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May 27, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Nebraska Public Power District – Cooper Nuclear Station
Docket No. 50-298, License No. DPR-46
License Amendment Request – Revised Logic for Reactor Water Cleanup System Isolation, Secondary Containment Isolation, and Control Room Emergency Filter System Initiation

Reference: General Electric Service Information Letter No. 131, Containment Isolation Logic Change, dated March 31, 1975

The purpose of this letter is for the Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 in accordance with the provisions of 10 CFR 50.4 and 10 CFR 50.90 to revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). The proposed amendment lowers the reactor vessel water level at which the Reactor Water Cleanup (RWCU) System isolates, Secondary Containment isolates, and the Control Room Emergency Filter System starts.

General Electric (GE) Service Information Letter (SIL) No. 131, dated, March 31, 1975 (Reference), discussed problems that result from isolation of RWCU and start of Standby Gas Treatment (SGT) System, in conjunction with isolation of secondary containment. The SIL recommended that the vessel water level at which these actions occur be lowered, thereby eliminating these problems and the resulting unnecessary complications with scram recovery. The proposed changes to CNS TS are in accordance with SIL 131 Recommendations 1 and 2.

NPPD requests U.S. Nuclear Regulatory Commission (NRC) approval of the proposed TS change and issuance of the requested license amendment by December 15, 2004. Approval by that date is needed to allow time to finalize the scope of work for CNS Operating Cycle 22 refueling outage, scheduled to start on January 15, 2005. The amendment will be implemented upon startup in Operating Cycle 23.

Attachment 1 provides NPPD's evaluation of the proposed change. Attachment 2 provides the proposed changes to the current CNS TS on marked up pages. Attachment 3 provides the

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revised TS pages in final typed format. Attachment 4 provides the corresponding proposed changes to the current TS Bases on marked up pages for your information.

The proposed TS changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 203, dated March 31, 2004, have been incorporated into this request. This request is submitted under oath pursuant to 10 CFR 50.30(b) using the alternate provisions of NRC Regulatory Issue Summary 2001-18, dated August 22, 2001.

By copy of this letter and its attachments, the appropriate State of Nebraska official is notified in accordance with 10 CFR 50.91(b)(1). Copies to the NRC Region IV office and the CNS Resident Inspector are also being provided in accordance with 10 CFR 50.4(b)(1).

If you have any questions concerning this matter, please contact Mr. Paul Fleming at (402) 825-2774.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 5/27/04
(date)

Randall K. Edington

Randall K. Edington
Vice President – Nuclear and
Chief Nuclear Officer

/rer

Attachments

cc: Regional Administrator w/ attachments
USNRC - Region IV

Senior Project Manager w/ attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/ attachments
USNRC

Nebraska Health and Human Services w/ attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

CNS Records w/ attachments

Attachment 1

Revised Logic for Reactor Water Cleanup System Isolation, Secondary Containment Isolation, and Control Room Emergency Filter System Initiation

Cooper Nuclear Station, NRC Docket 50-298, DPR-46

Revised Pages
Technical Specification Pages
3.3-52
3.3-57
3.3-63

1.0 Description

This letter is a request by the Nebraska Public Power District (NPPD) to amend Operating License No. DPR-46 for Cooper Nuclear Station (CNS.)

The proposed changes to License No. DPR-46 revise the Technical Specifications (TS) by lowering the reactor vessel water level at which three automatic actions occur. The three actions are (1) isolation of the Reactor Water Cleanup (RWCU) System, (2) isolation of Secondary Containment (SC), and (3) initiation of the Control Room Emergency Filter System (CREFS).

Following an automatic reactor shutdown (scram) the water level in the reactor vessel drops due to collapse of voids in the water. This can cause the three actions mentioned above, resulting in unnecessary complications and distractions to the Main Control Room staff during recovery from the scram.

The current reactor vessel water level at which these three automatic actions occur is Level 3, with a TS Allowable Value of ≥ 3 inches. This level is equivalent to 161.19 inches above the top of active fuel (TAF.) It is proposed that this be lowered to Level 2, with a TS Allowable Value of ≥ -42 inches. This level is equivalent to 116.19 inches above TAF. These changes will reduce the potential for these actions to occur in response to a scram.

NPPD plans to make these changes to the logic during the Cycle 22 Refueling Outage, currently scheduled for January 15, 2005. These revised TS are needed to support CNS operation with the modified logic.

2.0 Proposed Change

The following are the three changes proposed in this amendment request.

1. TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," identifies the plant parameters (functions) that cause isolation of primary containment. The table also identifies the plant modes in which the functions are applicable, the number of channels required for each trip system, applicable surveillance requirements (SRs), and the Allowable Value for each function that causes primary containment isolation.

   Function 5 is "Reactor Water Cleanup (RWCU) System Isolation." The current function 5.d is "Reactor Vessel Water Level – Low (Level 3)," with an Allowable Value of "≥ 3 inches." The proposed change is to revise function 5.d to "Reactor Vessel Water Level – Low Low (Level 2)," and to lower the Allowable Value to "≥ -42 inches."

2. TS Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," identifies the plant parameters (functions) that cause SC isolation. The table also identifies the plant modes in which the functions are applicable, the number of channels required for each

trip system, the applicable SRs, and the Allowable Value for each function that causes SC isolation.

The current function 1 is "Reactor Vessel Water Level – Low (Level 3)," with Allowable Value of "≥ 3 inches." The proposed change is to revise function 1 to "Reactor Vessel Water Level – Low Low (Level 2)," and to lower the Allowable Value to "≥ -42 inches."

3. TS Table 3.3.7.1-1, "Control Room Emergency Filter System Instrumentation," identifies the plant parameters that cause start of the Control Room Emergency Filter System (CREFS). The table also identifies the plant modes in which the functions are applicable, the number of channels required for each trip system, applicable SRs, and the Allowable Value for each function that causes start of CREFS.

