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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

June 21, 2004

L-MT-04-035
10 CFR 50.90

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Response to Request for Additional Information Regarding License Amendment Request for Technical Specification Table 3.2.1 and 3.2.4

Reference 1) NMC letter to NRC, "License Amendment Request For Technical Specification Tables 3.2.1 and 3.2.4," dated December 23, 2003

Reference 2) NRC letter to NMC, "Monticello Nuclear Generating Plant – Request for Additional Information Related to License Amendment Request For Technical Specification Tables 3.2.1 and 3.2.4 (TAC No. MC1804)," dated May 13, 2004

In Reference 1, Nuclear Management Company, LLC (NMC) requested the Nuclear Regulatory Commission (NRC) to modify the Monticello Nuclear Generating Plant (MNGP) Technical Specifications to eliminate the Reactor Head Cooling containment isolation function, and to correct and clarify existing requirements, make wording enhancements, and revise an existing limiting condition for operation (LCO) for radiation monitors used to isolate Reactor Building Ventilation and initiate the Standby Gas Treatment System.

In Reference 2, the NRC requested that additional information be provided to support the exemption requested in Reference 1.

Enclosures 1 and 2 contain NMC's response to the requested information by the NRC. Enclosure 3 contains proposed marked-up Technical Specification (TS) Changes. Enclosure 4 contains proposed final TS changes.

The above referenced TS changes are administrative and editorial in nature and have no technical impact on pages or justifications previously submitted. These changes to Monticello TS are needed to correctly reflect the intent of the original license amendment request.

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The original changes were evaluated in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and were determined not to involve any significant hazards consideration. The attached information does not impact that determination; therefore, the Determination of No Significant Hazards Consideration submitted by the original letter dated December 23, 2003 (Reference 1), is also applicable to this submittal.

Additionally, the original changes were evaluated and determined to meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9) and pursuant to 10 CFR 51.22(b) an Environmental Assessment was not required. The attached information does not impact that determination; therefore, the Environmental Assessment submitted by the original letter dated December 23, 2003 (Reference 1), is also applicable to this submittal.

This letter makes no new commitments or changes to any existing commitments.

If you have any questions please contact John Fields, Senior Regulatory Affairs Engineer (763-295-1663).

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Power Plant
Nuclear Management Company, LLC

Enclosures (4)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC

ENCLOSURE 1

NMC RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

NRC Request #1:

The regulation at 10 CFR 50.36(c)(2)(ii) specifies that a TS Limiting Condition for Operation (LCO) must be established for each item meeting one or more of four criteria. Generic Letter 95-10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation," uses the same criteria for relocating any item from the TSs. Justify removing the reactor head cooling containment isolation function from the TSs based on the four criteria in 10 CFR 50.36(c)(2)(ii).

NMC Response:

Provided below is the justification for removal of the reactor head cooling line containment isolation function. The justification compares the criteria of 10CFR50.36(c)(2)(ii) with the reactor head cooling line isolation function:

Criteria 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The head cooling line does not communicate with the reactor coolant pressure boundary, therefore, this criteria is not applicable. As identified in section 3.1 of Reference 1, the head cooling line spool piece was removed from the vessel via a plant modification and the vessel nozzle flanged to prevent hydrogen gas build up and thus relieve the detonation potential described in GE SIL 643, "Potential for Radiolytic Gas Detonation." This was considered Part A of the modification, which has been completed. Part B of the modification will remove the containment isolation valves associated with the head cooling line and cut and cap the penetration. See NMC Response under NRC Request #2 for further details on the plant modification. No change has been made to any instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Therefore, criteria 1 does not apply.

Criteria 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criteria 3. A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criteria 4. A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The reactor head cooling line penetration of primary containment is labeled penetration X-17. Penetration X-17 meets 10CFR50.36(c)(2)(ii) Criteria 2 in its current configuration, since the reactor head cooling line containment isolation valves function as a design feature that is an initial condition of a design bases accident or transient analysis that ensures the integrity of a fission product barrier (primary containment). The penetration X-17 containment isolation valves are designed to close automatically upon receipt of a Group 2 isolation signal and are included in Monticello Technical Specification (TS) Table 3.2.1.

Penetration X-17 meets Criteria 3 in its current configuration, since the containment isolation valves function to mitigate a design basis accident or transient by ensuring the integrity of a fission product barrier (primary containment). The penetration X-17 containment isolation valves are closed automatically upon receipt of a Group 2 isolation signal.

Penetration X-17 meets Criteria 4 in its current configuration, since the containment isolation valves' function is risk significant to public health and safety (primary containment isolation). The penetration X-17 containment isolation valves are closed automatically upon receipt of a Group 2 isolation signal.

As identified in Section 4.1 of Reference 1, the purpose of this license amendment request is to remove the reactor head cooling line primary containment isolation valve function from the TS. To do this the reactor head cooling line primary containment isolation valves must be removed and the penetration process pipe welded closed to ensure the integrity of the fission product barrier. No changes to the development of the Group 2 isolation signal will be performed. Penetration X-17 will become a spare containment penetration that meets all the design criteria as the containment and will continue to be leak rate tested as indicated in USAR section 5.2.4.2.

Penetration X-17 will continue to be controlled by the MNGP TSs, since the definition of Primary Containment Integrity must still be met. However, control of this penetration will be relocated to TS section 3.7.A.2 and the leakage rate testing requirements of 4.7.A.2. These sections of the TS are provided below.

