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October 8, 2013

AEP-NRC-2013-79
10 CFR 50.90
10 CFR 50.36

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
License Amendment Request Regarding Restoration of Normal Reactor Coolant System Operating Pressure and Temperature Consistent With Previously Licensed Conditions

REFERENCES:

1. Meeting Summary: "Summary of June 26, 2013, Pre-Application Meeting to Discuss Pending License Amendment Request to Modify the CNP Unit 1 Normal Operating Pressure and Temperature (TAC NO. MF1973)." ADAMS Accession Number ML13210A227.

2. Meeting Slides: June 26, 2013, Public Meeting Slides Concerning D.C. Cook Unit 1 License Amendment Request (LAR) to Return to Normal Operating Pressure and Temperature. ADAMS Accession Number ML13176A133.

3. Letter from J. P. Gebbie, Indiana Michigan Power Company (I&M), to Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1 2011 Steam Generator Tube Inspection Report," dated April 17, 2012. ADAMS Accession Number ML12124A314.

4. Letter from J. P. Gebbie, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1 2011 Steam Generator Tube Inspection Report Response to Request for Additional Information," dated August 30, 2012. ADAMS Accession Number ML12255A402.

5. Letter from T. J. Wengert, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Unit 1 - Review of the 2011 Refueling Outage Steam Generator Tube Inservice Inspection Results (TAC NO. ME8466)," dated December 20, 2012. ADAMS Accession Number ML12324A418.

6. Letter from J. P. Gebbie, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2 Response to Information Request Pursuant to 10 CFR 50.54(f) Related to the Estimated Effect on Peak Cladding Temperature Resulting from Thermal Conductivity Degradation in The Westinghouse-Furnished Realistic Emergency Core Cooling System Evaluation (TAC NO. M99899)," dated March 19, 2012. ADAMS Accession Number ML12088A104.

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7. Letter from J. P. Gebbie, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Unit 1 30-Day Report of Changes to or Errors in an Evaluation Model," dated August 30, 2013. ADAMS Accession Number ML13247A174.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, proposes to amend the Appendix A Technical Specifications (TS) to Renewed Facility Operating License No. DPR-58 and Updated Final Safety Analysis Report (UFSAR) Section 6.3.2. I&M proposes to increase normal reactor coolant system temperature and pressure, consistent with previously licensed conditions, to mitigate ongoing steam generator (SG) tube wear. Implementation is planned to occur prior to CNP Unit 1 Cycle 26 startup (October 2014); approval is therefore requested by August 2014. This proposed change was discussed with the Nuclear Regulatory Commission (NRC) Staff in a public meeting on June 26, 2013 (References 1 and 2).

Background

During the Unit 1 refueling outage in late 2011 prior to Cycle 24 operation, eddy current examinations of SG tubing showed tube wear in the vicinity of fan bar supports in the U-bend region of each SG. Based on the step change in the number of reported indications, a root cause analysis was performed in 2012 with the assistance of an independent consultant and review by a third party. The analysis indicated that the tube degradation is caused by vibration in the SG U-bend region, and that the low SG steam pressure associated with current Unit 1 Reactor Coolant System (RCS) temperature and pressure operating conditions contribute to the vibration. The prescribed corrective action to mitigate the tube wear progression was to increase the full-power secondary-side SG steam pressure from approximately 675 psig to approximately 800 psig by restoring the RCS to near its previously licensed "normal" conditions. In addition to beginning pursuit of the prescribed corrective action, I&M conservatively planned an additional inspection for the 2013 refueling outage to monitor the condition and to validate the root cause results and recommendation. Additional information on the 2011 SG inspection results for Unit 1 and associated correspondence between the NRC and I&M are found in References 3, 4, and 5.

In early 2013, the Unit 1 SG inspection during the refueling outage prior to Cycle 25 operation confirmed that tube wear at fan bar intersections is continuing, although the wear rate is low. Assessment of these inspection results also confirmed that raising the SG steam pressure is an appropriate corrective action and will help ensure long-term reliability of the Unit 1 steam generators by mitigating ongoing wear between fan bar supports and SG tubes in the U-bend region. A determination was made that the desired steam pressure can be accomplished by increasing RCS pressure from 2100 psia to 2250 psia and increasing full power average coolant temperature from 556°F to 571°F.

License Amendment Request Scope

The proposed changes to Unit 1 RCS normal operating pressure and full power average temperature to 2250 psia and 571 °F, respectively, involve revisions to Unit 1 TSs, including a revised start time for the Containment Air Recirculation/Hydrogen Skimmer (CEQ) System to support Best Estimate Loss of Coolant Accident (BELOCA) Peak Cladding Temperature (PCT) analysis; adoption of a modified set of accident analysis inputs for BELOCA PCT Thermal Conductivity Degradation (PCT-TCD) evaluation; and a revision to the UFSAR to describe the containment spray system (CTS) actuation delay time as an input to BELOCA PCT. The list of specific licensing basis changes submitted for approval with this LAR and the enclosed supporting documentation are identified in the next two subsections.

Licensing Changes

In order to restore RCS operating pressure and RCS average temperature, I&M is requesting NRC approval of proposed changes to the following:

- SR 3.4.14.1, RCS Pressure Isolation Valve (PIV) Leakage is being changed to account for the higher RCS pressure.
- SR 3.5.5.1, Seal Injection Flow is being changed to account for the higher RCS pressure.
- SR 3.6.10.1, Containment Air Recirculation/Hydrogen Skimmer (CEQ) System timing is being changed to gain accident analysis margin.
- UFSAR Section 6.3.2 is being changed to acknowledge use of CTS timing to support achievement of accident analyses acceptance criteria.