   The current function 1 is "Reactor Vessel Water Level – Low (Level 3)," with Allowable Value of "≥ 3 inches." The proposed change is to revise function 1 to "Reactor Vessel Water Level – Low Low (Level 2)," and to lower the Allowable Value to "≥ -42 inches."

Conforming changes to the Bases to reflect that these three actions will occur at vessel water Level 2 instead of Level 3 are needed. These changes are included in this amendment request for information.

In summary, NPPD requests amendment of the CNS operating license to lower the reactor vessel water level at which three automatic actions occur. The three automatic actions are (1) closure of valves in the Reactor Water Cleanup System, a system that connects with the reactor vessel, thereby isolating the system from the reactor vessel, (2) isolation of secondary containment with attendant start of the Standby Gas Treatment System, a system that filters radioactive particulates and gases from the atmosphere of the Reactor Building, and (3) start of the Control Room Emergency Filter System, a system that filters the air supplied to the Main Control Room during an emergency. The Allowable Value of reactor vessel water level for isolation of the RWCU System, isolation of SC, and start of CREFS, is being lowered by 45 inches, from ≥ 3 inches (161.19 inches above TAF) to ≥ -42 inches (116.19 inches above TAF.)

3.0 Background

The changes to the logic in this proposed amendment were recommended by General Electric, the supplier of the CNS Nuclear Steam Supply System, in Service Information Letter (SIL) 131, "Containment Isolation Logic Change," dated March 31, 1975. SIL 131 states that reactor scrams from power levels greater than 50% may result in reactor vessel water level transients below the reactor low water level scram setpoint (Level 3) due to void collapse (shrink), and that the resulting isolation of RWCU causes problems such as loss of the filter media in the filter-demineralizers, and loss of the ability to remove water from the reactor vessel immediately after a scram, which complicate scram recovery. The SIL also discusses how isolation of secondary containment can cause increased temperature in the reactor building, and result in a thermal transient on the building and equipment. The requested revision of the logic for RWCU isolation and SGT initiation to lower reactor vessel water level from Level 3 to Level 2 were identified as Recommendations 1 and 2, respectively, in GE SIL no. 131.

Isolation of Reactor Building normal ventilation, which occurs as part of Secondary Containment isolation, results in isolation of cooling air supply to the Reactor Recirculation Motor Generator (RRMG) sets. If the cooling air supply is not restored promptly, the RRMG sets may trip due to high air temperature, resulting in tripping of the Reactor Recirculation pumps. Tripping of these pumps results in loss of forced recirculation flow through the core. Combined with an isolated RWCU system, loss of forced recirculation flow through the core results in temperature stratification in the reactor vessel. Such temperature stratification results in complication of scram recovery and vessel level control.

By reducing the potential for unnecessary isolations of RWCU and SC, with attendant initiation of Standby Gas Treatment (SGT) and CREFS, in the event of lowered vessel level following a reactor scram, this logic change will reduce the potential for post-scram operator challenges. Eliminating post-scram operator challenges will enhance plant safety.

The following presents background information regarding plant systems involved with each of the three proposed changes to TS.

1. TS Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, Function 5.d, Reactor Water Cleanup System Isolation on Reactor Vessel Water Level.

   The primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

   Primary containment isolation instrumentation has inputs to the trip logic of RWCU isolation on reactor vessel level. Functional diversity is provided by monitoring a wide range of independent parameters. Reactor vessel level is one of the input parameters to the isolation logic.

   Reactor Vessel Water Level – Low (Level 3) signals are initiated from four level switches. Eight channels of Reactor Vessel Water Level – Low (Level 3) Function are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.

   The RWCU system performs no safety function. The RWCU system removes impurities from the reactor coolant water by continuously removing a portion of the reactor coolant from the bottom head drain and the suction side of a reactor recirculation pump. The removed flow is sent through filter-demineralizer units for mechanical filtration and ion exchange processes. The processed fluid is returned to the reactor through the reactor feedwater line.

2. TS Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation, Function 1, Reactor Vessel Water Level.

   SC isolation instrumentation automatically initiates closure of appropriate SC isolation valves and starts the SGT System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following a postulated DBA. SC isolation and establishment of vacuum with the SGT System within the required time limits ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be operable are maintained within applicable limits.

   Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (1) reactor vessel water level, (2) drywell pressure, and (3) reactor building ventilation exhaust plenum radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation.

   The design basis of the SGT System is to mitigate the consequences of a loss-of-coolant accident (LOCA) and fuel handling accident. For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment. The functions of the SGT System are (1) to filter particulates and iodines from the air being exhausted from the Reactor Building in order to reduce the level of airborne contamination released to the environs and (2) to reduce and hold the building at a minimum average subatmospheric pressure of 0.25 inches of water (under neutral wind conditions, i.e., wind speeds of 2 to 5 mph).

   The SGT System automatically starts and operates in response to actuation signals indicative of conditions of an accident that could require operation of the system. Following initiation, both SGT filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

3. TS Table 3.3.7.1-1, Control Room Emergency Filter System Instrumentation, Function 1, Reactor Vessel Water Level.

   CREFS is designed to provide a radiological controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The instrumentation and controls for CREFS automatically isolate the normal ventilation intake and initiate action to pressurize the main control room and filter incoming air to minimize the infiltration of radioactive material into the control room environment.

In the event of a LOCA signal (Reactor Vessel Water Level – Low, Level 3 [to be changed to Low Low, Level 2] or Drywell Pressure – High) or Reactor Building Ventilation Exhaust Plenum Radiation – High signal, the normal control room inlet supply damper closes and CREFS is automatically started in the emergency bypass mode. The air drawn in from the outside passes through a high efficiency filter and a charcoal filter in sufficient volume to maintain the control room slightly pressurized with respect to the adjacent areas.

