooling system (ECCS), the safety evaluation shall include a determination as to whether the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50 will continue to be satisfied with the MOV closed. If the MOV is not in an ECCS, the safety evaluation shall include a determination as to whether operation with the MOV closed presents an unreviewed safety question as defined in 10 CFR 50.59(a)(2). If the requirements of 10 CFR 50.46 and Appendix K have not been satisfied, or if an unreviewed safety question exists as defined in 10 CFR 50.59, then the facility shall be shut down within 30 days of the date of this Order and remain shutdown until check valves are satisfactorily tested in accordance with the Technical Specifications set forth in Attachment 3.

4. The records of the check valve tests required by this Order shall be made available for inspection by the NRC's Office of Inspection and Enforcement.

IV

The licensee or any other person who has an interest affected by this Order may request a hearing on this Order within 25 days of its publication in the Federal Register. A request for hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A copy of the request shall also be sent to the Executive Legal Director at the same address, and to G. F. Trowbridge, Esquire; Shaw, Pittman, Potts and Trowbridge; 1800 M Street NW, Washington, DC 20036, attorney for the licensee. If a hearing is requested by a person other than the licensee, that person shall describe, in accordance with 10 CFR 2.714(a)(2), the manner in which his or her interest is affected by this Order. ANY REQUEST FOR A HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

If a hearing is requested by the licensee or other person who has an interest affected by this Order, the Commission will issue an order designating the time and place of any such hearing. If a hearing is held, the issues to be considered at such a hearing shall be:

(a) Whether the licensee should be required to individually leak test check valves in accordance with the Technical Specifications set forth in Attachment 3 to this Order.

(b) Whether the actions required by Paragraphs 2 and 3 of Section III of this Order must be taken if check valves have not been tested within 12 months preceding the date of this order.

Operation of the facility on terms consistent with this Order is not stayed by the pendency of any proceedings on this Order. In the event that a need for further action becomes apparent, either in the course of proceedings on this Order or any other time, the Director will take appropriate action.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

Effective Date: April 20, 1981
Bethesda, Maryland

Attachments:
1. Safety Evaluation Report
2. Technical Evaluation Report
3. Technical Specifications

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

Attachment 1

SAFETY EVALUATION REPORT
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES
(WASH-1400, EVENT V)

1.0 Introduction

The Reactor Safety Study (RSS), WASH-1400, identified in a PWR an intersystem loss of coolant accident (LOCA) which is a significant contributor to risk of core melt accidents (Event V). The design examined in the RSS contained in-series check valves isolating the high pressure Primary Coolant System (PCS) from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the Event V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.

In order to better define the Event V concern, all light water reactor licensees were requested by 10 CFR 50.54(f) letter, dated February 23, 1980, to identify valve configurations of concern and prior valve test results, if any. By letter dated March 14, 1980, the licensee responded to our request and this information was subsequently transmitted to our contractor, the Franklin Research Center, for verification that the licensee had correctly identified the subject valve configurations.

2.0 Evaluation

In order to prepare the Technical Evaluation Report (TER) it was necessary that the contractor verify and evaluate the licensee's response to our February 1980 letter. The NRC acceptance criteria used by Franklin were based on WASH-1400 findings, probabilistic analyses and appropriate Standard Review Plan requirements. With respect to the verification of the licensee's response to our information request, the Franklin evaluation was based on FSAR information, ISI/IST site visit data, and other previously docketed information. The attached Franklin TER correctly identifies the subject valve configurations.

3.0 Conclusion

Based on our review of the Franklin TER, we find that the valve configurations of concern have been correctly identified. Since periodic testing of these PCS pressure isolation valves will reduce the probability of an intersystem LOCA we, therefore, conclude that the requirement to test these valves should be incorporated into the plant's Technical Specifications.

Dated: April 20, 1981

ATTACHMENT 2

TECHNICAL EVALUATION REPORT

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON UNIT 2

NRC DOCKET NO. 50-261

NRC TAC NO. 12934

FRC PROJECT C5257

NRC CONTRACT NO. NRC-03-79-118

FRC TASK 265

Prepared by

Franklin Research Center
The Parkway at Twentieth Street
Philadelphia, PA 19103

Author: P. N. Noell
T. C. Stilwell
FRC Group Leader: P. N. Noell

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: P. J. Polk

October 24, 1980

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Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

1.0 INTRODUCTION

The NRC has determined that certain isolation valve configurations in systems connecting the high-pressure Primary Coolant System (PCS) to lower-pressure systems extending outside containment are potentially significant contributors to an intersystem loss-of-coolant accident (LOCA). Such configurations have been found to represent a significant factor in the risk computed for core melt accidents.