Enclosures

A description of the LAR, including No Significant Hazards Consideration Determination, is provided in Enclosure 2. Enclosures 3, 4, and 5 provide markups of Technical Specifications, Technical Specification Bases, and UFSAR pages for CTS operation, respectively. Enclosures 6 (WCAP-17762-NP, Revision 1), 7 (Programs and Systems Evaluations), and 8 (Radiological Dose Evaluation) document the technical and accident analyses reviews performed. Finally, Enclosure 9 provides the "Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break." The summary shows the anticipated changes in margin with an overall improvement in peak clad temperature. The analysis includes consideration of previous 10 CFR 50.46 evaluations for BELOCA PCT-TCD (Reference 6) and for Revised Heat Transfer Multiplier Distributions (Reference 7).

Conclusion

I&M has evaluated the proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

New clean Unit 1 TS pages with proposed changes incorporated will be provided to the NRC Licensing Project Manager when requested.

Copies of this letter and its enclosures are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

HLE/amp

Enclosures:

1. Affirmation
2. Proposed License Amendment Request Regarding Restoration of Normal Reactor Coolant System Operating Pressure and Temperature Consistent with Previously Licensed Conditions
3. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked to Show Proposed Changes
4. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Bases Pages Marked to Show Proposed Changes
5. UFSAR Section 6.3.2 Pages Marked to Show Proposed Changes
6. WCAP-17762-NP, Revision 1, "D. C. Cook Unit 1 Return to Reactor Coolant System Normal Operating Pressure/Normal Operating Temperature Program – Licensing Report," September 2013.
7. Programs and Systems Evaluations
8. Radiological Dose Evaluation
9. Expected LOCA Peak Clad Temperature Summary

c: J. T. King – MPSC
S. M. Krawec, AEP Ft. Wayne, w/o enclosures
MDEQ – RMD/RPS
NRC Resident Inspector
C. D. Pederson -- NRC Region III
T. J. Wengert – NRC Washington DC

Enclosure 1 to AEP-NRC-2013-79

AFFIRMATION

I, Joel P. Gebbie, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Joel P. Gebbie
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 8 DAY OF October, 2013

Notary Public

My Commission Expires 04-04-2018

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2018
Acting in the County of Berrien

Enclosure 2 to AEP-NRC-2013-79

Proposed License Amendment Request Regarding
Restoration of Normal Reactor Coolant System Operating
Pressure and Temperature Consistent With Previously Licensed Conditions

Enclosure 2 to AEP-NRC-2013-79

Proposed License Amendment Request Regarding Restoration of Normal Reactor Coolant System Operating Pressure and Temperature Consistent With Previously Licensed Conditions

1.0 DESCRIPTION

Donald C. Cook Nuclear Plant (CNP) Unit 1 was originally licensed at a reactor coolant system (RCS) operating pressure of 2250 psia, a nominal core inlet temperature of 536°F and a nominal core outlet temperature of 599°F (see Section 1.2 of Reference 1). The reactor vessel average temperature (T_{avg}) for CNP Unit 1 has been revised in support of a number of License Amendments, including the 30 percent Steam Generator Tube Plugging (SGTP) Program (Reference 2), wherein a range of values from 553°F - 576.3°F was used in accident analyses and an upper end T_{avg} value of 576.3°F was approved in the CNP Unit 1 Technical Specifications (TS).

As a result of steam generator (SG) tube degradation in the late 1980s, CNP Unit 1 implemented a SG preservation program, which included reducing pressurizer pressure to 2100 psia and T_{avg} to approximately 556°F. The approved Licensing Amendment (Reference 3) issued in 1989 authorized operation at either 2100 psia or 2250 psia; Unit 1 began reduced temperature and pressure operations in 1990. Subsequently, the CNP Unit 1 SGs were replaced in 1999, however, the reduced RCS operating conditions were maintained. Eventually, the high pressure option that had been retained in the Unit 1 TSs since 1989 was removed during the conversion to the Improved Standard Technical Specifications (Reference 4) to reduce the number of exceptions being taken.

During the Unit 1 refueling outage in late 2011 prior to Cycle 24 operation, eddy current examinations of SG tubing showed tube wear in the vicinity of fan bar supports in the U-bend region of each SG. Based on the step change in the number of reported indications, a root cause analysis was performed in 2012 with the assistance of an independent consultant and review by a third party. The analysis indicated that the tube degradation is caused by vibration in the SG U-bend region, and that the low SG steam pressure associated with current Unit 1 Reactor Coolant System (RCS) temperature and pressure operating conditions contribute to the vibration. The prescribed corrective action to mitigate the tube wear progression was to increase the full-power secondary-side SG steam pressure from approximately 675 psig to approximately 800 psig by restoring the RCS to near its previously licensed "normal" conditions. In addition to beginning pursuit of the prescribed corrective action, I&M conservatively planned an additional inspection for the 2013 refueling outage to monitor the condition and to validate the root cause results and recommendation.

In early 2013, the Unit 1 SG inspection during the refueling outage prior to Cycle 25 operation confirmed that tube wear at fan bar intersections is continuing, although the wear rate is low. Assessment of these inspection results also confirmed that raising the SG steam pressure is an appropriate corrective action and will help ensure long-term reliability of the Unit 1 steam generators by mitigating ongoing wear between fan bar supports and SG tubes in the U-bend region. A determination was made that the desired steam pressure can be accomplished by increasing RCS pressure from 2100 psia to 2250 psia and increasing full power average coolant temperature from 556°F to 571°F.

I&M proposes to implement a return to RCS Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) conditions for CNP Unit 1 by increasing the current operating nominal full-power pressurizer pressure from 2100 psia to 2250 psia and increasing the current operating nominal full-power T_{avg} from 556°F to 571°F. Implementation of the program is proposed to occur prior to CNP Unit 1 Cycle 26 startup (October 2014).

This enclosure includes a description of the Proposed Change, summaries of the Technical Evaluations and Radiological Dose Consequence Analyses, and the Regulatory Evaluation (i.e., No Significant Hazards Consideration). Detailed information regarding the technical and radiological dose evaluations can be found in Enclosures 6 through 8.