The Monticello TS define Primary Containment Integrity as follows:

"Primary Containment Integrity - Primary Containment Integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied.

1. *All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.*
2. *At least one door in the airlock is closed and sealed.*

3. *All automatic containment isolation valves are operable or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed*
4. *All blind flanges and manways are closed."*

TS section 3.7.A.2 states:

"Primary Containment Integrity
a. (1) Primary Containment Integrity as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.A.2.a.(2) or 3.7.A.2.a.(3)."

And TS 4.7.A.2 states:

"Primary Containment Integrity
a. Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program."

NMC must continue to meet the definition of Primary Containment Integrity to satisfy TS 3.7.A.2. Therefore, removal of the containment isolation valves associated with penetration X-17 does not present a challenge to the integrity of the primary containment. Therefore, upon completion of the modification, criteria 2, 3 and 4 will still apply, but will be implemented through a different TS section.

10CFR50.36 establishes criteria to apply to structures, systems, or components for creating limiting conditions for operation if the specified criteria are met. As demonstrated above, the criteria for inclusion of head cooling in TS Table 3.2.1 will no longer apply after removal of the head spray containment isolation valves and welding of a cap over the penetration process pipe. However, the MNGP TS will still control the penetration by requiring that Primary Containment Integrity be maintained through TS section 3.7.A.2 and the leak testing requirements of TS section 4.7.A.2

NRC Request #2:

Has the NRC staff reviewed NMC's plant modification to the reactor head cooling system in response to General Electric's Service Information Letter 643,"Potential for Radiolytic Gas Detonation"? If so, identify where the NRC staff reviewed and accepted the modification. If not, provide NMC's 10 CFR 50.59 evaluation for the modification.

NMC Response:

No, the NRC staff has not reviewed the NMC modification to the reactor head cooling system.

The purpose of the modification is to remove the RHR reactor head cooling piping. The removal of the head cooling piping will be completed in two parts (A & B). Part A has been completed. Part B cannot be completed until approval of a Technical Specification change.

Part A removed the reactor head cooling spool piece that connected the reactor head cooling line to the vessel head. The modification then installed a blind flange on the vessel head. Part A retained MO-2026 and MO-2027 as primary containment isolation valves. NMC Standard 10 CFR 50.59 Screening Form No. SCR-03-075, Rev. 01 is provided in Enclosure 2.

Part B of the modification will remove the remaining components. The final intention is to weld a cap on the process pipe end of primary containment penetration X-17, which will leave X-17 as a spare penetration. Since, components associated with the primary containment isolation valves are planned to be removed, this activity requires a Technical Specification change. This is identified on NMC Standard 10 CFR 50.59 Screening Form No. SCR-03-642, Rev. 00 as provided in Enclosure 2.

NRC Request #3:

Proposed TS Table 3.2.4 shows the total number of instrument channels per trip system is two for Function 3 (reactor building plenum radiation monitors) and Function 4 (refueling floor radiation monitors). However, each radiation monitor inputs to both of the trip systems. Therefore, failure of one channel will result in the failure of one channel in each trip system. Based on this, justify how GE's topical report NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation, July 1990," applies to the system since NMC uses the report as the basis for changing the LCO requirements.

NMC Response:

NEDC-31677P-A, Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation, July 1990 (Reference 2) is directly applicable to the radiation monitor configuration as described in Reference 1 section 3.2. NEDC-31677P-A, section 5.5, entitled Application of Results to Other Plants, specifically discusses the case where the number of sensors is less than the majority of other plants. The following is an excerpt from NEDC-31677P-A, section 5.5.

"There are, however, specific cases where the number of sensor variables is less than the number for the majority of plants. For example, the RWCU System is isolated for the majority of BWR-3/4 plants on either RWCU high flow, high area temperature, or high delta temperature. For four of the plants, the RWCU System is isolated on only two sensor variables. The effect of this difference in the number of sensor variables on the overall analysis results was considered to be small. This assessment is based on

the results from case studies performed for different types of instrumentation logic and number of sensor variables...It was therefore concluded that, where there is a smaller number of sensor variables which initiate a system isolation, the overall effect of the proposed STIs and AOTs on the isolation function failure frequency is acceptably small."

In addition, NEDC-31677P-A (Reference 2), Appendix C1 specifically indicates that this report applies to the Monticello plant secondary containment isolation functions performed by the radiation monitors.

The NRC SE (Reference 3), Enclosure 2 entitled "*Example of Modified Isolation Actuation Instrumentation Technical Specification*" contained marked up TS pages from Grand Gulf Unit 1. The page with approved inserts (after page 3-9) specifically adds the following:

"Place one trip system (with the most inoperable channels) in the tripped condition. The trip system need not be placed in the tripped condition when this would cause the isolation to occur."

This statement indicates that single inoperable channels were considered as an acceptable configuration and are directly applicable to the radiation monitors for MNGP.

In the Technical Evaluation Report (TER) (Enclosure 1 to the NRC SE), the following statements are made concerning failure rates and number of components within a logic channel.

"To 'envelope' the effect of the variation in failure rates and number of components within a logic channel, GE increased the sensor and relay failure rates by a factor of 3 in the 'maximum' scenario.... There is no verification supporting the fact that a factor of 3 increase in the failure rate will in fact bound the variation among individual plants within a product line. Although this approach is not consistent with the intent of performing an analysis applicable to an entire product line, we feel that this assumption is adequate for estimating base case 'maximum' unavailability values. Since the actuation logic selected for the base case is already considered to be a representative configuration, this increase in failure rates will then only add conservatism to the isolation function trip unavailability value."

