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NLS2004116
September 28, 2004

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

Subject: Response to Request for Additional Information Regarding Containment Isolation Logic Change for Reactor Vessel Water Level
Cooper Nuclear Station, Docket 50-298, DPR-46

Reference: Letter to R. K. Edington, Nebraska Public Power District, from M. C. Honcharik, U.S. Nuclear Regulatory Commission, dated August 13, 2004, "Request for Additional Information Re: Containment Isolation Logic Change for Reactor Vessel Water Level (TAC No. MC3320)."

The purpose of this letter is for the Nebraska Public Power District to submit to the U. S. Nuclear Regulatory Commission the response to the request for additional information provided by the referenced letter. The response is attached.

The additional information submitted with this response does not result in a revision of the requested license amendment and does not affect the 10CFR50.91 evaluation of no significant hazards consideration submitted with the original license amendment request.

If you have any questions regarding this submittal please call Paul Fleming, Licensing Manager, at (402) 825-2774.

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

Executed on 9/28/04
(date)

Stewart B. Minahan
General Manager of Plant Operations

/rer

Attachment

A001

cc: Regional Administrator w/attachment
USNRC – Region IV

Senior Project Manager w/attachment
USNRC – NRR Project Directorate IV-1

Senior Resident Inspector w/attachment
USNRC

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

NPG Distribution w/attachment

Records w/attachment

**Response to Nuclear Regulatory Commission (NRC)
Request For Additional Information Regarding
Containment Isolation Logic Change For Reactor Vessel Water Level**

Cooper Nuclear Station, NRC Docket No. 50-298

1. NRC Request

   Discuss the instrument setpoint methodology used to calculate the reactor vessel water level allowable values.

   CNS Response

   Cooper Nuclear Station (CNS) uses the General Electric (GE) Setpoint Methodology, as specified by NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996, to determine instrument allowable values (AV). This methodology is consistent with Method 2 of the Instrumentation, Systems, and Automation Society's (ISA) Recommended Practice ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation", dated January 1, 2000. In addition, the GE Setpoint Methodology incorporates a Licensing Event Report Avoidance evaluation that can provide additional margin between the AV and the Nominal Trip Setpoint.

   Calculation of the AV for both Level 3 and Level 2 were performed applying the GE Setpoint Methodology. CNS determined that these calculated AVs were not affected by the proposed change to implement GE Service Information Letter No. 131.

   The NRC approved use of the GE Setpoint Methodology at CNS by License Amendment No. 178 dated July 31, 1998.

2. NRC Request

   The licensee stated in the May 27, 2004, submittal that reactor vessel water level is one of the input parameters to the isolation logic. The NRC staff is concerned that the proposed change from Level 3 to Level 2 would delay the protective action of the primary containment isolation, the secondary containment isolation, and the control room emergency filter system initiation. Verify that these changes will not affect the safety analyses assumptions with respect to the response time requirements by the protective actions.

   CNS Response

   Note: The isolation signals and the systems isolated for the different Groups are presented in Updated Safety Analysis Report Table VII-3-6 (pages VII-3-21 and -22).

The Reactor Pressure Vessel (RPV) water level setpoint for Reactor Water Cleanup (RWCU) isolation (Group 3) and Secondary Containment isolation (Group 6) is being lowered from Level 3 to Level 2. In addition, the RPV level at which Standby Gas Treatment System (SGTS) and Control Room Emergency Filter System (CREFS) will initiate is being lowered from Level 3 to Level 2. The RPV level for Primary Containment (Group 2) isolation is not being changed from the current Level 3.

The design basis Loss of Coolant Accident (LOCA) [a double-ended break in the recirculation system piping] inside Primary Containment will result in high drywell pressure, which will then cause reactor scram, and Groups 2 and 6 isolation. For this design basis case the proposed change in isolation from Level 3 to Level 2 will not result in a delay in the protective actions for Group 6 since high drywell pressure will occur prior to RPV level dropping to Level 3 or proposed Level 2. Primary Containment Group 2 isolation is unaffected since no change is made to RPV level isolation setpoint.

The proposed change from Level 3 to Level 2 for RWCU isolation does not change the results of (1) the fuel clad heat-up analysis because the RWCU break is bounded by the larger design basis recirculation break, (2) the long-term Primary Containment analysis because the total mass of the RWCU system has already been accounted for in the present analysis, (3) the short-term Primary Containment analysis because the peak containment pressure is reached very early in the event, prior to completion of blowdown, and adding more liquid to the available vessel liquid volume does not affect the results, or (4) the LOCA dose consequence calculation because radiological analyses contain conservative, non-mechanistic assumptions regarding the timing of releases. Additionally, the Allowable Value of vessel water level for Level 3 is 161.19 inches of water above top of active fuel (TAF) and for Level 2 is 116.19 inches above TAF. With this level of water above TAF, core damage is not expected at either Level 3 or Level 2.

For RWCU breaks outside Primary Containment, in addition to isolation on low RPV water level, Group 3 isolation occurs in response to RWCU high flow and high area temperature. Because these other isolation signals occur well before RPV water level reaches Level 3 (and therefore before reaching Level 2), the isolation signal based on RPV level is a backup to these other Group 3 isolation signals.