2.0 PROPOSED CHANGE

The proposed change revises 1) TS Surveillance Requirements that use the normal operating RCS pressure value to reflect the increased pressurizer pressure of 2250 psia, 2) Containment Air Recirculation/Hydrogen Skimmer (CEQ) fan start time acceptance criteria for consistency with the new accident analyses and evaluations documented in WCAP-17762-NP, Revision 1 (Reference 5) in Enclosure 6, and 3) the Updated Final Safety Analysis Report (UFSAR) description of the containment spray system to specifically acknowledge credit being taken for system actuation delay time as an input to Best Estimate Loss of Coolant Accident (BELOCA) Peak Cladding Temperature (PCT). The following changes are proposed by this amendment request:

- TS 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage,
  Surveillance Requirements SR 3.4.14.1 – revised to reflect the proposed change in nominal RCS pressure range (from 2065 psig ≤ RCS Pressure ≤ 2105 psig to 2215 psig ≤ RCS Pressure ≤ 2255 psig)
- TS 3.5.5, Seal Injection Flow
  Surveillance Requirements SR 3.5.5.1 – revised to reflect the proposed change in new nominal RCS pressure (from 2065 psig ≤ RCS Pressure ≤ 2105 psig to 2215 psig ≤ RCS Pressure ≤ 2255 psig)
- TS 3.6.10 Containment Air Recirculation / Hydrogen Skimmer (CEQ) System
  Surveillance Requirements SR 3.6.10.1 – revised to reflect a surveillance delay time range for the containment air recirculation fans that is consistent with the accident analyses and evaluations in Enclosure 6 (from 108 seconds ≤ CEQ fan delay ≤ 132 seconds to 270 seconds ≤ CEQ fan delay ≤ 300 seconds)
- UFSAR Section 6.3.2.
  Revised to credit containment spray system (CTS) actuation time delay consistent with the accident analyses and evaluations in Enclosure 6 of this letter

3.0 TECHNICAL EVALUATION

The proposed return to NOP/NOT conditions has involved a comprehensive review of the Unit 1 design and licensing bases, including impact considerations on accident analyses and Nuclear Steam Supply System and Balance of Plant operations and performance. WCAP-17762-NP, Revision 1 (Enclosure 6) documents the results of the evaluations and analyses performed by

Westinghouse in support of the CNP Unit 1 Return to RCS NOP/NOT Program and associated TS changes. Programs and Systems Evaluation results are documented in Enclosure 7. Finally, an impact assessment of the pressurizer nozzle weld overlays installed in 2006 was conducted. This review found that, although the stress analyses performed at the time used currently licensed normal operating RCS temperatures and pressures as inputs, the increased NOP/NOT RCS temperature and pressure can be accommodated within the ASME Code Allowables. Formal revisions to the applicable nozzle stress analyses to incorporate NOP/NOT temperatures and pressures are scheduled for completion by July 1, 2014, and will be tracked to closure by the formal Engineering Change Package prepared for NOP/NOT implementation.

The documented analyses and evaluation results, a summary of which are provided in the subsections that follow, demonstrate that CNP Unit 1 can be safely returned to NOP/NOT conditions, consistent with the licensing basis defined in this License Amendment Request.

3.1 Westinghouse Analyses

As noted in Section 3.0, Westinghouse performed analyses and evaluations to assess the impacts of Unit 1's Return to RCS NOP/NOT conditions on the accident analyses defined in applicable sections of Chapter 14 of the CNP UFSAR. The principal inputs used by Westinghouse and documented in WCAP-17762-NP, Revision 1 (Enclosure 6 of this letter) are provided in the bulleted list following this paragraph. Specific assumptions, acceptance criteria, descriptions of analyses/evaluations, results, and conclusions are presented in the appropriate sections of WCAP-17762-NP, Revision 1. Evaluations and analyses were performed using currently approved analytical techniques/accepted approaches to demonstrate compliance with the licensing criteria and standards applicable to CNP Unit 1. Results of WCAP-17762-NP, Revision 1 support the regulatory evaluation contained in Section 5.0 of this enclosure.

- Nuclear steam supply system power of 3327 MWt, which includes a reactor coolant pump net heat input of 12 MWt
- Nominal reactor coolant pressure of 2250 psia
- Reactor vessel T_{avg} range of 553.7°F to 575.4°F
- Nominal feedwater temperature of 437.4°F
- Babcock & Wilcox International Series 51 replacement steam generators
- Westinghouse 15x15 Upgrade Fuel
- SGTP levels of 0 and 30 percent

3.2 Programs and Systems Evaluations

A review of Programs and Systems potentially affected by the return to NOP/NOT was performed and is documented in Enclosure 7 of this letter. Evaluated Programs and Systems were either determined to be unaffected by the requested increase in RCS pressure and temperature or found to be bounded by previously licensed operating conditions. This review concluded that there are no significant adverse effects on Programs or Systems as a result of the proposed increase in normal RCS operating pressure and temperature.

4.0 RADIOLOGICAL DOSE CONSEQUENCE ANALYSES

Offsite radiological dose consequence analyses for CNP are based on the Technical Information Document (TID)-14844 (Reference 6) methodology described in Regulatory Guide 1.195 (Reference 7). Control room habitability dose consequence analyses utilize the Alternative Source Term methodology of 10 CFR 50.67 (Reference 8) and Regulatory Guide 1.183 (Reference 9). Both offsite and control room habitability radiological dose consequence analyses were assessed for potential impact from the proposed increase in RCS operating pressure and temperature for Unit 1 and associated changes. The review is documented in Enclosure 8 of this letter.