In conclusion, the GE topical report (Reference 2) took into account different configurations of instrumentation, specifically where a less number of instrument channels were available and determined that this had little consequences on the analysis and was acceptable. Reference 2 also recognized the MNGP specific configuration. The NRC SE (Reference 3) endorsed TS changes recognized that not all inoperable channels could be placed in the tripped condition, because it could cause an isolation to occur, which indicates the existence of single channel instrument systems. Finally, the TER in Reference 3 endorsed GE's approach of increasing the failure rate by a factor of 3 to compensate for variations in failure rates and number of components

within a logic channel. Therefore, the MNGP configuration has been adequately considered in both the GE topical report and the NRC SE.

NRC Request #4:

Proposed TS Table 3.2.4 specifies a 6-hour allowed outage time (AOT) for surveillance testing without entering the TS actions, and a 24-hour repair time before placing an inoperable channel in the trip condition for the Function 3 and Function 4 radiation monitors. These allowances are based on the reliability analysis of NEDC-31677-P-A. What NRC staff safety evaluation report approved applying this topical report to Table 3.2.4, Functions 3 and 4?

NMC Response:

NEDC-31677P-A, (Reference 2) is an NRC approved GE Topical Report. The NRC provided a safety evaluation in a letter from Mr. Charles E. Rossi, Director US NRC to Mr. S. D. Floyd, BWR Owner's Group, entitled "General Electric Company (GE) Topical Report NEDC-31677P-A, 'Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," dated June 18, 1990 (Reference 3).

Section 5.0 and section 5.1 (applicable to Monticello) of NEDC-31677P-A discusses the acceptability of changing the channel functional testing time allowance from 2 to 6 hours and the acceptability of changing the repair allowed outage time (AOT) from 1 to 24 hours. The NRC considered this acceptable as described in section 5, entitled, Conclusions, of Enclosure 1 to the referenced letter.

NEDC-31677P-A, Table 5-1 contains the *"Rx. Bldg. Exh. Rad. High"* and *"Ref. Fl. High Radiation"* actuations, which are directly applicable to MNGP. In addition, the NRC SE (Reference 3), Enclosure 2 entitled "*Example of Modified Isolation Actuation Instrumentation Technical Specification*" contained marked up TS pages from Grand Gulf Unit 1. Page 3-21 of Enclosure 2 contains Secondary Containment Isolation Instrumentation and includes Fuel Handling Area Ventilation Exhaust Radiation – High High, and Fuel Handling Area Pool Sweep Exhaust Radiation – High High. These instruments provide an equivalent function to the Monticello Nuclear Generating Plant Refueling Floor Radiation Monitor and the Reactor Building Plenum Radiation Monitor (MNGP does not have a separate system for Fuel Handling area ventilation).

NRC Request #5:

Current Table 3.2.4, Note 5(a), as applied to Function 1 (low-low reactor water level) and Function 2 (high drywell pressure), states "With one required instrument channel inoperable <u>*per trip function*</u> (emphasis added), *place the inoperable channel or trip system in the tripped condition within 12 hours." Proposed Table 3.2.4, Note 2(a)1) revises Note 5(a) as follows: "With one instrument channel* <u>*per Trip System*</u> (emphasis

added) *inoperable: 1) For Functions 1 or 2, place the inoperable Channel or Trip System in the tripped condition within 12 hours". The current TS action applies only to one channel inoperable per function, whereas the proposed TS applies to the condition of one channel inoperable per Trip System. The difference between the current and proposed action requirements is not evaluated. Provide additional justification for the proposed change.*

NMC Response:

NMC agrees with this comment. Note (2)(a) should read: "*With one instrument Channel per Trip Function inoperable...* "The wording in the submittal is open to unintended interpretation. It can be read to require that both conditions (2)(a)(1) and (2)(a)(2) would be entered when any one instrument in any one trip system is inoperable. This was not what was intended. Therefore, the wording has been changed to the original wording. Therefore, no additional justification is required.

See Enclosure 3 for the marked up revised TS Table 3.2.4 pages and Enclosure 4 for the final revised TS Table 3.2.4 pages. These pages supercede the corresponding TS pages provided in Reference 1.

NRC Request #6:

Current Table 3.2.4, Note 4, applies to Functions 3 and 4. Note 4 says "One of the two monitors may be bypassed for maintenance and/or testing." Proposed Table 3.2.4 would delete Note 4. NMC justified this as being acceptable because the current TS reference to a single-train instrument channel is used in conjunction with two trip systems. However, the NRC staff notes that the 6-hour AOT for surveillance testing without placing a channel in the tripped condition (proposed Note 1) essentially replaces the current note with an approved topical report allowance. Provide a safety-basis justification for the proposed change.

NMC Response:

The current note (4) states: "*One of the two monitors may be bypassed for maintenance and/or testing*". This is a rather open-ended statement allowing a radiation monitor to be out of service indefinitely for either of these reasons (maintenance or testing). NMC determined that this was not the practice of the industry and elected to submit the technical specification change (Ref. 1) to remedy the condition. The suggested wording is based on the information contained in the NRC approved topical report NEDC-31677-P-A. The NRC approval of the topical report provides the safety-basis justification for imposing a more restrictive technical specification limitation on these instrument channels. See also NMC Response to NRC Request #4 for further details on the NRC approval of changing the functional testing time allowance to 6 hours.