The ability of CREFS to maintain the habitability of the control room is explicitly assumed for certain accidents as discussed in the USAR safety analyses. CREFS operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents that assume CREF System operation, does not exceed the limits set by General Design Criterion (GDC) 19 in 10 CFR 50, Appendix A.

4.0 Technical Analysis

The proposed logic changes associated with this amendment request are to lower the reactor vessel water level at which RWCU isolates, SC isolates (with attendant start of SGT System), and CREFS initiates, from Level 3 to Level 2. The Allowable Value for Level 3 is ≥ 3 inches, and for Level 2 is ≥ -42 inches. (These values of reactor vessel water level are relative to a level designated as Instrument Zero, which is 516.75 inches above the bottom of the reactor pressure vessel, and 158.19 inches above TAF.) Level 2 is the level at which HPCI and RCIC initiate.

SIL 131 states that lowering the reactor vessel level for RWCU isolation from Level 3 to Level 2 is justified if an RWCU break detection system is added for automatic isolation on a RWCU System break. CNS has a functional RWCU leak/break detection system that was part of the original design. The RWCU leak/break detection system provides for automatic isolation upon detection of either high flow in the RWCU System or high temperature in the vicinity of high temperature RWCU piping. (Area high temperature in the vicinity of the high temperature RWCU System equipment indicates a leak or a break in the system. High flow rate in the RWCU System indicates a break in the RWCU System.) For leaks/breaks that are not detected by the RWCU leak detection system manual operator actions are taken, based on other indications, to isolate RWCU.

The isolation signals of high RWCU flow and high RWCU space temperature are in CNS TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," as Function 5.a ("RWCU Flow-High") and Function 5.b ("RWCU System Space Temperature-High.")

Isolation of the RWCU System is initiated when high flow is sensed to prevent exceeding offsite dose limits. The RWCU Flow – High function receives input from two channels, with each channel in one trip system using a one-out-of-one logic, with one channel tripping the inboard RWCU isolation valve and one channel tripping the outboard RWCU isolation valve. The high RWCU flow signals are initiated from differential pressure switches that are connected to an elbow flow tap on the inlet pump suction line of the

RWCU System. Two channels are available and are required to be operable to ensure that no single instrument failure can preclude the isolation function.
RWCU isolation on high space temperature occurs even when very small leaks have occurred and is diverse to the high flow instrumentation for the hot portions of the RWCU System. The RWCU System Space Temperature – High function receives input from 48 bimetallic temperature switches. These switches are physically located at six locations along and in the vicinity of the RWCU System high temperature piping in groups of eight. For each location four switches input into trip system A, and the other four switches input into trip system B. Each set of four switches is arranged in a one-out-of-two taken twice logic in series with a normally deenergized trip relay. Trip system A isolates the RWCU supply line inboard isolation valve, and trip system B isolates the RWCU supply line outboard isolation valve.

Thus, CNS satisfies the SIL 131 provision that a RWCU System break detection system be provided.

The following discusses why the proposed changes to the logic for isolation of RWCU and SC (with attendant SGT start) and start of CREFS are acceptable.

<u>Isolation of the Reactor Water Cleanup System</u>

With the current logic Group 3 isolation (RWCU isolation) occurs on one of four signals, as identified by Function 5 in TS Table 3.3.6.1-1. These four signals are:

1. Low Reactor Vessel Water Level – Level 3. The Allowable Value (AV) in TS is "≥ +3 inches." This level is equivalent to 161.19 inches above TAF.
2. High flow in RWCU. The AV in TS is "≤ 191% flow."
3. High temperature in RWCU space. The AV in TS is "≤ 195°F."
4. Standby Liquid Control System initiation. There is no AV in TS.

The proposed change to the logic is to lower the reactor vessel water level at which RWCU would isolate. The isolation would occur at Reactor Vessel Water Level Low Low – Level 2. The AV for this level is – 42 inches. This is 116.19 inches above TAF. As explained above RWCU will continue to isolate in response to a break or leak in the system based on isolation signals 2 and 3, or in response to operation of the Standby Liquid Control System based on isolation signal 4. No changes are made to isolation signals 2, 3, or 4.

<u>Isolation of Secondary Containment and Start of Control Room Emergency Filter System</u>

With the current logic a Group 6 isolation (SC isolates, Primary Containment [Drywell and Suppression Chamber] Atmospheric Control Isolation Valves close, Primary Containment Atmospheric Monitor Isolation Valves close, and SGT and CREFS start) occurs on one of three signals. These three signals are:

1. Low Reactor Vessel Water Level – Level 3. The Allowable Value (AV) in TS is "≥ +3 inches." This level is equivalent to 161.19 inches above TAF.
2. High Drywell Pressure. The AV in TS is "≤ 1.84 psig."
3. High radiation in the Reactor Building Ventilation Exhaust Plenum. The AV in TS is "≤ 49 mR/hr."

The proposed change to the logic is to lower the reactor vessel water level (signal 1) at which the above actions would occur to Low Low Reactor Vessel Water Level – Level 2. This has a AV in TS of "≥ – 42 inches." This is 116.19 inches above TAF. High drywell pressure (signal 2) will occur in response to a LOCA, and will continue to result in the above actions. The high radiation in Reactor Building exhaust plenum (signal 3) will occur in response to fuel damage, and will continue to result in the above actions. No changes are made to isolation signals 2 or 3.