As described in Enclosure 8 of this letter, the proposed changes to RCS normal operating pressure and temperature do not involve more than a minimal increase in consequences to the offsite and control room habitability radiological dose consequence analyses. Further, the Acceptance Criteria from 10 CFR 50.67 (control room habitability) and Regulatory Guide 1.195 (offsite) will continue to be met when considering the proposed input parameter changes described in Section 2.0 of this Enclosure.

5.0 REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements

10 CFR 50.36 (c)(3), Surveillance Requirements: Surveillance requirements relate to testing, calibration, or inspections that assure that the necessary quality of systems and components are maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

5.2 Precedents

No precedents were identified where a licensee reduced RCS pressure and temperature and subsequently restored them.

5.3 No Significant Hazards Consideration

I&M is requesting an amendment to the CNP Unit 1 Renewed Facility Operating License No. DPR-58 to revise the following Technical Specifications and UFSAR Section:

- SR 3.4.14.1 (TS 3.4.14, RCS PIV Leakage – Surveillance Requirements)

  The RCS Pressure Isolation Valve (PIV) RCS pressure range (TS SR 3.4.14.1 and associated Bases Limited Conditions for Operations) is revised based on the program parameters. All of the acceptance criteria for the accident analysis continue to be met with the revised pressures.

- SR 3.5.5.1 (TS 3.5.5, Seal Injection Flow – Surveillance Requirements)

  The pressurizer pressure range (TS SR 3.5.5.1) is revised and the low pressure operation (Bases) is eliminated based on the program parameters. All of the acceptance criteria for the accident analysis continue to be met with the revised pressure.

- SR 3.6.10.1 (TS 3.6.10, Containment Air Recirculation / Hydrogen Skimmer (CEQ) System – Surveillance Requirements)

  The minimum and maximum delay times for the containment air recirculation fans (TS SR 3.6.10.1) are revised based on the program parameters. In association with this change, the related Bases (Background, Applicable Safety Analyses, and SR) have updated minimum and maximum start times where applicable. All of the acceptance criteria for the accident analysis continue to be met with the revised delay and/or start times.

- UFSAR Section 6.3.2, Containment Spray Systems, System Design

  The CTS design at CNP includes a time delay relay in the CTS pump start circuitry that is presently used to properly sequence the pump onto the Emergency Diesel Generator (EDG) bus and prevent overloading of the diesel. To offset the adverse effects of the proposed increase in full power average RCS temperature on BELOCA-PCT, the setting of this time delay relay is increased to further delay CTS actuation following a Hi-Hi containment pressure signal. The new time delay setting continues to support proper EDG bus loading, but also results in a higher containment pressure during large break loss of coolant accident RCS blowdown, which limits the rate of RCS mass release to containment and improves BELOCA-PCT results. Accident analyses results accounting for the new time delay setting are described in Enclosure 6 of this letter. This License Amendment Request includes NRC review and approval of proposed changes to UFSAR Section 6.3.2, which describe use of the CTS pump time delay relay to affect BELOCA-PCT results.

5.4 No Significant Hazards Consideration Determination

I&M has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

- SR 3.4.14.1 RCS PIV Leakage – Surveillance Requirements

  The proposed change to the RCS PIV RCS pressure range does not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR. The analytical and evaluation efforts performed for the NOP/NOT conditions were shown to be acceptable. The systems and components (including interface systems and control systems) will function as designed and all performance requirements for these systems remain acceptable. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. No significant increase in the consequences has been identified. The NOP/NOT conditions do not introduce the possibility of a change in the frequency of an accident because the parameter changes are not an initiator of any accident previously considered and no new failure modes have been introduced. Therefore, neither the probability nor the consequences of an accident previously evaluated has been significantly increased.

- SR 3.5.5.1 Seal Injection Flow – Surveillance Requirements

  The proposed change to the pressurizer pressure range and the elimination of the low pressure operation does not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR. The analytical and evaluation efforts performed for the NOP/NOT conditions were shown to be acceptable. The systems and components (including interface systems and control systems) will function as designed and all performance requirements for these systems remain acceptable. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. No significant increase in the consequences has been identified. The NOP/NOT conditions do not introduce the possibility of a change in the frequency of an accident because the parameter changes are not an initiator of any accident previously considered and no new failure modes have been introduced. Therefore, neither the probability nor the consequences of an accident previously evaluated has been significantly increased.

- SR 3.6.10.1 Containment Air Recirculation / Hydrogen Skimmer (CEQ) System – Surveillance Requirements

  The proposed change to the containment air recirculation fan delay/start times does not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR. The analytical and evaluation efforts performed for the NOP/NOT conditions were shown to be acceptable. The systems and components (including interface systems and control systems) will function as designed and all performance requirements for these systems remain acceptable. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. No significant increase in the consequences has been identified. The NOP/NOT conditions do not introduce the possibility of a change in the frequency of an accident because the parameter changes are not an initiator of any accident previously considered and no new failure modes have been introduced. Therefore, neither the probability nor the consequences of an accident previously evaluated has been significantly increased.

- UFSAR Section 6.3.2, Containment Spray Systems, System Design

  The proposed revision to UFSAR Section 6.3.2 specifically recognizes use of the CTS pump time delay relay in mitigating the consequences of postulated accidents. Previously, the setting of this relay was established to support proper EDG bus loading and it was accounted for as an input to accident analyses. Use of the time delay relay setting to mitigate the consequences of an accident does not significantly increase the probability or consequences of an accident previously evaluated in the UFSAR. The analytical and evaluation efforts performed for the NOP/NOT conditions were shown to be acceptable. The systems and components (including interface systems and control systems) will function as designed and all performance requirements for these systems remain acceptable. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. No significant increase in the consequences has been identified. The NOP/NOT conditions do not introduce the possibility of a change in the frequency of an accident because the parameter changes are not an initiator of any accident previously considered and no new failure modes have been introduced. Therefore, neither the probability nor the consequences of an accident previously evaluated has been significantly increased.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   - SR 3.4.14.1 RCS PIV Leakage – Surveillance Requirements