As stated in Section 4.2(f) of Reference 1, the statement was applicable only for the "single channel" aspects of the two trip systems. Further, as stated in Section 4.2(b) of Reference 1, note (4) can be deleted because each of the radiation monitors is shared by the redundant trip systems of the function, in effect creating a configuration of two instrument channels per trip system. The monitor functions provide assurance that protective actions will be initiated in the event of high radiation levels (upscale trip). In addition, downscale trips will be annunciated (for one channel) or actuated (for two channels) in the event of monitor failure.

In summary, note (4) permitted a single radiation monitor to be left bypassed for an indeterminate amount of time for testing or maintenance. The proposed TS change (elimination of note 4) provides a defined time frame for one radiation monitor to be placed in an inoperable status for testing (Note 1 - 6 hours) or for maintenance (Note 2a2 – 24 hours). This approach is consistent with the NRC approved topical report NEDC-31677-P-A. Therefore, the proposed change is justified.

References

1. NMC letter to NRC, "License Amendment Request For Technical Specification Tables 3.2.1 and 3.2.4," dated December 23, 2003.

2. NEDC-31677P-A, Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation, July 1990.

3. NRC letter from Mr. Charles E. Rossi, Director US NRC to Mr. S. D. Floyd, BWR Owner's Group, "General Electric Company (GE) Topical Report NEDC-31677P, 'Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," dated June 18, 1990.

ENCLOSURE 2

NMC Standard 10 CFR 50.59 Screening Forms

SCR-03-075, Rev. 01
SCR-03-642, Rev. 00

14 pages follow

N04180:56

MONTICELLO NUCLEAR GENERATING PLANT	3278
TITLE: NMC STANDARD 10 CFR 50.59 SCREENING FORM	Revision 3*
	Page 1 of 7

Screening Number:[1] SCR-03-0075 rev. 01

ASSOCIATED REFERENCE(S) *02Q295, Part A* Rev: *0*
RHR Head Spray Removal

PART I: Describe the Proposed Activity and Search the UFSAR (Resource Manual 5.3.1)

1. Briefly Describe the proposed activity.

(You may refer to and/or attach to the hardcopy appropriate descriptive material .)

The purpose of this modification is to remove the RHR head spray piping. The removal of the head spray piping will be done in parts since full removal of the RHR head spray mode of the Residual Heat Removal system can't be accommodated in the outage schedule for the Cycle 22 refuel outage. Thus Part A will remove the spool piece above 1003' elevation that connects to the vessel head. Part B or a separate modification will remove the remaining components. The intention is to weld caps on the pipe ends on each side of primary containment penetration X-17, which will leave X-17 as a spare penetration. This screening includes procedures, calculations and Tech Spec basis revision.

2. Does the activity involve a change to the Technical Specifications?

Yes[2] ☑ No

3. Search the Updated Final Safety Analysis Report to identify the relevant sections. Describe the pertinent design function(s), performance requirements; and methods of evaluation of the affected SSCs and where the function(s) are described in the USAR.

(It is acceptable to attach and highlight applicable portions of the USAR.)

USAR Location	*USAR Design Function*
1.3.4	*Describes the function of the RHR system and one of the functions is for reactor shutdown cooling and head spray cooling mode of RHR to remove decay heat and sensible heat from the reactor primary system so that the reactor can be shut down for a normal refueling and service operation.*
10.2.4	*Used with shutdown cooling to cool reactor water after a reactor shutdown when reactor dome pressure is less than 75 psig. Head spray is not used to mitigate an accident.*
10.2.4.3	*Head spray is placed into operation during a normal plant cooldown when reactor dome pressure is below 75 psig.*
7.6.3.3.1	*Included in a sub-list of instrumentation and electrical equipment located within the primary containment required to mitigate the effects of a loss-of-coolant accident is RHR head spray isolation valve (isolation only).*
E.2	*No design function, E.2 references USAR 10.2.4 which is discussed above*
Figure 7.6-4	*This figure is a block diagram of primary containment isolation, which includes head spray valves as Group 2 isolation valves.*
1.2.1.9	*The function of the RHR system regarding HELB is that TW26-4"-ED (head spray) terminating at penetration X-17 and MO-2026 is a high energy line.*

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Table 14.9-4	*This table shows dose rate contributors in plant areas and head spray in the phase separator room is listed as a dose rate contributor.*
Table 5.2-3a and Table 5.2-3b	*MO-2026 and MO-2027 function as primary containment isolation valves with isolation Group 2 signals and are tested as Type C (Table 5.2-3a)*
Table I.2-1 High Energy Pipe List	*Lists pipes and pipe segments designated as high energy lines*

1 Obtain number form USAR Change Coordinator when screening is completed.

2 Changes to the Technical Specifications require a License Amendment Request.

3087 (DOCUMENT CHANGE, HOLD, AND COMMENT FORM)					
FOR ADMINISTRATIVE USE ONLY	Resp Supv: ENGR	Assoc Ref: 4 AWI-05.06.02	SR: N		Freq: 2 yrs
	ARMS: 3278	Doc Type: 3172	Admin Initials:	Date	

Part II: Determine if the Change Involves a Design Function (Resource Manual 5.3.2)

Yes	No	
✓		Does the proposed activity involve Safety Analyses or an SSC credited in the Safety Analyses?
✓		Does the proposed activity involve SSCs that support SSC(s) credited in the Safety Analyses?
	✓	Does the proposed activity involve SSCs whose failurecould initiate a transient (e.g., scram, loss of feedwater, etc.) or accident, or impact SSC(s) credited in the Safety Analyses?
✓		Does the proposed activity involve USAR-described SSCs or procedural controls that perform functions that are required by, or are otherwise necessary to comply with, regulations, license conditions, orders, or technical specifications?
	✓	Does the activity involve a METHOD OF EVALUATION described in the USAR?
	✓	Is the activity a TEST or EXPERIMENT? (i.e., a non-passive activity that collects data)
[3]	✓	Does the activity exceed or potentially affect a design basis limit for a fission product barrier (DBLFPB)?