<u>Impact on Station Safety Analysis</u>

The four design basis accidents (DBAs) are discussed in the CNS Station Safety Analysis in USAR Chapter XIV. They are (1) Control Rod Drop Accident, (2) Loss-of-Coolant Accident (LOCA), (3) Fuel Handling Accident (FHA), and (4) Main Steam Line Break. The USAR discussion of these DBA analyses was reviewed to determine the impact of the lowered reactor vessel level. The following is a summary of that review and the impacts.

1. Control Rod Drop Accident

   The Control Rod Drop Accident for CNS is discussed in USAR Section XIV-6.2. No reactor coolant boundary breach is associated with this accident. Reactor vessel inventory is not a consideration other than initially being at a normal level. (Normal reactor vessel level is consistent with the conservative intent of the analysis since normal level would contribute to neutron thermalization and power production.) To achieve the most limiting results, initial reactor power is assumed to be less than 10 percent. A reactor scram occurs following the control rod drop due to high neutron flux.

   The existing analysis assumes that the CREF System is manually initiated after 24 hours. SIL 131 indicates that scrams from greater than 50 percent usually result in a Level 3 trip. Due to the low initial power, it is unlikely that reactor water level would drop to the low level (Level 3) trip during the post-scram reactor water level transient (shrink), and cause an RWCU isolation and SGT and CREFS initiation. However, since there is no reactor coolant pressure boundary breach, the reactor vessel Level 3 trip is not a mitigating action for this event. Consequently, the change in isolation and SGT system initiating vessel level has no affect on the plant response to a Control Rod Drop Accident or on any associated analyses.

   The conclusions stated in the USAR that (1) this accident will not result in any radiological doses that endanger the health and safety of the public, and (2) the Control

Room occupant doses are within the limits of 10 CFR 50 Appendix A, General Design Criteria (GDC) 19, remain valid with the proposed logic changes.

2. Loss of Coolant Accident

The Loss of Coolant Accident (LOCA) analysis for CNS is discussed in USAR Section XIV-6.3. A LOCA results in high pressure in the drywell and low reactor vessel water level. The high drywell pressure signal will cause a reactor scram in less than one second from receipt of the signal, which occurs almost concurrently with the line break. High drywell pressure signal also initiates Primary Containment Isolation System (PCIS) Group 2 (RHR shutdown cooling, drywell floor and equipment drains, RHR to Radwaste isolation valves) and Group 6 (SC, SGT and CREFS initiation) isolations. (A complete description of the PCIS signals and group isolations is provided in CNS USAR Table VII-3-6.) No changes are proposed to the high drywell pressure portion of the logic.

Currently, decreasing vessel level initiates Group 3 (RWCU) isolation at Level 3 (Low Water Level) and Groups 1 (closure of main steam isolation valves and drains) and 7 (closure of reactor water sample valves) isolations at Level 1 (Low Low Low Water Level.) The effect is an isolated Primary and Secondary Containment with SGT System in operation.

The proposed changes will cause a Group 3 isolation to occur at Level 2 (TS Allowable Value of ≥ -42 inches) rather than Level 3 (TS Allowable Value of ≥ 3 inches). This is a decrease in reactor vessel level of 45 inches. However, nearly 10 feet of water still remains above the top of active fuel.

In the event offsite power is unavailable the Group 3 and Group 6 isolations initiate concurrent with the loss of power. In this case the proposed logic changes would result in no change in the CNS response.

The DBA large break LOCA results in Group 6 initiating on high drywell pressure essentially instantaneously with the break. This occurs before vessel water level would drop to Level 3 in the current logic. Consequently, the change from level 3 to level 2 does not alter the CNS response to the DBA LOCA, including SC isolation and initiation of SGT and CREFS.

The conclusions stated in the USAR that (1) this accident will not result in any radiological doses that endanger the health and safety of the public, and (2) that the Control Room occupant doses are within the limits of 10 CFR 50 Appendix A, General Design Criteria (GDC) 19, remain valid with the proposed logic changes.

3. Fuel Handling Accident

The design basis Fuel Handling Accident (FHA) is discussed in USAR Section XIV-6.4. The FHA results in high radiation levels in the reactor building exhaust plenum causing Group 6 isolation, which isolates the normal reactor building ventilation, and starts SGT and CREFS. The reactor building ventilation exhaust high radiation trip function is unaffected by the proposed lowering of reactor vessel water level. Reactor coolant inventory is unaffected by the FHA. The proposed change to the vessel water level isolation logic signal will have no effect on CNS response to the FHA.

It is concluded, therefore, that the proposed logic changes will not result in a revision of the existing analyses including both the onsite and offsite radiological consequences. The conclusions stated in the USAR that (1) the offsite radiological doses are well within the 10 CFR 100 dose limits, and (2) the Control Room occupant doses are within the limits of 10 CFR 50 Appendix A, GDC 19, remain valid with the proposed logic changes.

4. Main Steam Line Break

The design basis Main Steam Line Break (MSLB) accident is discussed in USAR Section XIV-6.5. The CNS response to the MSLB is affected to a slight degree by the proposed logic changes. Specifically, the vessel level induced initiation of PCIS Group 3 (RWCU isolation) and PCIS Group 6 (SC isolation, SGT and CREFS initiation) will occur later in the sequence of events since Level 2 is reached after Level 3. (The slight delay, however, has no effect on reactor vessel water inventory). If a concurrent loss of offsite power is assumed, the Group 3 and Group 6 isolations initiate with the loss of power and the proposed logic change has no effect.

The delay in initiating the Group 6 isolation means that CREFS initiation occurs slightly later in the sequence of events. However, CREFS is not credited with providing any filtration of the intake air following a design basis MSLB. The evaluation of offsite and Control Room doses assumes that the release from the steam line break outside SC forms a cloud which completely passes over the Control Room air intake in less than one minute followed by a continuous supply of clean air. After the cloud passes Control Room ventilation is assumed to convert to the CREF System, which substantially reduces the air intake flow rate, and consequently, the dilution rate. Since only clean air is being drawn in, no filtration is assumed.