     The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. This proposed change has no adverse effects on any safety related system and does not challenge the performance or integrity of any safety related system. The specified RCS pressure functions support meeting the accident analyses criteria. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

   - SR 3.5.5.1 Seal Injection Flow – Surveillance Requirements

     The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. This proposed change has no adverse effects on any safety related system and does not challenge the performance or integrity of any safety related system. The specified pressurizer pressure range supports meeting all of the accident analyses criteria. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

- SR 3.6.10.1 Containment Air Recirculation / Hydrogen Skimmer (CEQ) System – Surveillance Requirements

  The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. This proposed change has no adverse effects on any safety related system and does not challenge the performance or integrity of any safety related system. The delay/start time functions support meeting all of the accident analyses criteria. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

- UFSAR Section 6.3.2, Containment Spray Systems, System Design
  The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR because this change simply recognizes potential use of the existing CTS pump time delay relay setting to mitigate the consequences of an accident. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change. This proposed change has no adverse effects on any safety related system and does not challenge the performance or integrity of any safety related system. The delay/start time functions support meeting all of the accident analyses criteria. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

- SR 3.4.14.1 RCS PIV Leakage – Surveillance Requirements

  The proposed change does not involve a significant reduction in a margin of safety. Analyses and evaluations supporting the Return to NOP/NOT Program conditions demonstrate that all acceptance criteria continue to be met. There are no changes to the design, material, and construction standards that are applicable to any System, Structure, or Component (SSC). There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. Also, there is no change to a Design Basis Limit for Fission Product Barriers (DBLFPB). Therefore, the proposed change does not involve a significant reduction in margin of safety.

- SR 3.5.5.1 Seal Injection Flow – Surveillance Requirements

  The proposed change does not involve a significant reduction in a margin of safety. Analyses and evaluations supporting the Return to NOP/NOT Program demonstrate that all acceptance criteria continue to be met. There are no changes to the design, material, and construction standards that are applicable to any SSC. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. Also, there is no change to a DBLFPB. Therefore, the proposed change does not involve a significant reduction in margin of safety.

- SR 3.6.10.1 Containment Air Recirculation / Hydrogen Skimmer (CEQ) System – Surveillance Requirements

  The proposed change does not involve a significant reduction in a margin of safety. Analyses and evaluations supporting the Return to NOP/NOT Program conditions demonstrate that all acceptance criteria continue to be met. There are no changes to the design, material, and construction standards that are applicable to the CEQ System. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. Also, there is no change to a DBLFPB. Therefore, the proposed changes do not involve a significant reduction in margin of safety.

- UFSAR Section 6.3.2, Containment Spray Systems, System Design

  The proposed change does not involve a significant reduction in a margin of safety. There are no changes to the design, material, and construction standards that are applicable to the Containment Spray System. There are no physical changes being made to the fuel cladding, the RCS pressure boundary, or the containment. Also, there is no change to a DBLFPB. Therefore, the proposed changes do not involve a significant reduction in margin of safety.

5.5 Conclusion

Based on the above information and on the analyses and evaluations performed to support the return to NOP/NOT Program conditions, these proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NRC Safety Evaluation Report, "Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company, Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316," September 11, 1973.

2. "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 - Issuance of Amendments Re: Increased Steam Generator Plugging Limit (TAC NOS. M92587 and M92588)," March 13, 1997.

3. Letter from Jack Donohew, NRC, to M. K. Nazar, I&M, "D.C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments for the Conversion to the Improved Technical Specifications with Beyond Scope Issues (TAC NOS. MC2629, MC2630, MC2653 through MC2687, MC2690 through MC2695, MC3152 through MC3157, MC3432 through MC3453), dated June 1, 2005. ADAMS Accession Number ML050620034.

4. Letter from J. F. Stang, NRC, to M. P. Alexich, I&M, "Amendment No. 126 to Facility Operating License No. DPR-58 (TAC NO. 71062)," dated June 9, 1989. ADAMS Accession Number ML021050051.

5. WCAP-17762-NP, Revision 1, "D. C. Cook Unit 1 Return to Reactor Coolant System Normal Operating Pressure/Normal Operating Temperature Program – Licensing Report," September 2013.

6. J.J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, United States Atomic Energy Commission, March 1962.

7. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.

8. 10 CFR 50.67, "Accident Source Term," December 1999.

9. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

Enclosure 3 to AEP-NRC-2013-79

Donald C. Cook Nuclear Plant Unit 1
Technical Specification Pages Marked To Show Proposed Changes

SR 3.4.14.1
SR 3.5.5.1
SR 3.6.10.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. RHR System interlock function inoperable.	C.1	Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-----NOTE----- Only required to be performed in MODES 1 and 2. ---------- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ ~~2065~~ 2215 psig and ≤ ~~2105~~ 2255 psig.	In accordance with the Inservice Testing Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow resistance shall be ≥ 0.227 ft/gpm^2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow resistance not within limit.	A.1 Restore seal injection flow resistance to within limit.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.5.1	-------------------------------NOTE------------------------------ Not required to be performed until 4 hours after the pressurizer pressure stabilizes at ≥ ~~2065~~ 2215 psig and ≤ ~~2105~~ 2255 psig. --- Verify seal injection flow resistance is ≥ 0.227 ft/gpm^2.	31 days

3.6 CONTAINMENT SYSTEMS

3.6.10 Containment Air Recirculation/Hydrogen Skimmer (CEQ) System

LCO 3.6.10 Two CEQ trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CEQ train inoperable.	A.1 Restore CEQ train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	-----NOTE----- Only required to be met in MODES 1, 2, and 3. ---------- Verify each CEQ System fan starts on an actual or simulated actuation signal, after a delay of ≥ ~~108~~ 270 seconds and ≤ ~~132~~ 300 seconds, and operates for ≥ 15 minutes.	92 days

Enclosure 4 to AEP-NRC-2013-79

Donald C. Cook Nuclear Plant Unit 1
Technical Specification Bases Pages Marked To Show Proposed Changes

B 3.4.14
B 3.5.5
B 3.6.6
B 3.6.10

BASES

BACKGROUND (continued)

PIVs required by the LCO are provided to isolate the RCS from the Residual Heat Removal (RHR) System.

