DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

LIMITING CONDITION FOR OPERATION

3.6.5.5 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be OPERABLE and closed.

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

ACTION:

a. With a personnel access door or equipment hatch (Other than one pressurizer enclosure hatch) inoperable or open except for personnel transit entry, restore the door or hatch to OPERABLE status or to its closed position (as applicable) within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one pressurizer enclosure hatch open or inoperable, restore the hatch to operable status or to its closed position (as applicable) within 6 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.5.1 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined closed by a visual inspection prior to increasing the Reactor Coolant System T_{avg} above 200°F and after each personnel transit entry when the Reactor Coolant System T_{avg} is above 200°F.

4.6.5.5.2 The personnel access doors and equipment hatches between the containment's upper and lower compartments shall be determined OPERABLE by visually inspecting the seals and sealing surfaces of these penetrations and verifying no detrimental misalignments, cracks or defects in the sealing surfaces, or apparent deterioration of the seal material:

a. Prior to final closure of the penetration each time it has been opened, and

b. At least once per 10 years for penetrations containing seals fabricated from resilient materials.

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ATTACHMENT 2

CATAWBA NUCLEAR STATION
TECHNICAL SPECIFICATION CHANGE REQUEST

PROPOSED TECHNICAL SPECIFICATION AMENDMENT

This proposed amendment to the Technical Specifications (TS) will add a new ACTION statement to TS 3.6.5.5. This ACTION statement will read as follows:

> "With one pressurizer enclosure hatch open or inoperable, restore the hatch to operable status or to its closed position (as applicable) within 6 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours."

DISCUSSION

During operation, situations arise where it is necessary to enter the Pressurizer (PZR) cavity to perform inspections and maintenance. The most recent inspection was performed as a conservative measure in response to the failure of the stem material of a PZR PORV block valve. The reason for this inspection, which required a Regional waiver of compliance, is described below.

Reactor Coolant (NC) System pressure is controlled by use of the PZR which is designed to accommodate positive and negative surges caused by load transients. The PZR provides a point in the NC System where liquid and vapor can be maintained in equilibrium under saturated conditions for pressure control purposes.

The PZR Power Operated Relief Valves (PORVs); NC-32B, NC-34A, and NC-36B, are designed to limit system pressure during a large power mismatch and to prevent actuation of a high pressure reactor trip. The operation of these valves also limits the undesirable lifting of the spring loaded PZR safety valves. The PZR PORVs open automatically when NC pressure exceeds the Process Control System setpoint.

Each PZR PORV block valve (NC-31B, NC-33A, and NC-35B) is used to isolate its associated PZR PORV. Each PZR relief discharge line has an alarm to alert operators when an increase in temperature occurs (representing relief valves lifting or leaking).

On December 9, 1991 it was determined that the stem of 2NC-31B (PORV block valve) was broken. The stem material underwent metallurgical analysis. The analysis indicated that the failed stem material had reduced ductility at room temperature. As a result of this situation, Catawba committed to verify the position of the block valves following valve stroke tests (LER 414/91-016). It was determined that radiography would be the best method available to verify valve position.

TS 3.4.4 requires that the PZR PORV's and their associated block valves be OPERABLE in Modes 1, 2, and 3. Surveillance requirement 4.4.4.2 requires that each block valve be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. After completion of this surveillance, the block valves must be radiographed per the LER 414/91-06 commitment to verify their position. Performance of this radiography requires that the Pressurizer Enclosure Hatch be open longer than the 1 hour allowed by TS.

The effect of having the PZR enclosure hatch open was evaluated and determined to have no significant safety impact. Catawba's letters dated March 18, 1992, and March 24, 1992, requested waiver's of compliance from the current requirement of TS 3.6.5.5. These requests for waivers of compliance were granted, and the PORV block valves were radiographed successfully.

The Unit 1 PORV block valve stem material will be replaced during the End of Cycle 6 refueling outage. Therefore, the Unit 1 valves will not have to be radiographed again during this cycle. The Unit 2 valves will require quarterly testing three more times before the stem material can be changed during the End of Cycle 5 refueling outage. After replacement of the valve stem material, the block valves will not have to be radiographed quarterly following valve strokes.

Entries into the pressurizer cavity are made during startup and shutdown to check for leaks. During operation, if a leak is suspected in the pressurizer cavity, it is necessary to open the pressurizer enclosure hatch to perform an inspection and, if needed, perform repairs. If repairs need to be made, or inspections are time consuming (such as radiography of the PORV block valves) the pressurizer enclosure hatch must be open for longer than 1 hour, which requires a waiver of compliance from TS 3.6.5.5.

Since there is an ongoing need to enter the pressurizer cavity for more than one hour, and removing a pressurizer enclosure hatch does not have a significant affect on safety, Catawba Nuclear Station is requesting that TS 3.6.5.5 be modified as described above.

TECHNICAL JUSTIFICATION

There are five pressurizer enclosure hatches. These hatches are concrete plugs which must be removed with a crane to access the pressurizer cavity. The evaluation for steam bypassing the ice condenser, and the drop analysis were both done for the largest hatch and bound the four smaller hatches.