Therefore, it is concluded that, with respect to Control Room personnel radiological dose following a design basis MSLB, this change does not result in an adverse consequence and is bounded by the existing analysis.

For main steam line breaks occurring within SC, no credit is given to SGT since SC is not assumed to remain intact. For main steam line breaks occurring outside SC, the difference in initiation levels for SGT (via the Group 6 isolation logic) has no impact since the release occurs outside SC.

The conclusions stated in the USAR that (1) the health and safety of the public is not endangered as a consequence of this accident, and (2) that the Control Room occupant doses are within the limits of 10 CFR 50 Appendix A, General Design Criteria (GDC) 19, remain valid with the proposed logic changes.

Conclusion

The CNS system for detecting a leak or break in the RWCU System and initiating isolation of RWCU is not impacted by the proposed change to the logic. No change is made to the logic for isolation of SC and initiation of SGT and CREFS on high drywell pressure or high radiation in the Reactor Building exhaust plenum. The proposed change to lower the value of reactor vessel water level at which RWCU and SC would isolate and SGT and CREFS would start does not adversely impact CNS response to the design basis accidents as discussed in USAR Section XIV-6, Station Safety Analysis. No changes are made to the Reactor Protection System logic for initiating automatic reactor shutdown or to the Emergency Core Cooling System initiation logic. Therefore, the proposed changes to the CNS operating license do not impact the ability of CNS to continue to operate safely.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

10 CFR 50.91(a)(1) requires that licensee requests for operating license amendments be accompanied by an evaluation of significant hazard posed by issuance of an amendment. Nebraska Public Power District (NPPD) has evaluated this proposed amendment with respect to the criteria given in 10 CFR 50.92 (c).

NPPD is requesting an amendment of the operating license for the Cooper Nuclear Station (CNS.) The requested amendment proposes to lower the reactor vessel water level at which (1) valves in a system that connects with the reactor vessel would automatically close, thereby closing this system off from the reactor vessel, (2) a system that filters radioactive particulates and gases from the atmosphere of the reactor building would automatically start, and (3) a system that filters the air being supplied to the plant's control room would automatically start.

The system that connects with the reactor vessel whose automatic closure setpoint is proposed to be changed is the Reactor Water Cleanup (abbreviated as "RWCU") System. The purpose of this system is to remove impurities from the water that circulates through the reactor vessel. The reason that the valves in this system would automatically close is to ensure that as much of the available water is retained in the reactor vessel.

The RWCU System currently isolates on reactor vessel water level dropping to 3 inches above a vessel level referred to as "instrument zero" (equivalent to 161.19 inches above the top of the active fuel in the reactor core), flow in the RWCU

System increasing to 191% of rated flow, temperature in the RWCU System space increasing to 195°F, and initiation of the Standby Liquid Control System.

The system that filters the atmosphere of the reactor building is called the Standby Gas Treatment (abbreviated as "SGT") System. This filtering of any radioactivity in the building atmosphere would reduce the amount that might be released to the environment in the event of an accident at the plant. The SGT System initiates automatically so that the filtering of any radioactivity in the building atmosphere would be started without the need for any human action.

The system that filters the air being supplied to the plant's control room in the event of an accident is called the Control Room Emergency Filter System (abbreviated as "CREFS.") The filtering of the air supplied to the control room will reduce the occupational dose that the plant operators would receive in the event of an accident at the plant. CREFS initiates automatically so that the filtering of any radioactivity in the control room atmosphere would be started without the need for any human action.

Both SGT System and CREFS currently start on reactor vessel water level dropping to 3 inches above "instrument zero" (equivalent to 161.19 inches above the top of the active fuel in the reactor core), pressure in the drywell (the primary containment structure that houses the reactor pressure vessel) increasing above 1.84 psig, or the level of radiation in the Reactor Building increasing above 0.049 Rem per hour (49 milli-Rem/hour).

It is proposed that the reactor vessel water level at which the RWCU System would automatically isolate and at which the SGT System and CREFS would automatically start be lowered by 45 inches. The new reactor vessel water level would be 42 inches below instrument zero (equivalent to 116.19 inches above the top of the active fuel in the reactor core.) No changes are being made to the high flow or high temperature RWCU isolation signals.

This change in the reactor vessel water level for RWCU isolation and SGT start was recommended by General Electric, the supplier of the CNS Nuclear Steam Supply System. This lowering of the RWCU isolation and the SGT initiation setpoints will improve the ability of the plant operating staff to respond to possible automatic shutdown of the reactor.

The following evaluation supports a finding of "no significant hazards" for the proposed amendment.

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   No. The values of various plant parameters at which piping connected to the reactor vessel and containment isolates and air-filtering systems start are not accident precursors. Thus, lowering the reactor vessel water level at which RWCU and secondary containment isolate and SGT and CREFS initiate has no impact on the probability of a design basis accident evaluated in the CNS Station Safety Analysis. Therefore the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed logic changes involve no changes to the logic of the Reactor Protection System that initiates automatic reactor shutdown in response to an accident. The proposed logic changes involve no changes to the logic of the Emergency Core Cooling System (ECCS) that initiates automatic actions to ensure adequate core cooling and containment integrity in response to an accident. The CNS response to the design basis accidents (DBAs) addressed in the Station Safety Analysis with the proposed changes to the logic was evaluated. This evaluation has demonstrated that there is no increase in the offsite radiological doses to the public resulting from these accidents.