The sequence of events leading to the core melt is initiated by the concurrent failure of two in-series check valves to function as a pressure isolation barrier between the high-pressure PCS and a lower-pressure system extending beyond containment. This failure can cause an overpressurization and rupture of the low-pressure system, resulting in a LOCA that bypasses containment.

The NRC has determined that the probability of failure of these check valves as a pressure isolation barrier can be significantly reduced if the pressure at each valve is continuously monitored, or if each valve is periodically inspected by leakage testing, ultrasonic examination, or radiographic inspection. The NRC has established a program to provide increased assurance that such multiple isolation barriers are in place in all operating Light Water Reactor plants designated by DOR Generic Implementation Activity B-45.

In a generic letter of February 23, 1980, the NRC requested all licensees to identify the following valve configurations which may exist in any of their plant systems communicating with the PCS: 1) two check valves in series or 2) two check valves in series with a motor-operated valve (MOV).

For plants in which valve configurations of concern are found to exist, licensees were further requested to indicate: 1) whether, to ensure integrity of the various pressure isolation check valves, continuous surveillance or periodic testing was currently being conducted, 2) whether any check valves of concern were known to lack integrity, and 3) whether plant procedures should be revised or plant modifications be made to increase reliability.

Franklin Research Center (FRC) was requested by the NRC to provide technical assistance to NRC's B-45 activity by reviewing each licensee's submittal

against criteria provided by the NRC and by verifying the licensee's reported findings from plant system drawings. This report documents FRC's technical review.

2.0 CRITERIA

2.1 Identification Criteria

For a piping system to have a valve configuration of concern, the following five items must be fulfilled:

1) The high-pressure system must be connected to the Primary Coolant System;

2) there must be a high-pressure/low-pressure interface present in the line;

3) this same piping must eventually lead outside containment;

4) the line must have one of the valve configurations shown in Figure 1; and

5) the pipe line must have a diameter greater than 1 inch.

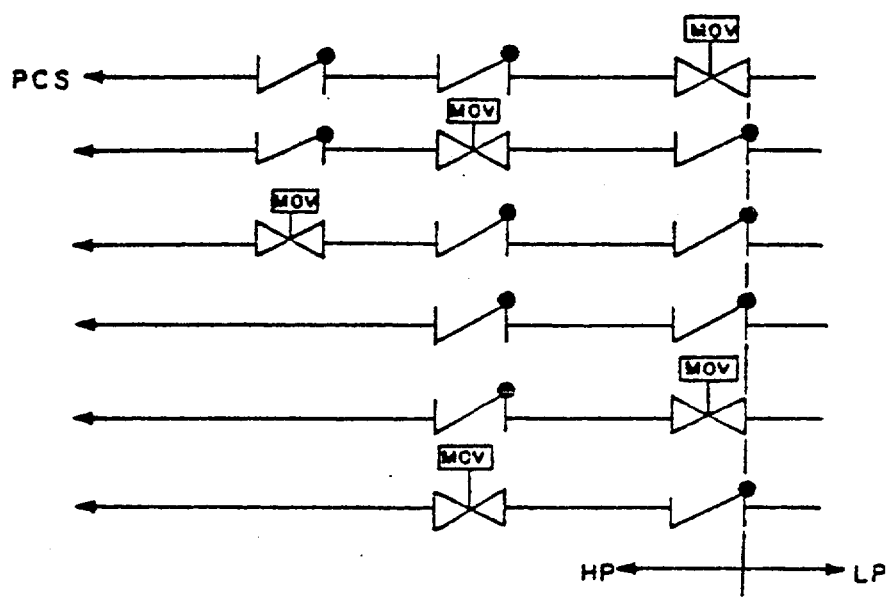

Figure 1. Valve Configurations Designated by the NRC To Be Included in This Technical Evaluation

2.2 Periodic Testing Criteria

For licensees whose plants have valve configurations of concern and choose to institute periodic valve leakage testing, the NRC has established criteria for frequency of testing, test conditions, and acceptable leakage rates. These criteria may be summarized as follows:

2.2.1 Frequency of Testing

Periodic hydrostatic leakage testing* on each check valve shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, each time any check valve may have moved from the fully closed position (i.e., any time the differen- tial pressure across the valve is less than 100 psig), and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

2.2.2 Hydrostatic Pressure Criteria

Leakage tests involving pressure differentials lower than function pressure differentials are permitted in those types of valves in which service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force. Gate valves, check valves, and globe-type valves, having function pressure differential applied over the seat, are examples of valve applications satisfying this requirement. When leakage tests are made in such cases using pressures lower than function maximum pressure differential, the observed leakage shall be adjusted to function maximum pressure differential value. This adjustment shall be made by calculation appropriate to the test media and the ratio between test and function pressure differential, assuming leakage to be directly proportional to the pressure differential to the one-half power.