The PIVs required by this LCO are listed in the Technical Requirements Manual (Ref. 6).

Violation of the PIV leakage limit could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

Two motor operated valves are included in series in the suction piping of the RHR System to isolate the high pressure RCS from the low pressure piping of the RHR System when the RCS pressure is above the design pressure of the RHR System piping and components. Ensuring the RHR interlock that prevents the valves from being opened is OPERABLE ensures that RCS pressure will not pressurize the RHR System beyond its design pressure of 600 psig.

APPLICABLE SAFETY ANALYSES

Reference 3 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

RCS PIV Leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. The PIVs required by this LCO are listed in the Technical Requirements Manual (Ref. 6). Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm at an RCS pressure ≥ ~~2065~~ 2215 psig and ≤ ~~2105~~ 2255 psig. This criteria is based on a study by the Idaho National Engineering Laboratory (Ref. 7).

Reference 8 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure

BASES

SURVEILLANCE REQUIREMENTS (continued)

The seal injection flow resistance, R_{SL}, is determined from the following expression:

$R_{SL} = 2.31(P_{CHP}-P_{SI})/Q^2$

where:

P_{CHP} = charging pump header pressure (psig);
P_{SI} = ~~2148 psig (low pressure operation) or~~ 2300 psig (high pressure operation); and
Q = total seal injection flow (gpm).

The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

As noted, the Surveillance is not required to be performed until 4 hours after the pressurizer pressure has stabilized within a ± 20 psig range of normal operating pressure. The pressurizer pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The pressurizer pressure indications are averaged to determine whether the appropriate pressure has been achieved. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. UFSAR, Section 14.3.1.
2. UFSAR, Section 14.3.2.
3. UFSAR, Section 14.2.4.
4. UFSAR, Section 14.2.5.

BASES

BACKGROUND (continued)

This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to the Containment Spray System are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the CEQ System being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure of 11.75 psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature of 324.7°F results from the SLB analysis and was calculated to exceed the containment design temperature or a short time during the DBA SLB. The basis of the containment design temperature, however, is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3). Thermal analyses showed that the time interval during which the containment atmosphere temperature exceed the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the Containment Pressure - High High signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time ~~of 115 seconds~~ includes signal delay, diesel generator startup, and system startup time.

(after ESFAS setpoint), which is assumed in the LOCA and SLB Containment Integrity Analysis,

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the

Evaluations performed for LOCA and SLB Containment Integrity Analysis support a maximum containment spray total response time of 315 seconds.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Containment Air Recirculation/Hydrogen Skimmer (CEQ) System

BASES

BACKGROUND

The CEQ System is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a Design Basis Accident (DBA). The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting post accident pressure and temperature in containment to less than design values. Limiting pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The CEQ System provides post accident hydrogen mixing in selected areas of containment. The hydrogen skimmer portion of the CEQ System consists of two hydrogen skimmer headers routed to potential hydrogen pockets in containment, terminating on the suction side of either of the two CEQ System fans at the header isolation valves. The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

The CEQ System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, dampers, two upper compartment headers, and a hydrogen skimmer header with an isolation valve. Each train is powered from a separate Engineered Safety Features (ESF) bus.

4.5 - 5.0

The CEQ fans are automatically started by the Containment Pressure - High signal in ~~approximately 2~~ minutes after the containment pressure reaches the pressure setpoint. This also supports the required ice melt during a small break loss of coolant accident (LOCA) to ensure adequate containment recirculation sump inventory for initiation of the recirculation mode. The hydrogen skimmer header isolation valve opens when the CEQ System fan starts. The time delay ensures that the core reflood time assumed in the LOCA peak clad temperature analysis is met.

After starting, the fans displace air from the upper compartment to the lower compartment, thereby returning the air that was displaced by the high energy line break blowdown from the lower compartment and equalizing pressures throughout containment. After discharge into the lower compartment, air flows with steam produced by residual heat through the ice condenser doors into the ice condenser compartment where the steam portion of the flow is condensed. The air flow returns to the upper compartment through the top deck doors in the upper portion of the ice condenser compartment. The CEQ System fans operate

BASES

BACKGROUND (continued)

continuously after actuation, circulating air through the containment volume and purging all potential hydrogen pockets in containment.

The CEQ System also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the Containment Spray System can cool it.

The CEQ System is designed to ensure that the heat removal capability required during the post accident period can be attained. The operation of the CEQ System, in conjunction with the ice bed, the Containment Spray System, and the Residual Heat Removal (RHR) System spray, provides the required heat removal capability to limit post accident conditions to less than the containment design values.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment temperature and pressure are the LOCA and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train each of the Containment Spray System, RHR System, and CEQ System being inoperable (Ref. 1). The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis and is calculated to be less than the containment design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures, in accordance with 10 CFR 50, Appendix K (Ref. 2).

The modeled CEQ System actuation from the containment analysis is based upon a response time associated with exceeding the Containment Pressure - High Function setpoint to achieving full CEQ System air flow. The response time band ensures that containment temperature and pressure profiles are as assumed in the overall accident analyses (i.e., containment structural response and peak clad temperature analyses). The CEQ System total response time ~~of 132 seconds~~ includes the built in signal delay.

(after ESFAS setpoint), which is assumed in the LOCA Containment Integrity Analysis,

and diesel generator startup due to LOOP

Evaluations performed for LOCA Containment Integrity Analysis support a maximum CEQ total response time of 300 seconds.