Based on the above NPPD concludes that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed lowering of the level of water in the reactor vessel at which certain automatic actions would occur changes the operation of various systems at CNS. However, the change in system operation is not significant. Currently automatic actions occur in the RWCU System, SGT System, CREFS, and secondary containment in response to reactor vessel water level. Changing the level at which these automatic actions occur is not a significant change in the systems operation. Hardware changes needed to implement the modified logic are minor. Lowering the reactor vessel water level for these actions does not introduce a new mode of plant operation and does not create a potential for any new failure mechanisms, malfunctions, or accident initiators. Making this change does not involve adding new systems to the CNS design.

   Based on the above NPPD concludes that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

   The safety margin associated with dose consequences to the public following DBAs is based on, in part, automatic operation of systems that shut down the reactor, automatic initiation of ECCS, and automatic isolation of primary and secondary containment. The proposed changes to the CNS TS make no changes that affect the automatic shutdown of the reactor or the automatic initiation and operation of ECCS. The plant response to DBAs with the proposed revisions to the RWCU isolation (primary containment) and SGT and CREFS initiation (secondary containment) have been evaluated and shown to not result in any increase in dose to the public. The safety margin associated with dose consequences to the control room operators is based on automatic isolation of secondary containment, and initiation of CREFS. The plant response to DBAs with the proposed revisions to the RWCU isolation (primary containment) and SGT and CREFS initiation (secondary containment) have been evaluated and shown to not result in any increase in dose to the control room operators.

Based on the above NPPD concludes that the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above evaluation of the three criteria, NPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

1. 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." Paragraph 100.11(a)(1) requires that an exclusion area be established with dose limits of 25 Rem whole body and 300 Rem thyroid from iodine exposure to an individual at the boundary for two hours following onset of fission product release from an accident.

   As discussed in Section 4.0, "Technical Analysis" the proposed logic changes do not result in any change in the offsite dose consequences of the design basis accidents presented in the CNS Station Safety Analysis (USAR Section XIV-6.)

2. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19, "Control Room." This criterion requires, in part, that the control room for the nuclear power unit be provided with adequate radiation protection to permit access and occupancy under accident conditions without personnel receiving radiation exposures in excess of 5 Rem whole body, or its equivalent to any part of the body, for the duration of the accident. CNS is committed to the radiation protection provisions of this GDC.

   As discussed in Section 4.0, "Technical Analysis" the proposed logic changes do not result in any change in the control room dose consequences of the design basis accidents presented in the CNS Station Safety Analysis (USAR

Section XIV-6.) As a result GDC 19 continues to be satisfied with the proposed logic changes.

CNS continues to satisfy applicable regulatory requirements with the proposed changes to the logic.

6.0 Environmental Consideration

10 CFR 51.22(b) allows that an environmental assessment (EA) or an environmental impact statement (EIS) is not required for any action included in the list of categorical exclusions in 10 CFR 51.22(c). 10 CFR 51.22(c)(9) identifies an amendment to an operating license which changes a requirement with respect to installation or use of a facility component located within the restricted area, or which changes an inspection or a surveillance requirement, as a categorical exclusion 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 amount of any effluents that may be released off-site, or (3) result in an increase in individual or cumulative occupational radiation exposure.

NPPD has reviewed the proposed license amendment and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no EIS or EA needs to be prepared in connection with issuance of the proposed license changes. The basis for this determination is as follows:

1. The No Significant Hazards Consideration evaluation presented in the above Section 5.1 concluded that the requested change in the logic for isolation of RWCU and secondary containment and start of SGT and CREFS does not involve a significant hazards consideration.

2. Isolation of RWCU and secondary containment, and start of SGT and CREFS, or the setpoints of various parameters at which these automatic actions occur, does not impact any effluent at CNS. Therefore, the proposed lowering of the reactor vessel water level setpoint at which these automatic actions would occur 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. One of the CREFS safety design basis is to limit the radiation exposure to main control room personnel during any one of the postulated design basis events to within regulatory limits. Therefore, a change in when CREFS would start following a design basis accident might result in a change in the radiological dose that personnel in the main control room might receive. CREFS automatically initiates on low water level in the reactor vessel, high pressure in containment, or high radiation in the Reactor Building (RB) ventilation exhaust plenum. No change is being made to the CREFS initiation on high containment pressure or high RB radiation. Lowering the reactor vessel water level at which CREFS would start will not impact the dose to control room personnel because CREFS

will continue to start in response to high drywell pressure or high radiation in the Reactor Building. Therefore, the requested lowering of the reactor vessel water level setpoint for this logic does not significantly increase individual or cumulative occupational radiation exposure.

7.0 References

1. Monticello Nuclear Generating Plant – Amendment No. 91

   Amendment No. 91 dated September 9, 1994, to License No. DPR-22, issued revised technical specifications to Monticello that changed the initiating parameter for secondary containment isolation and standby gas treatment initiation from Low Reactor Water Level to Low-Low Reactor Water Level. This amendment was issued in response to a license amendment request from Northern State Power Company dated March 28, 1994. The NRC issued the amendment in response to the initial license submittal without a request for additional information.

2. Monticello Nuclear Generating Plant – Amendment No. 117

   Amendment No. 117 dated March 7, 2001, to License No. DPR-22, issued revised technical specifications to Monticello for numerous changes. One of the changes was to lower the RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level. This amendment was issued in response to a license amendment request from Nuclear Management Company, LLC, dated July 20, 2000. The NRC issued the amendment in response to the initial license submittal without a request for additional information.