If the answer to all of these questions is NO, skip Part III and proceed directly to Part IV.

If any of the above questions are marked YES, identify the specific Design function, method of evaluation, or DBLFPB involved. Proceed to Part III.

The primary containment isolation valves MO-2026 and MO-2027 serve as the primary containment boundary and isolate on Group 2 signals to assure containment boundary following a LOCA. This modification does not change that function. The function for reactor shutdown cooling and head spray cooling mode of RHR to remove decay heat from the reactor primary system is affected in that heat will be removed by shutdown cooling mode only. The function of shutdown cooling is to provide the capability to reach cold shutdown in 24 hours per technical specifications.

[3] This activity requires a 10 CFR 50.59 Evaluation. Skip to Part IV.

PART III: Determine Whether the Activity Involves Adverse Effects (Resource Manual 5.3.3)

Answer the following questions to determine if the activity has an adverse effect on a design function. Any YES answer in Part III means that a 10 CFR 50.59 Evaluation is required, except where noted in Question 3.

1. Changes to the Facility or Procedures

Yes	No	
	✓	Does the activity adversely affect the design function of an SSC credited in safety analyses?
	✓	Does the activity adversely affect the method of performing a design function of an SSC credited in safety analyses?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion:

The project to remove the vessel head spray mode of RHR from service does not adversely affect that design function of the RHR system to perform its design function credited in a safety analysis. The design function of primary containment isolation valves MO-2026 and MO-2027 remain the same, that is they continue to be primary containment isolation valves and isolate by Group 2 logic. Tech Spec Table 3.2.1 basis is changed to clarify that these valves provide isolation for penetration X-17 rather than head cooling. The head spray system itself does not have a design function credited in a safety analysis and is not required to mitigate a design basis accident. The RHR piping continues to meet code requirements to assure pressure integrity of the RHR system is maintained. Automated Engineering Services (AES) calculations assure the piping configuration meets code compliance. The piping will be pressure tested to comply with Section XI. USAR 10.2.4.1 contains a discussion concerning RHR Shutdown Cooling Mode capability for establishing reactor coolant temperature of 125 F in 24 hours after dome pressure is reduced to 75 psig, however, this Shutdown cooling Design Capability does not rely on Head Spray. USAR 10.2.4.2 states reactor shutdown cooling and reactor vessel head spray system is an integral part ot the RHR System and is placed in operation during a normal shutdown and cool down. Ops Manual B.03.04-05 does not allow head spray valves MO-2026 or MO-2027 to be opened until reactor temperature is less than 212 F as described in SCR-02-0659. Therefore, the head spray system does not contribute to the Technical Specification requirement to have capability to reach cold shutdown (less than 212 F) within 24 hours. Plant procedures, including EOPs will be revised to correctly represent plant configuration. EOPs include the use of head spray as a source of water, but it is not relied upon to mitigate an accident, therefore there is no adverse affect on an SSC credited in safety analyses.

USAR 10.2.4.2 continues with the fact that part of shutdown cooling may be diverted to a spray nozzle in the reactor head. This spray maintains saturated

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conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and therefore limits thermal stress in the reactor vessel during cooldown. This appears to be saying that head spray is necessary to raise water level to cool the head and limit thermal stress. It means that condensing the steam with spray allows water level to be increased sooner without increasing the pressure and that the spray does not exceed the cooldown rate that could affect thermal stress. Thermal stress is increased by spray and thus not a concern if spray is not initiated. Since head spray is not permitted (by procedure) until the vessel temperature is less than 212 F, the water to steam interface will serve to condense the steam and allow raising vessel water level to cool the mass of metal of the reactor vessel.

2. Changes to a Method of Evaluation

(If the activity does not involve a method of evaluation, these questions are **NOT APPLICABLE)**

Yes	No	
	✓	Does the activity use a revised or different method of evaluation for performing safety analyses than that described in the USAR?
	✓	Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in the USAR?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion:

The piping assiciated with removing the head spray function of RHR is analyzed to comply with USAR described methods.

3. Tests or Experiments

a. (If the activity does not involve a test or experiment,3.a and 3.b are ✓ **NOT APPLICABLE)**

Yes	No	
		Is the proposed test or experiment bounded by other tests or experiments that are described in the USAR?
		Are the SSC's affected by the proposed test or experiment isolated from the facility?