The CEQ System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

In the event of a DBA, one train of the CEQ System is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses. To ensure this requirement is met, two trains of the CEQ System must be OPERABLE. This will ensure that at least one train will operate, assuming the worst case single failure occurs.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the CEQ System. Therefore, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the CEQ System is not required to be OPERABLE in these MODES.

ACTIONS

A.1

If one of the trains of the CEQ System is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the flow and hydrogen skimming needs after an accident. The 72 hour Completion Time was developed taking into account the redundant flow and hydrogen skimming capability of the OPERABLE CEQ System train and the low probability of a DBA occurring in this period.

B.1 and B.2

If the CEQ System train cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Verifying that each CEQ System fan starts on an actual or simulated actuation signal, after a delay ≥ ~~108~~ 270 seconds and ≤ ~~132~~ 300 seconds, and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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This SR has been modified by a Note that states that this Surveillance is only required to be met in MODES 1, 2, and 3. This allowance is

300 necessary since the specified delay (i.e., ≥ ~~108~~ seconds and ≤ ~~132~~ seconds) is only applicable to the automatic actuation signal (i.e., Containment Pressure - High), which is only required to be OPERABLE in MODES 1, 2, and 3. In addition, LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," requires the CEQ System Manual Initiation Function to be OPERABLE in MODE 4 and requires the performance of a TADOT every 24 months. This requirement will ensure the Manual Initiation Function can actuate the required equipment in MODE 4.

SR 3.6.10.2

Verifying, with the return air fan discharge backdraft damper locked closed and the fan motor energized, the static pressure between the fan discharge and the backdraft damper is ≥ 4.0 inches water gauge confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such tests confirm component OPERABILITY and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flow path will exist when the fan is started. By applying the correct counterweight, the damper operation can be confirmed. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

SR 3.6.10.4

Verifying the OPERABILITY of the motor operated valve in the hydrogen skimmer header provides assurance that the proper flow path will exist when the valve receives an actuation signal. The 92 day Frequency was developed considering the known reliability of the motor operated valves and controls and the two train redundancy available. Operating experience has also shown this Frequency to be acceptable.

Enclosure 5 to AEP-NRC-2013-79

UFSAR Section 6.3.2 Pages Marked To Show Proposed Changes

Containment Spray Systems, System Design

6.3 CONTAINMENT SPRAY SYSTEMS

6.3.1 APPLICATION OF DESIGN CRITERIA

The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident to prevent containment pressure from exceeding the design value. The design of the Containment Spray System is based on the conservative assumption that the core residual heat is released to the containment as steam. The heat removal capability of each Containment Spray System is sized to remove the reactor residual heat during cool down following a loss-of-coolant accident from operation at a calculated power level of 102% of 3413 MWt (3481 MWt, which bounds the MUR power uprate on either unit). The residual heat (during ice melt) plus an undefined energy margin of 50 x 10^6 BTU is absorbed by the operation of the Containment Spray System and Ice Condenser, respectively. The sizing of the Containment Spray Systems also provides for absorption of steam leaking through the operating deck at the maximum long term deck differential pressure (1/2 to 1 lb per square foot, the pressure required to open the Ice Condenser doors). Refer to Chapter 14.3 for Containment Integrity Analysis including Containment Spray System Modeling.

The secondary purpose of the Containment Spray System is the removal of fission products (radioactive iodine isotopes) from the containment atmosphere. The Containment Spray System is designed to deliver sufficient sodium hydroxide solution which, when mixed with water from the Refueling Water Storage Tank which contains approximately 1.5% by weight boric acid (2400 to 2600 ppm Boron), accumulator water, reactor coolant system water and the melted ice, results in the solution recirculated within containment after a LOCA having a pH in the range of 7.0 to 10.0. The performance of the Containment Spray System for iodine removal with a single Containment Spray Pump operating adequately fulfills the requirement of 10 CFR 100 as described in Chapter 14.

The Containment Spray Pumps (CTS) are equipped with two recirculation test loops to provide the capability of verifying full design flow of the CTS pumps. As illustrated in Figure 6.3-1, water is recirculated through the test loops by the Containment Spray pumps with a portion of the discharge being fed back to the pump suction and the remainder returned to the Refueling Water Storage Tank. Each recirculation test loop includes a flow meter to verify pump capacity during testing. The motor-operated valves in the RHR spray lines downstream of the RHR heat exchangers remain closed during testing of that portion of the RHR system which is a part of the spray system. Testing of this flow path is accomplished by a recirculation flow around the Residual Heat Removal Heat Exchanger. This portion of the test loop may also be used for mixing water in the RWST to acquire a homogeneous solution for adjusting boron concentration.

The Spray Additive System is tested periodically to demonstrate the delivery capability of concentrated sodium hydroxide (NaOH) solution from the spray additive tank (SAT). A water test line from the RWST suction to the eductor inlet is used to simulate the SAT flow. Acceptable eductor performance was derived using proportionality principles between test criteria and a prediction of actual system performance.

Test connections are provided downstream of the block valves for checking (with air) for unobstructed flow through the spray nozzles.