3. General Electric Service Information Letter (SIL) 131, "Containment Isolation Logic Change" dated March 31, 1975.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION REVISIONS
(MARK-UP)

COOPER NUCLEAR STATION
NRC DOCKET 50-298, LICENSE DPR-46

Technical Specification Pages
3.3-52
3.3-57
3.3-63

Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 250% rated steam flow
b. HPCI Steam Line Flow-Time Delay Relays	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 6 seconds
c. HPCI Steam Supply Line Pressure – Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 107 psig
d. HPCI Steam Line Space Temperature – High	1,2,3	2 per location	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 288% rated steam flow
b. RCIC Steam Line Flow-Time Delay Relays	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 6 seconds
c. RCIC Steam Supply Line Pressure – Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 61 psig
d. RCIC Steam Line Space Temperature – High	1,2,3	2 per location	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 191% of Rated
b. RWCU System Space Temperature – High	1,2,3	2 per location	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
c. SLC System Initiation	1,2	1	H	SR 3.3.6.1.6	~~NA~~
d. Reactor Vessel Water Level – Low Low (Level ~~3~~ 2)	1,2,3	4	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ ~~3~~ -42 inches

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level —Low Low (Level ~~3~~ 2)	1,2,3, (a)	4	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ ~~3~~ -42 inches
2. Drywell Pressure —High	1,2,3	4	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation —High	1,2,3, (a),(b)	4	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 49 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filter System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low (Level ~~3~~) Low 2	1,2,3, (a)	4	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	-42 ≥ ~~3~~ inches
2. Drywell Pressure - High	1,2,3	4	SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	4	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 49 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.

ATTACHMENT 3
PROPOSED TECHNICAL SPECIFICATION REVISIONS
(FINAL TYPED)

COOPER NUCLEAR STATION
NRC DOCKET 50-298, LICENSE DPR-46

Technical Specification Pages
3.3-52
3.3-57
3.3-63

Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 250% rated steam flow
b. HPCI Steam Line Flow-Time Delay Relays	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 6 seconds
c. HPCI Steam Supply Line Pressure - Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 107 psig
d. HPCI Steam Line Space Temperature - High	1,2,3	2 per location	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 288% rated steam flow
b. RCIC Steam Line Flow-Time Delay Relays	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 6 seconds
c. RCIC Steam Supply Line Pressure - Low	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 61 psig
d. RCIC Steam Line Space Temperature - High	1,2,3	2 per location	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
5. Reactor Water Cleanup (RWCU) System Isolation					
a. RWCU Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 191% of Rated
b. RWCU System Space Temperature - High	1,2,3	2 per location	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 195°F
c. SLC System Initiation	1,2	1	H	SR 3.3.6.1.6	NA
d. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3	4	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ - 42 inches

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3, (a)	4	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	$\geq$ - 42 inches
2. Drywell Pressure - High	1,2,3	4	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	$\leq$ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	4	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	$\leq$ 49 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filter System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low (Level 2)	1,2,3, (a)	4	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	$\geq$ - 42 inches
2. Drywell Pressure - High	1,2,3	4	SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	$\leq$ 1.84 psig
3. Reactor Building Ventilation Exhaust Plenum Radiation - High	1,2,3, (a),(b)	4	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	$\leq$ 49 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.

ATTACHMENT 4
PROPOSED TECHNICAL SPECIFICATIONS BASES REVISIONS
MARKUP FORMAT

COOPER NUCLEAR STATION
NRC DOCKET 50-298, LICENSE DPR-46

Technical Specification Bases Pages

B3.3-141
B3.3-155
B3.3-156
B3.3-169
B3.3-170
B3.3-185
B3.3-187

Note: TS Bases pages are provided for information. Following approval of the proposed TS change, Bases changes will be implemented in accordance with TS 5.5.10, "Technical Specification (TS) Bases Control Program."

BASES

BACKGROUND

3. 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation (continued)

drain pot drain valves, and cause a HPCI turbine trip which closes the HPCI minimum flow valve. The Functions associated with RCIC cause a RCIC turbine trip which closes the RCIC minimum flow valve.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level—Low Low (Level ~~3~~ 2) Isolation Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected into two one-out-of-two taken twice trip systems. The RWCU Flow—High Function receives input from two channels, with each channel in one trip system using a one-out-of-one logic, with one channel tripping the inboard RWCU isolation valve and one channel tripping the outboard RWCU isolation valve. The RWCU System Space Temperature-High Function receives input from 48 bimetallic temperature switches. These switches are physically located along and in the vicinity of the RWCU system high temperature piping in groups of eight (8). Thus, there are six (6) locations; RWCU HX room NW (RWCU supply line), RWCU pump rooms (2 locations), RWCU HX room (pump discharge line to Regenerative HX), torus area south, and torus area east. For each location, four switches input into trip system A, the other four switches input into trip system B. Each set of four switches is arranged in a one-out-of-two taken twice logic in series with a normally deenergized trip relay. Actuation of the correct combination of two switches will initiate the corresponding trip system. Trip system A isolates the RWCU supply line inboard isolation valve, and trip system B isolates the RWCU supply line outboard isolation valve. For purposes of this specification, each temperature switch is considered a "channel".

The SLC System Isolation Function receives input from two channels (one channel in each trip system), arranged in a one-out-of-one logic. A channel consists of one of the two control room SLC pump start switches which inputs directly into one of the two RWCU isolation logic trip systems. Placing the SLC Pump A control switch to "Start" will isolate the RWCU inboard isolation valve. Placing the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>5.b. RWCU System Space Temperature - High</u> (continued)

system must be in different trip strings, i.e., they must be connected such that isolation occurs when both required channels actuate (one of two parallel logic pairs of switch channels in one trip string must trip in combination with the tripping of one of two additional parallel logic switch channels in the other trip string in order to actuate the trip system.

The RWCU System Space Temperature—High Allowable Values are set low enough to detect a leak.

These Functions isolate the Group 3 valves, as listed in Reference 1.

<u>5.c. Standby Liquid Control (SLC) System Initiation</u>

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 9). RWCU isolation signals from the SLC system actuation are initiated from the two control room SLC pump start signals.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels of the SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

This Function isolates the inboard and outboard RWCU suction valves.