If both answers are NO, continue to 3.b. If the answer to either question is YES, then briefly describe the basis:

b. Answer these additional questions only for tests or experiments that do not meet the criteria above. If the answer to either question in 3.a is YES, then these questions are: ☐ **NOT APPLICABLE**

Yes	No	
		Does the activity utilize or control an SSC in a manner that is outside the reference of the bounds of the design basis as described in the USAR?
		Does the activity utilize or control an SSC in a manner that isinconsistent with the analyses or descriptions in the USAR?
		Does the activity place the facility in a condition not previously evaluated or that could affect the capability of an SSC to perform its intended functions?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If all answers are NO, describe the basis for the conclusion:

Part IV : Conclusion
(Resource Manual 5.3.4)

Check all that apply:

1. A 10 CFR 50.59 Evaluation is ☐[4] required or ☑[5] NOT required.

2. A USAR change is ☑[6] required or ☐ NOT required.

Additional Comments and Reviews (Additional reviews may be required by 4 AWI-05-06.02, paragraph 4.6.5):

See Ops Man C.3 step 11 for description of use of head spray.

Assign Screening number (page 1)

Prepared by: LE Thompson — Laverne Thompson — 3/26/03 ~~2/5/2003~~ Date

Reviewed by: Michael Hippe — 3/26/03 ~~2/5/2003~~ Date

[4] If it is concluded that a 50.59 Evaluation is required, it is not necessary to complete the screening

[5] If it is concluded that a 50.59 Evaluation is NOT required, assign a screening number and document the reviews as necessary.

[6] If the activity requires a USAR change, Form 3473 needs to be completed even if a 10 CFR 50.59 Evaluation is not required. List the screening number on the USAR change request.

Screening Number:[1] SCR-03-0642 rev. 00

ASSOCIATED REFERENCE(S): *02Q295 Part B* Rev: *0*
RHR Head Spray Removal

PART I: Describe the Proposed Activity and Search the UFSAR (Resource Manual 5.3.1)

1. Briefly Describe the proposed activity.

(You may refer to and/or attach to the hardcopy appropriate descriptive material .)

The purpose of this modification is to remove containment isolation valves MO-2026 and MO-2027 and weld a cap on the process pipe outboard of containment Penetration X-17, remove the abandoned head spray piping and weld a seal plate over the refuel cavity seal following removal of the refuel cavity seal bellows for the head spray pipe. MO-2026 and MO-2027 receive Group 2 isolation signals. The Group 2 logic will be changed to delete MO-2026 and MO-2027. The controls and flow indication will be removed from the control room.

This screening includes the procedures and drawings listed in Modification 02Q295 Part B, and the Technical Specification change revising Table 3.2.1 that deletes head spray from Group 2 instrumentation.

2. Does the activity involve a change to the Technical Specifications

☑ Yes[2] ☐ No

3. Search the Updated Final Safety Analysis Report to identify the relevant sections. Describe the pertinent design function(s), performance requirements, and methods of evaluation of the affected SSCs and where the function(s) are described in the USAR.

(It is acceptable to attach and highlight applicable portions of the USAR.)

USAR Location	*USAR Design Function*
10.2.4.3	*This section describes the function of containment isolation valves MO-2026 and MO-2027 as deactivated in their closed position to provide containment isolation only.*
12.2.1.10.2	*"Current Analysis of Equipment and Piping"* *Addition of new system or re-evaluation of existing systems is done using current mehtods of analysis.*
Drawings	*P&IDs and drawings associated with MO-2026, and 2027.*
Figure 7.6-4	*This figure is a block diagram of primary containment isolation , which includes Containment Penetration X-17 isolation valves associated with Group 2.*
Table 5.2-3a and Table 5.2-3b	*MO-2026 and MO-2027 function as primary containment isolation valves with isolation Group 2 signals and leak testing requirement.*

1 Obtain number form USAR Change Coordinator when screening is completed.

2 Changes to the Technical Specifications require a License Amendment Request.

3087 (DOCUMENT CHANGE, HOLD, AND COMMENT FORM)
Incorporated:

FOR ADMINISTRATIVE USE ONLY	Resp Supv: ENGR	Assoc Ref: 4 AWI-05.06.02	SR: N	Freq: 2 yrs
	ARMS: 3278	Doc Type: 3172	Admin Initials:	Date

APPROVED (Signatures Available in Master File.)

Part II: Determine if the Change Involves a Design Function (Resource Manual 5.3.2)

Yes	No	
☑	☐	Does the proposed activity involve Safety Analyses or an SSC credited in the Safety Analyses?
☑	☐	Does the proposed activity involve SSCs that support SSC(s) credited in the Safety Analyses?
☐	☑	Does the proposed activity involve SSCs whose failurecould initiate a transient (e.g., scram, loss of feedwater, etc.) or accident, or impact SSC(s) credited in the Safety Analyses?
☑	☐	Does the proposed activity involve USAR-described SSCs or procedural controls that perform functions that are required by, or are otherwise necessary to comply with, regulations, license conditions, orders, or technical specifications?
☑	☐	Does the activity involve a METHOD OF EVALUATION described in the USAR?
☐	☑	Is the activity a TEST or EXPERIMENT? (i.e., a non-passive activity that collects data)
☐ [3]	☑	Does the activity exceed or potentially affect a design basis limit for a fission product barrier (DBLFPB)?

If the answer to all of these questions is NO, skip Part III and proceed directly to Part IV.

If any of the above questions are marked YES, identify the specific Design function, method of evaluation, or DBLFPB involved. Proceed to Part III.

The primary containment isolation valves MO-2026 and MO-2027 serve as the primary containment boundary with isolatation on Group 2 signals to assure containment boundary integrity following a LOCA. The head spray function of RHR was removed as described in SCR-03-0075. MO-2026 and MO-2027 are currently deactivated in the closed position to provide containment isolation only.

[3] This activity requires a 10 CFR 50.59 Evaluation. Skip to Part IV.