2.2.3 Acceptable Leakage Rates:

- Leakage rates less than or equal to 1.0 gpm are considered acceptable.
- Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount

*To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

- Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
- Leakage rates greater than 5.0 gpm are considered unacceptable.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Response to the Generic Letter

In response to the NRC's generic letter [Ref. 1], the Carolina Power & Light Company (CPL) stated [Ref. 2] that, "The high-pressure to low-pressure system piping interfaces at H. B. Robinson Unit 2 have been reviewed and three Event V type interfaces (two check valves in series or two check valves in series with a motor-operated valve) have been identified."

The three interfaces identified were:

-High-Pressure Cold-Leg Injection

-Low-Pressure Injection/Residual Heat Removal

-Low-Pressure Injection/Accumulators

The Licensee further stated, "The current surveillance requirements provide for periodic leak testing of at least one check valve in each interface to ensure integrity and functionality as a pressure and/or flow isolation barrier."

Of the three above-mentioned interface systems only the Low-Pressure Injection/Residual Heat Removal System has an Event V-type interface. An additional system, the High-Pressure Hot-Leg Injection System, also contains a valve configuration of concern, as will be described in the following sections.

It is FRC's understanding that, with CPL's concurrence, the NRC will direct CPL to change its Plant Technical Specifications as necessary to ensure that periodic leakage testing (or equivalent testing) is conducted in accordance with the criteria of Section 2.2.

3.2 FRC Review of Licensee's Response

FRC has reviewed the licensee's response against the plant-specific Piping and Instrumentation Diagrams (P&IDs) [Ref. 3] that might have the valve configurations of concern.

FRC has also reviewed the efficacy of instituting periodic testing for the check valves involved in this particular application with respect to the reduction of the probability of an intersystem LOCA in both the Low-Pressure Injection/Residual Heat Removal and High-Pressure Hot-Leg Injection Systems.

In its review of the P&IDs [Ref. 3] for H. B. Robinson Unit 2, FRC found the following two piping systems to be of concern:

> The valve configuration of concern existing in the Low-Pressure Injection/Residual Heat Removal System consists in part of two in-series check valves in each of the cold-leg branches communicating with each of the three reactor coolant loops. Beyond these check valves the three cold-leg branches lead away from the PCS to two motor-operated valves (MOVs) before the two piping trains of this system penetrate the primary containment. The high-pressure/low-pressure interface exists on the upstream side of both MOVs.
>
> In respect to the High-Pressure Hot-Leg Injection System, the valve configuration of concern consists of a single check valve in series with a MOV. This configuration exists in both hot-leg branches leading to Reactor Coolant System Loops 2 and 3. Again, the high-pressure/low-pressure interface exists upstream of each MOV. The appropriate valves for both systems are listed below:

Low-Pressure Injection/Residual Heat Removal

<u>Loop 1, cold leg</u>

high-pressure check valve, 875A

high-pressure check valve, 876A

high-pressure MOV, 744A, normally closed (n.c.)

<u>Loop 2, cold leg</u>

high-pressure check valve, 875B

high-pressure check valve, 876B

high-pressure MOV, 744B, n.c.

Loop 3, cold leg

high-pressure check valve, 875C
high-pressure check valve, 876C
high-pressure MOV, 744A, n.c.

High-Pressure Injection

Loop 2, hot leg

high-pressure check valve, 874B
high-pressure MOV, 866B, n.c.

Loop 3, hot leg

high-pressure check valve, 874A
high-pressure MOV, 866A, n.c.

In accordance with the criteria of Section 2.0, FRC found no other valve configurations of concern existing in this plant.

FRC reviewed the effectiveness of instituting periodic leakage testing of the check valves in these lines as a means of reducing the probability of an intersystem LOCA occurring. FRC found that introducing a program of check valve leakage testing in accordance with the criteria summarized in Section 2.0 will be an effective measure in substantially reducing the probability of an intersystem LOCA occurring in these lines, and a means of increasing the probability that these lines will be able to perform their safety-related functions. It is also a step toward achieving a corresponding reduction in the plant probability of an intersystem LOCA in H. B. Robinson Unit 2.

4.0 CONCLUSION

It has been determined that the Low-Pressure Injection/Residual Heat Removal System in Robinson Unit 2 contains valving in one of the configurations (identified in Figure 1) designated by the NRC as a valve configuration of concern. Moreover, based on the previously docketed information and drawings made available for FRC review, FRC found that the hot-leg branches of the High-

Pressure Injection System also incorporate a valve configuration of concern. Thus, if the licensee's review of the valving configuration contained in the hot-leg branches of the High-Pressure Injection System confirms FRC's finding, then valve configurations of concern exist in two systems of Robinson Unit 2 and incorporate the valves listed in Table 1.0.