<u>5.d. Reactor Vessel Water Level - Low, Low (Level ~~3~~ 2)</u>

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level ~~3~~ 2 supports actions to ensure that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.d. Reactor Vessel Water Level—Low Low (Level ~~3~~ 2) (continued)

peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level—~~Low (Level 3)~~ Function associated with RWCU isolation is not directly assumed in the USAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

Low Low (Level 2)

Reactor Vessel Water Level—~~Low (Level 3)~~ signals are initiated from four level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Eight channels of Reactor Vessel Water Level—~~Low (Level 3)~~ Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—~~Low (Level 3)~~ Allowable Value was chosen to be the same as the ~~ECCS~~ Reactor Vessel Water Level—~~Low (Level 3)~~ Allowable Value (LCO 3.3.5.1~~)~~ and LCO 3.3.5.2), since this could indicate that the capability to cool the fuel may be threatened.

This Function isolates the Group 3 valves, as listed in Reference 1.

High Pressure Coolant Injection/ Reactor Core Isolation Cooling (HPCI/RCIC)

Shutdown Cooling System Isolation

6.a. Reactor Pressure - High

The Reactor Pressure—High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This Function is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the USAR.

The Reactor Pressure—High signals are initiated from two pressure switches that are connected to different taps on a recirculation pump suction line. Two channels of Reactor Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis or appropriate documents. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when SCIVs and the SGT System are required.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low Low (Level ~~3~~ 2)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The Reactor Vessel Water Level ~~- Low (Level 3)~~ Low-Low (Level2) Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation systems on Reactor Vessel Water Level ~~- Low (Level 3)~~ support actions to ensure that any offsite releases are within the limits calculated in the safety analysis.

Reactor Vessel Water Level ~~- Low (Level 3)~~ signals are initiated from level switches that sense the difference between the pressure due to a constant column of water

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level-Low (Level 3) (continued)

(reference leg) and the pressure due to the actual water level (variable leg) in the ~~vessel.~~ ~~Eight~~ channels of Reactor Vessel Water Level—Low Low (Level ~~3~~ 2) Function are available and are required ~~to be OPERABLE~~ in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low (Level ~~3~~ 2) Allowable Value was chosen to be the same as the ~~RPS Level scram~~ High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level-Low Low (Level 2) Allowable Value ~~(LCO 3.3.1.1) to enable initiation of isolation at the earliest indication of a breach in the nuclear system process barrier, yet far enough below normal operational levels to avoid spurious isolation.~~ (LCO 3.3.5.1 and LCO 3.3.5.2) since this could indicate that the capability to cool the fuel is being threatened.

The Reactor Vessel Water Level—Low Low (Level ~~3~~ 2) Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

2. Drywell Pressure-High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. The Drywell Pressure—High Function associated with isolation is not assumed in any USAR accident or transient analyses, but will provide an isolation and initiation signal. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Filter (CREF) System Instrumentation

BASES

BACKGROUND

The CREF System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The instrumentation and controls for the CREF System automatically isolate the normal ventilation intake and initiate action to pressurize the main control room and filter incoming air to minimize the infiltration of radioactive material into the control room environment.

In the event of ~~a loss of coo~~lant accident (LOCA) signal (Reactor Vessel Water Level — Low, Low Level ~~3~~ 2 or Drywell Pressure — High) or Reactor Building Ventilation ~~Exhaust~~ Plenum Radiation — High signal, the normal control room inlet supply damper closes and the CREF System is automatically started in the emergency bypass mode. The air drawn in from the outside passes through a high efficiency filter and a charcoal filter in sufficient volume to maintain the control room slightly pressurized with respect to the adjacent areas.

The CREF System instrumentation has two trip systems. Each trip system includes the sensors, relays, and switches necessary to cause initiation of the CREF System. Each trip system receives input from each of the Functions listed above (each sensor ~~sends a signal to~~ both trip systems). The Reactor Vessel Water Level — Low, Low Level ~~3~~ 2, Drywell Pressure — High, and Reactor Building Ventilation ~~Exhaust~~ Plenum Radiation — High are each arranged in a one-out-of-two taken twice logic for each trip system. The channels include electronic and electrical equipment (e.g., switches and trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREF System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the control room is explicitly assumed for certain accidents as discussed in the USAR safety analyses (Refs. 1, 2, and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents that assume CREF System operation, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level — Low Low (Level ~~3~~ 2)

Low reactor pressure vessel (RPV) water level indicates that the capability of cooling the fuel may be threatened. A low reactor vessel water level could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Reactor Vessel Water Level — Low Low (Level ~~3~~ 2) signals are initiated from level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Eight channels of Reactor Vessel Water Level — Low Low (Level ~~3~~ 2) Function are available and are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude CREF System initiation.

Secondary Containment Isolation

The Reactor Vessel Water Level — Low Low (Level ~~3~~ 2) Allowable Value was chosen to be the same as the ~~RPS Level scram~~ Allowable Value (LCO 3.3.~~1.1~~ 6.2) to enable initiation of the CREF System at the earliest indication of a breach in the nuclear system process barrier, yet far enough below normal operational levels to avoid spurious initiation.

The Reactor Vessel Water Level — Low Low (Level ~~3~~ 2) Function is required to be OPERABLE in MODES 1, 2, and 3, ~~and during~~ operations with a potential for draining the reactor vessel (OPDRVs) to ensure that the Control Room personnel are protected during a LOCA. In MODES 4 and 5 at times other than OPDRVs, the probability of a vessel draindown event resulting in the release of radioactive material to the environment is minimal. Therefore, this Function is not required in other MODES and specified conditions.

2. Drywell Pressure — High

High drywell pressure can indicate a break in the reactor coolant pressure boundary. A high drywell pressure signal could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Correspondence Number: NLS2004051

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing & Regulatory Affairs Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	