PART III: Determine Whether the Activity Involves Adverse Effects (Resource Manual 5.3.3)

Answer the following questions to determine if the activity has an adverse effect on a design function. Any YES answer in Part III means that a 10 CFR 50.59 Evaluation is required, except where noted in Question 3.

1 Changes to the Facility or Procedures

Yes	No	
☐	☑	Does the activity adversely affect the design function of an SSC credited in safety analyses?
☐	☑	Does the activity adversely affect the method of performing a design function of an SSC credited in safety analyses?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO. describe the basis for the conclusion:
The design function of primary containment isolation valves MO-2026 and MO-2027 is to provide primary containment isolation for containment Penetration X-17. This modification removes MO-2026 and MO-2027 and welds a cap on the process pipe through penetration X-17. Therefore, any reference to MO-2026 and MO-2027 is removed from the USAR. Also, reference to Head Spray in Technical Specification Table 3.2.1 for Group 2 isolation signals will be deleted from the Technical Specifications by the License Ammendment Request process.

Procedures and drawings listed in Modification 02Q295 Plant Inpact List will be revised to provide configuration control associated with MO-2026 and MO-2027 removal with Group 2 logic change.

The function of Penetration X-17 continues to be primary containment boundary and meets the requirements described in USAR 5.2.1.3, Containment Penetrations. The basic design objective for the penetrations was to ensure that the integrity of the containment is maintained under the loading conditions defined below:
A. Normal operating conditions. The penetrations were designed for loads resulting from the full combination of normal operating pressure, thermal expansion seismic and dead loads.
B. Accident Condtions. The penetrations were designed for the loads resulting from the full combination of dead loads, seismic, thermal growth, and pressure conditions due to loss of coolant within the drywell, acting coincident with the larger of the following:
1. The jet reaction force on a penetration resulting from a circumferential (guillotine-type) break of the associated process line.
2. The jet reaction force on a penetration resulting from a longitudinal-type break of the associated process line.

3. The jet impingement loading on a penetration and its associated process line resulting from rupture of an adjacent process line.
C. Hydrodynamic LOCA Loads - not applicable to penetration X-17
The removal of MO-2026 and MO-2027 and the associated abandoned head spray process piping inside and outside containment removes all these loads and load combinations because these loads are dependant on forces imparted by or on the process pipe.

USAR 5.2.4.2 Containment Penetrations, states leakage rate tests of penetrations and across openings are conducted to verify the capability of the penetrations to maintain overall containment leakage within acceptable limits. Penetration X-17 is a bellows construction type that is testable and will remain in the test program. The construction test will test the welded cap to ensure it meets leakage requirement. In fact it will assure there is no leakage on the weld. Thus replacing the containment penetration valves with a welded cap on the process pipe ensures the total containment leakage rate is not affected. The reanalysis of the process pipe demonstrates that the plant piping is within code allowable stress limits with the removal of the head spray piping and valves. There is no adverse effect.

2 Changes to a Method of Evaluation

(If the activity does not involve a method of evaluation, these questions are ☐ **NOT APPLICABLE)**

Yes	No	
☐	☑	Does the activity use a revised or different method of evaluation for performing safety analyses than that described in the USAR?
☐	☑	Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in the USAR?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion:
The analysis of the pipe through X-17 with the removal of inboard and outboard containment isolation valves is in accordance with the method of evaluation for performing a piping analyses as discribed in the USAR, Section 12.2.1.10.2, thus there is no change.

3 Tests or Experiments

a. (If the activity does not involve a test or experiment,3.a and 3.b are ☑ **NOT APPLICABLE)**

Yes No

☐	☐	Is the proposed test or experiment bounded by other tests or experiments that are described in the USAR?
☐	☐	Are the SSC's affected by the proposed test or experiment isolated from the facility?

If both answers are NO, continue to 3.b. If the answer to either question is YES, then briefly describe the basis:

b. Answer these additional questions only for tests or experiments that do not meet the criteria above. If the answer to either question in 3.a is YES, then these questions are: ☐ **NOT APPLICABLE**

Yes	No	
☐	☐	Does the activity utilize or control an SSC in a manner that is outside the reference of the bounds of the design basis as described in the USAR?
☐	☐	Does the activity utilize or control an SSC in a manner that isinconsistent with the analyses or descriptions in the USAR?
☐	☐	Does the activity place the facility in a condition not previously evaluated or that could affect the capability of an SSC to perform its intended functions?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If all answers are NO, describe the basis for the conclusion:

Part IV : Conclusion
(Resource Manual 5.3.4)

Check all that apply:

1. A 10 CFR 50.59 Evaluation is ☐[4] required or ☑[5] NOT required

2. A USAR change is ☑[6] required or ☐ NOT required

Additional Comments and Reviews (Additional reviews may be required by 4 AWI-05-06.02, paragraph 4.6.5):

Assign Screening number (page 1)

Prepared by: LE Thompson — Laverne Thompson — 2/10/04 ~~10/31/2003~~ Date

Reviewed by: TMKinne — 2/10/04 ~~10/31/2003~~ Date

4 If it is concluded that a 50.59 Evaluation is required, it is not necessary to complete the screening.

5 If it is concluded that a 50.59 Evaluation is NOT required, assign a screening number and document the reviews as necessary.