Lowering the RPV water level setpoint for Secondary Containment isolation (Group 6), including SGTS initiation, will not impact the dose consequence results of the safety analyses. SGTS initiation and Secondary Containment isolation limit the release of radioactivity to the environment by controlling the leakage pathways. The Group 6 isolation also initiates CREFS, which controls dose to the operators in the control room. In addition to the RPV water level, Group 6 isolation occurs, and SGTS and CREFS start, in response to high drywell pressure and high radiation in the Reactor Building exhaust plenum.

3. NRC Request

In the General Electric Service Information Letter No. 131, Containment Isolation Logic Change, dated March 31, 1975, it was recommended to add an reactor water clean up (RWCU) break detection system for automatic isolation on a cleanup system line break if the RWCU system isolation logic is changed to initiate at reactor vessel water Level 2. The purpose of the detection system is to meet the criteria for minimizing the radiological consequences of a pipe break outside the primary containment. In order to meet this requirement, the licensee stated in its submittal that CNS has a functional RWCU leak/break detection system. The 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. The high RWCU flow signals are initiated from differential pressure (DP) switches that are connected to an elbow flow tap on the inlet pump suction line of the RWCU System. Explain the system response to a RWCU line break located between the RWCU outbound isolation valve and the elbow flow tap. Also, please indicate the number of temperature elements in the area (i.e., in the RWCU heat exchanger room) and explain the logic that would initiate the protective response.

CNS Response

The first part of this request is to explain the system response to a RWCU line break located between the RWCU outboard isolation valve and the elbow flow tap.

The outboard isolation valve is located in the RWCU heat exchanger room. The high flow detection flow element is located in the same room, downstream of the outboard isolation valve. Temperature switches that initiate closure of RWCU isolation valves on high area temperature are also located in this room. A break of sufficient size, such that flow would exceed the setpoint of the flow element, would create a temperature in the RWCU heat exchanger room that exceeds the setpoint of the temperature switches. Therefore, for breaks in the RWCU piping between the outboard isolation valve and the high flow element, RWCU will isolate on high area temperature as detected by the temperature switches in the RWCU heat exchanger room.

The second part of this request was to indicate the number of temperature elements in the area of the piping between the outboard isolation valve and the flow element, and to explain the logic that would initiate the protective response.

There are eight (8) leak detection temperature switches for the RWCU leak detection system in the RWCU Heat Exchanger room. Four of these temperature switches are in Instrumentation Channel A and four are in Instrumentation Channel B. These temperature switches are wired in the respective Channel A and Channel B leak detection circuit along with the other temperature switches in the system.

The trip logic is a one out of two taken twice logic. Tripping of the right combination of two temperature switches in each channel due to sensed high temperature will result in actuation of either Channel A or Channel B Auxiliary Relay, which will result in RWCU (Group 3) system isolation.

Correction to CNS submittal dated May 27, 2004. The submittal erroneously indicated that the high flow instrument was an elbow flow tap. The high flow instrument is actually an annubar. The annubar provides a more accurate flow indication than the elbow flow tap.

4. NRC Request

For a spectrum of breaks inside containment, will the setpoint switch from Level 3 to Level 2 cause the delay to the secondary containment isolation? If it does, what is the impact on the dose calculation?

CNS Response

For breaks large enough to cause a water level decrease, the low water level signal for initiating Secondary Containment isolation (Group 6) would occur at the proposed Level 2 instead of Level 3 following implementation of the proposed logic revisions. If level in the RPV is decreasing, there will be a delay in the Secondary Containment isolation signal on level since Level 2 will occur subsequent to Level 3. It should be recognized that high pressure in the drywell (Primary Containment) is a separate and diverse signal that will also result in Secondary Containment isolation. Breaks inside primary containment with a flow rate greater than the High Pressure Coolant Injection (HPCI) System make-up capacity will result in high drywell pressure signal occurring before level drops to Level 2. For break sizes within HPCI make-up capacity, RPV water level will be restored and maintained. The setpoint for the high drywell pressure signal for Secondary Containment isolation is not being changed.

The above scenarios would occur if offsite power were available. Loss of offsite power automatically initiates Secondary Containment isolation.

The radiological consequences of the LOCA are based on an assumed fuel failure occurring as a result of a guillotine break in the Reactor Recirculation system. Lowering the RPV level setpoint for secondary containment isolation from Level 3 to Level 2 does not impact these analysis assumptions or results.

The dose calculations methodology and results are not affected as a result of changing the RPV level setpoint for secondary containment isolation from Level 3 to Level 2. The dose calculations remain bounding because of conservative assumptions that are used to perform these calculations. The design basis LOCA dose calculations assume:

- At the time of the accident the fission products (25% of all equilibrium iodine fission products and 100% of noble gas fission products) are available for release within a short time (effectively immediately) from the primary containment airspace.

- The fission products released to Secondary Containment are immediately available for release via SGTS (i.e., no credit is taken for Secondary Containment atmospheric dilution or plateout, or fission product decay).

5. NRC Request

Has the existing RWCU break detection system been graded as a safety system so that rigorous equipment qualification program is used to maintain the system?

CNS Response

The RWCU break detection system is classified as essential. (CNS uses the term "essential" for systems and components classified using the more common industry term of "safety-related"). The electrical components and wiring located inside the Reactor Building that are classified as essential, including the RWCU break detection system, are included in the scope of the CNS Environmental Qualification program. These components and wiring are maintained qualified for a harsh environment.

Correspondence Number: NLS2004116

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	