6.3.2 SYSTEM DESIGN

System Description

Adequate containment pressure reduction and iodine removal are provided by the Containment Spray Systems whose components operate in sequential modes as follows:

a) 'A' mode. Spraying a portion of the contents of the Refueling Water Storage Tank into the containment atmosphere using the Containment Spray Pumps. During this mode, the contents of the spray additive tank (sodium hydroxide solution) are mixed into the spray system to provide adequate iodine removal.

b) 'B' mode. Recirculation of water from the recirculation sump by the Containment Spray Pumps through Containment Spray Heat Exchangers and back to the containment after the Refueling Water Storage Tank has been isolated, but while there is still ice in the Ice Condenser. This spray reduces the containment atmosphere temperature and prolongs the effective life of the ice.

c) During the 'A' mode NaOH is metered into the spray solution by an eductor system, using the Containment Spray Pump discharge for motive water. If the Spray Additive Tank level decreases to the setpoint level during 'A' mode, the eduction of NaOH is automatically terminated. Eduction of NaOH is manually terminated early in the 'B' mode as soon as the Containment Spray Pumps have been restarted.

d) Diversion of a portion of the recirculation flow from the Residual Heat Removal System to additional redundant spray headers completes the containment spray system heat removal capability. This operation is initiated after the Ice Condenser has been depleted and in the event that containment pressure rises above a predetermined limit.

The Containment Spray System arrangement is shown in Figure 6.3-1. The Containment Spray System consists of two full-size (maximum heat-removal capability) redundant trains. Each train consists of the following:

a) A Containment Spray Pump, a Containment Spray Heat Exchanger, valves, piping, necessary instrumentation and controls and spray headers in both the upper and lower containment volumes. The flow of this train provides 2,000 GPM to the upper volume, 1,000 GPM to the lower volume and 200 GPM total to the two fan rooms in the lower volume outer annulus.

b) A Residual Heat Removal Pump, Residual Heat Removal Heat Exchanger, piping, valves, necessary controls and instrumentation and an individual spray header in the upper containment volume with a capacity of at least 1890 gpm.

For the iodine removal function, NaOH is added to the suction of the Containment Spray Pumps by the entrainment and mixing action provided by eductors which are powered by the discharge pressure of their respective pump. During the time period that NaOH solution is added to the spray flow, a design inlet flow of 26-28 GPM (approx.) is diverted from the Containment Spray Pump discharge to serve as motive fluid for eductor operation.

The eductor draws a design suction flow of 11-63 GPM (approx.) from the spray additive tank which produces a solution in the recirculation sump suitable for iodine retention. The two eductor loops are served by a shared spray additive tank through the necessary valves and piping equipped with the necessary instrumentation. The Containment Spray System is tested periodically to demonstrate the delivery capability of the pumps and spray additive components as described in Section 6.3.1.

Containment Spray System operation is automatically initiated. Containment pressure is monitored by 4 sensors in the lower volume. The output of these sensors is the hi-hi containment pressure signals occurring at approximately 3.0 psig. These signals are fed into a safeguards logic cabinet which contains the 2/4 logic to actuate the spray (Spray Actuation). Spray Actuation starts the Containment Spray Pumps, opens the discharge valves to the spray headers and opens the valves associated with the spray additive tanks. Similarly, 1.2 psig produces the Hi Containment pressure signal, which, through the safeguards logic cabinet results in the 2/3 logic matrix required for the Safety Injection Actuation signal.

Any system failure producing a rise in containment pressure to approximately 3.0 psig will result in actuation of the Containment Spray System.

If additional spray is required during the recirculation phase, a portion of the Residual Heat Removal System flow can be manually diverted to the spray mode.

Insert Paragraph "A"

Paragraph "A":

The start time of the Containment Spray Pumps, given a Spray Actuation signal, is determined by the setting of a time delay relay in the pump start circuitry. This relay setting is used to achieve desired load sequencing of a Containment Spray Pump onto an Emergency Diesel Generator and it may also be used to support achievement of accident analyses acceptance criteria. Delaying actuation of the Containment Spray Pumps for a period of time following a containment pressure Hi-Hi signal, for example, has the effect of allowing a higher containment backpressure during the initial phases of a loss of coolant accident, which provides a benefit to the Peak Clad Temperature analysis. Specific inputs and assumptions used for Unit 1 and 2 accident analyses are described in the respective sections of Chapter 14.

The portions of the Containment Spray System located outside of the containment which are designed to circulate radioactively contaminated water collected in the recirculation sump as the result of an accident meet the following requirements:

a) Radiation exposure considerably below 10 CFR 100 limits.
b) Means to restrict all system leakage (e.g., from pumps, seals, valve stems, etc.).
c) Means for isolation of any section under malfunction or failure conditions.
d) Means to detect and control radioactive leakage.
e) Provision of a leak detection system as described in Section 6.1.

Each of the spray trains provides complete backup for the other. The passive portions of the Containment Spray System located within the containment are designed to withstand the post-accident containment environment without loss of performance and to operate without maintenance. All active components of the Containment Spray System are located outside the containment, hence are not required to operate in the steam-air environment produced by an accident. All spray headers and supply pipes are missile shielded or are designed with enough separation so that system operation cannot be significantly impaired by any segment thereof being rendered inoperable by a missile. The spray headers located in the upper containment volume are separated from the reactor and reactor coolant loops by the operating deck and inner wall of the ice bed. These spray headers are therefore protected from missiles originating in lower containment. The spray headers for the lower containment spray nozzles are also protected from missiles originating in lower containment with the exception of the small feeder lines that serve no more than four spray nozzles each. A special leak rate limit has been established for the containment isolation valves in the containment spray supply headers. The limit ensures that the inventory of spray water resident in the containment spray headers inside the containment will not be depleted by leakage through the isolation check valve to a level which would expose the containment isolation valves to the post-LOCA containment environment, for a minimum of thirty days, in the event that a spray system is shut down.

The feeder lines are separated from each other so that they cannot be damaged by the same missile. A design criterion of the feeder lines is to control resistance in the piping system to the nozzle, such that if one of the feeder lines is severed by a missile the effect on the system capability will be negligible.

Hydraulic analyses of the feeder line design indicates that flow increase due to an open-ended feeder line is approximately 5 gpm, which is less than 0.2% of the flow from a single Containment Spray Pump.

Components

The Containment Spray System water is supplied from the Refueling Water Storage Tank during the injection phase and the Recirculation Sump during the recirculation phase.