6 If the activity requires a USAR change, Form 3473 needs to be completed even if a 10 CFR 50.59 Evaluation is not required. List the screening number on the USAR change request.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

This enclosure consists of current Technical Specification pages marked up with the proposed changes. The pages included in this exhibit are as listed below:

Pages
59
59a

NOTE: Replace the corresponding pages in the original LAR dated December 23, 2003 with the enclosed pages.

2 Pages follow

Table 3.2.4 Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation					
Function	Trip Settings	**Minimum No. of Operable or Operating Trip Systems**	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions*
1. Low Low Reactor Water Level	≥–48"	**2**	2	2 (Notes 1, 2, 3 ~~, 5, 6~~)	A. or B.
2. High Drywell Pressure	≤2 psig	**2**	2	2 (Notes 1, 2, 3 ~~, 5, 6~~)	A. or B.
3. Reactor Building Plenum Radiation Monitors	≤100 mR/hr	**2**	~~1~~2	~~1~~2 (Notes 1, 2 ~~, 4~~)	A. or B.
4. Refueling Floor Radiation Monitors	≤100 mR/hr	**2**	~~1~~2	~~1~~2 (Notes 1, 2 ~~, 4~~)	A. or B.

Notes:

(1) There shall be two operable or tripped Trip Systems for each function. **An Instrument Channel may be placed in an inoperable status for up to 6 hours for performance of required surveillances without placing the Trip System in the tripped condition provided that at least one other OPERABLE Channel in the same Trip System is monitoring that parameter.** ~~with two instrument channels per trip system and there shall be one operable or tripped trip system for each function with one instrument channel per trip system.~~

(2) Upon discovery that minimum requirements for the number of operable or operating Trip Systems or Instrument Channels are not satisfied action shall be initiated ~~to~~ **as follows:**

(a) ~~Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or~~ **With one Instrument Channel per Trip Function inoperable:**

1) For Functions 1 or 2, place the inoperable Channel or Trip System in the tripped condition within 12 hours

2) For Functions 3 or 4, place the inoperable Channel in a downscale trip condition, or place the Trip System in the tripped condition within 24 hours

-- OR --

Notes: (cont'd)

~~(b) Place the plant under the specified required conditions using normal operating procedures.~~

(b) With more than one Instrument Channel per Trip Function inoperable:

1) For Functions 1 or 2, immediately satisfy the requirements by placing the appropriate Channels or Trip Systems in the tripped condition,

2) For Functions 3 or 4, immediately proceed to Note 2c

-- OR --

(c) If (a) or (b) cannot be met, then place the plant under the specified required conditions using normal operating procedures.

(3) Need not be operable when primary containment integrity is not required.

~~(4) One of the two monitors may be bypassed for maintenance and/or testing.~~

~~(5) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated as follows:~~

~~(a) With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or~~

~~(b) With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing appropriate channels or systems in the tripped condition, or~~

~~(c) Place the plant under the specified required conditions using normal operating procedures.~~

~~(6) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other OPERABLE channel in the same trip system is monitoring that parameter.~~

* Required Conditions when minimum conditions for operation are not satisfied.

A. The reactor building ventilation system isolated and the standby gas treatment system operating.

B. Establish conditions where secondary containment is not required.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

This enclosure consists of revised Technical Specification pages that incorporate the proposed changes. The pages included in this exhibit are as listed below:

Pages
59
59a

NOTE: Replace the corresponding pages in the original LAR dated December 23, 2003 with the enclosed pages.

2 Pages follow

Table 3.2.4
Instrumentation That Initiates Reactor Building Ventilation Isolation
And Standby Gas Treatment System Initiation

Function	Trip Settings	Minimum No. of Operable or Operating Trip Systems	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions*
1. Low Low Reactor Water Level	≥-48"	2	2	2 (Notes 1, 2, 3)	A. or B.
2. High Drywell Pressure	≤2 psig	2	2	2 (Notes 1, 2, 3)	A. or B.
3. Reactor Building Plenum Radiation Monitors	≤100 mR/hr	2	2	2 (Notes 1, 2)	A. or B.
4. Refueling Floor Radiation Monitors	≤100 mR/hr	2	2	2 (Notes 1, 2)	A. or B.

Notes:

(1) There shall be two operable or tripped Trip Systems for each function. An Instrument Channel may be placed in an inoperable status for up to 6 hours for performance of required surveillances without placing the Trip System in the tripped condition provided that at least one other OPERABLE Channel in the same Trip System is monitoring that parameter.

(2) Upon discovery that minimum requirements for the number of operable or operating Trip Systems or Instrument Channels are not satisfied action shall be initiated as follows:

(a) With one Instrument Channel per Trip Function Inoperable:

1) For Functions 1 or 2, place the inoperable Channel or Trip System in the tripped condition within 12 hours

2) For Functions 3 or 4, place the inoperable Channel in a downscale trip condition, or place the Trip System in the tripped condition within 24 hours

-- OR --

Notes: (cont'd)

(b) With more than one Instrument Channel per Trip Function inoperable:

1) For Functions 1 or 2, immediately satisfy the requirements by placing the appropriate Channels or Trip Systems in the tripped condition,

2) For Functions 3 or 4, immediately proceed to Note 2c

-- OR --

(c) If (a) or (b) cannot be met, then place the plant under the specified required conditions using normal operating procedures.

(3) Need not be operable when primary containment integrity is not required.

* Required Conditions when minimum conditions for operation are not satisfied.

A. The reactor building ventilation system isolated and the standby gas treatment system operating.

B. Establish conditions where secondary containment is not required.