If CPL modifies the Plant Technical Specifications for H. B. Robinson Unit 2 to incorporate periodic testing (as delineated in Section 2.2) for the check valves itemized in Table 1.0, then FRC considers this an acceptable means of achieving plant compliance with the NRC staff objectives of Reference 1.

Table 1.0

Primary Coolant System Pressure Isolation Valves

System	Check Valve No.	Allowable Leakage*
Low-Pressure Safety Injection/ Residual Heat Removal		
Loop 1, cold leg	875A 876A	
Loop 2, cold leg	875B 876B	
Loop 3, cold leg	875C 876C	
High-Pressure Injection		
Loop 2, hot leg	874B	
Loop 3, hot leg	874A	

5.0 REFERENCES

1. Generic NRC letter, dated 2/23/80, from Mr. D. G. Eisenhut, Department of Operating Reactors (DOR), to Mr. E. E. Utley, Carolina Power & Light Company (CPL).

*To be provided by licensee at a future date in accordance with Section 2.2.3.

2. Carolina Power & Light Company's response to NRC's letter, dated 3/14/80, from Mr. E. E. Utley (CPL) to Mr. D. G. Eisenhut (DOR).

3. List of examined P&IDs:

FSAR Drawings of H. B. Robinson Unit 2:

541F055, (Rev. 6)
541F056, (Rev. 6)
541F058, (Rev. 7)
684J731, (Rev. 9)
684J753, (Rev. 10)
684J878, (Rev. 10)
684J925, (Rev. 8)
685J433, (Rev. 9)

ATTACHMENT 3

Technical Specification Changes

Remove Pages	Add Pages
3.1-16	3.1-16
-----	3.1-16a
-----	3.1-19a
4.1-10	4.1-10
-----	4.1-10a

3.1.5 Leakage

3.1.5.1 If the primary system leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.1.5.2 If the sources of leakage have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.1.5.3 If the leakage is determined to be primary to secondary steam generator leakage in excess of 0.35 gpm in any steam generator, or in excess of 1 gpm total for all three steam generators, the reactor shall be shutdown and the plant placed in the cold shutdown condition utilizing normal procedures within 30 hours after detection.

3.5.4

a. During reactor operation and hot shutdown conditions, all pressure isolation valves listed in Table 3.1-1 shall be functional as a pressure isolation device, except as specified in 3.1.5.4.b. Valve leakage shall not exceed the amounts indicated.

b. In the event that integrity of any pressure isolation valve specified in Table 3.1-1 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain, in the mode corresponding to the isolated condition. (Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized).

c. If Specifications 3.1.5.4a and b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown within 6 hours and in the cold shutdown condition within the following 30 hours.

Basis

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System, and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage is a conservative limit on what is allowable before the guidelines of 10 CFR Part 20 would be exceeded. This is shown as follows: If the reactor

TABLE 3.1-1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[a] Allowable Leakage
Low-Pressure Safety Injection/Residual Heat Removal		≤ 5.0 GPM for each valve
Loop 1, cold leg	875A 876A	
Loop 2, cold leg	875B 876B	
Loop 3, cold leg	875C 876C	
High-Pressure Injection		
Loop 2, hot leg	874B	
Loop 3, hot leg	874A	

[a] 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

TABLE 4.1-3 (Continued)

	Check	Frequency	Maximum Time Between Tests
13. Turbine Inspection	Visual, Magnaflux and Die Penetrant	Every five years	6 years
14. Fans and Associated Charcoal and Absolute Filters for Control Room and Residual Heat Removal Compartments (HVE-19, HVE-5a and 5b respectively)	Fans functioning, Laboratory tests on charcoal must show ≥ 99% iodine removal. In-place test must show ≥ 99% removal of polydispersed DOP particles by the HEPA filters and Freon by the charcoal filters.	Once per operating cycle	NA
15. Isolation Seal Water System	Functioning	Each refuleing shutdown	NA
16. Overpressure Protection System	Functioning	Each refueling shutdown	NA
17. Primary Coolant System check valves	Functioning	1. Periodic leakage testing[a][b] on each valve listed in Table 3.1-1 shall be accomplished prior to entering reactor operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to service after maintenance. repair or replacement work is performed.	

[a] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

[b] Minimum test differential pressure shall not be less than 150 psid.

TABLE 4.1-3 (Continued)

Check	Frequency	Maximum Time Between Test
	2. Whenever integrity of a pressure isolation valve listed in Table 3.1-1 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.	