Two potential concerns involved with the opening of the pressurizer enclosure hatch are:

1) The increase in steam flow which would bypass the ice condenser during a LOCA, and
2) The possibility of dropping the hatch while lifting it.

One of the main concerns with opening the pressurizer hatch is the increase in steam flow which would bypass the ice condenser during a LOCA. Westinghouse analyzed the effects of divider deck leakages for bypass areas of up to 50 ft^2. The results are presented in the FSAR Section 6.2.1.1.3.1, Table 6-12 and Figure 6-18. The results of this analysis show that the pressure peaks are below the design pressure.

The calculation of the new peak compression pressure consists of an extrapolation of Westinghouse results found in the Catawba FSAR, Section 6.2.1.1.3.1 (Loss of Coolant Accident). The compression peak pressure during the blowdown phase of the accident was calculated by Westinghouse to be 7.57 psig, which includes 0.4 psi for the effect of the containment deck bypass area which is assumed to be 5 ft^2.

The effect of the potential deck leakage is expressed by the following equation, which was derived by Westinghouse based on the Waltz Mill test results:

$$\Delta P_{deck} = \text{Bypass Flow Area} \times 0.080 \qquad (1)$$

Substituting the additional area of 3 x 2.5 ft = 7.5 ft^2, resulting from the open pressurizer hatch, in the above equation the following increase in peak pressure is obtained:

$$\Delta P_{deck} = 7.5\ ft^2 \times 0.080 = 0.6\ psi \qquad (2)$$

Hence, the new compression pressure is 8.17 psig, which is well below the acceptance criterion of 14.68 psig.

The open pressurizer hatch will not increase the long term containment peak pressure (14.05 psig), since this would delay melting of the ice, resulting in lower decay heat at the time the ice is depleted. The LOTIC analysis in the FSAR assumes a zero deck leakage for conservatism (faster ice meltout).

The limiting case for containment temperature is a steam line break with the peak occurring in the lower containment. Additional bypass area would result in a lower temperature peak, by directing part of the steam into the upper containment. However, the upper containment temperature is not a concern, since it is 150 to 200 °F below the peak in lower containment (FSAR Figure 6-20 through 6-22). The containment pressure from a steam line break is bounded by the Loss of Coolant Accident.

The removal of the pressurizer hatch for the purpose of performing work would not result in exceeding the containment design pressure should a LOCA occur while the hatch is removed.

The second area of concern is the removal of the pressurizer enclosure hatch during Modes 1-4. Duke Power's letters dated March 15, 1990 and March 21, 1990 requested NRC approval of a revision to a commitment that heavy loads would not be handled in the Catawba reactor building during plant operating Modes 1 (power operation) through 4 (hot shutdown). The conclusions of the drop analysis for the pressurizer enclosure hatch stated that the dropped hatch does not penetrate the operating floor and the structural stability and functions requirements are maintained. It was also stated that there are no safety related components in the drop zones, and that the analyses of the postulated load drops indicate that the intent of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", has been met.

A Safety Evaluation Report was received from the NRC on March 27, 1990, which concluded that the removal of the pressurizer enclosure hatch during Modes 1-4 to perform inspections inside the pressurizer is acceptable.

NO SIGNIFICANT HAZARDS ANALYSIS

10 CFR 50.92 states that a proposed amendment involves no significant hazards consideration if operation in accordance with the 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.

This proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated in the FSAR. Removal of the pressurizer hatch will not cause an increase in the probability of an accident which has been previously evaluated because the pressurizer hatch is not an accident initiator.

The consequences of an accident which has been previously evaluated will not be significantly increased by removal of the pressurizer hatch. As discussed in the preceding analysis, the new compression peak pressure of 8.17 psig is well below the acceptance criteria of 14.68 psig. In addition, the long term containment peak pressure will not be affected due to the delay time in melting of the ice.

The removal of the pressurizer hatch itself has been previously evaluated in modes 1, 2, 3, and 4. In a letter from your staff dated March 27, 1990, a safety evaluation report was issued which concluded that removal of the pressurizer hatch to facilitate inspections inside the pressurizer cavity was appropriate.

The possibility of a missile exiting through the open pressurizer hatch was also evaluated. FSAR Section 3.5.1.2 states that the only credible source of jet propelled missiles within the pressurizer cavity is from the pressurizer RTD wells. The physical location of these RTD wells with respect to the open pressurizer hatch has been reviewed. In the event that these wells became missiles, their location makes it incredible that they would exit the open pressurizer hatch.

This proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. As discussed above, extending the time that the pressurizer hatch is allowed to be open will not create any new or different accidents from those previously evaluated. Removal of the pressurizer hatch to perform inspections inside the pressurizer cavity has been previously evaluated and determined to be acceptable. The preceding analysis provides results which conclude that the containment compression peak pressure, and the long term containment peak pressure, are acceptable with the pressurizer hatch open.

This proposed change will not involve a significant reduction in the margin of safety. As discussed in the preceding analysis, the new compression peak pressure of 8.17 psig is well below the acceptance criteria of 14.68 psig. In addition, the long term containment peak pressure will not be affected due to the delay time in melting of the ice.

ENVIRONMENTAL IMPACT STATEMENT

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22 (c) (9) for categorical exclusion from the requirement for an Environmental Impact Statement.