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OPPD
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December 5, 2003
LIC-03-0150

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

Reference: Docket No. 50-285

SUBJECT: Response to Generic Letter (GL) 2003-01, "Control Room Habitability"

In accordance with 10 CFR 50.4, Omaha Public Power District (OPPD) hereby submits information in response to GL 2003-01 (Reference 10). Attachment 1 contains OPPD responses to the GL 2003-01 questions. Attachment 2 contains the applicable Fort Calhoun Station (FCS) General Design Criterion (GDC) from the FCS Updated Safety Analysis Report, Appendix G as requested by question 3 of GL 2003-01. Attachment 3 lists the commitments that OPPD is making in this submittal. Attachment 4 contains a list of references.

OPPD's review conducted for this response determined that no design basis changes are required. However, licensing basis changes will be needed to comply with Regulatory Guides (RGs) 1.196 (Reference 8) and 1.197 (Reference 9). Therefore, OPPD is preparing a License Amendment Request (LAR) modeled after revisions to the improved standard technical specifications (ISTS) proposed in TSTF-448, Revision 1 (Reference 11). The LAR will bring the applicable FCS Technical Specifications more closely into alignment with Combustion Engineering ISTS (Reference 5).

TSTF-448, Revision 1 has not yet been approved by the NRC. Therefore, OPPD proposes to finalize and submit the LAR within 120 days of the NRC's "best and final" offer on the revision to ISTS proposed in TSTF-448, Revision 1. This will allow OPPD to address any NRC recommendations affecting control room habitability (CRH) without delaying the submittal while issues unrelated to CRH are resolved. The Nuclear Energy Institute (NEI) and OPPD are closely monitoring TSTF-448 developments.

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If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833.

Sincerely,

R. T. Ridenoure
Vice President

RTR/MLE/mle

Attachments:

1. OPPD GL 2003-01 Response
2. USAR, Appendix G, Criterion
3. OPPD Commitments
4. References

c: B. S. Mallet, Regional Administrator, NRC Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector

Omaha Public Power District Response to Generic Letter 2003-01 "Control Room Habitability"

1. **Provide confirmation that your facility's control room meets the applicable habitability regulatory requirements (e.g., GDC 1, 3, 4, 5, and 19) and that the CRHSs are designed, constructed, configured, operated, and maintained in accordance with the facility's design and licensing bases. Emphasis should be placed on confirming:**

   (a) That the most limiting unfiltered inleakage into your CRE (and the filtered inleakage if applicable) is no more than the value assumed in your design basis radiological analyses for control room habitability. Describe how and when you performed the analyses, tests, and measurements for this confirmation.

   (b) That the most limiting unfiltered inleakage into your CRE is incorporated into your hazardous chemical assessments. This inleakage may differ from the value assumed in your design basis radiological analyses. Also, confirm that the reactor control capability is maintained from either the control room or the alternate shutdown panel in the event of smoke.

   (c) That your technical specifications verify the integrity of the CRE, and the assumed inleakage rates of potentially contaminated air. If you currently have a ΔP surveillance requirement to demonstrate CRE integrity, provide the basis for your conclusion that it remains adequate to demonstrate CRE integrity in light of the ASTM E741 testing results. If you conclude that your ΔP surveillance requirement is no longer adequate, provide a schedule for: 1) revising the surveillance requirement in your technical specification to reference an acceptable surveillance methodology (e.g., ASTM E741), and 2) making any necessary modifications to your CRE so that compliance with your new surveillance requirement can be demonstrated.

   If your facility does not currently have a technical specification surveillance requirement for your CRE integrity, explain how and at what frequency you confirm your CRE integrity and why this is adequate to demonstrate CRE integrity.

   Response:

   The Omaha Public Power District (OPPD) filed an application for a construction permit (CPPR41) for Fort Calhoun Station, Unit No. 1 (FCS) with the Atomic Energy Commission (AEC) on April 18, 1967. The AEC granted a construction permit to OPPD on June 7, 1968. OPPD submitted a Final Safety Analysis Report (FSAR) in support of its application for an operating license for FCS on November 28, 1969. The AEC issued the operating license on May 24, 1973, pursuant to the Atomic Safety and Licensing Board's Order of April 5, 1973.

   OPPD's recent implementation of an alternative source term (AST) pursuant to 10 CFR 50.67, "Accident Source Term," now requires FCS to meet 10 CFR 50, Appendix A, GDC 19. However, with respect to the other GDC noted above, OPPD is required to adhere to the draft general design criteria (GDC) published for comment in the Federal Register prior to issuance of the operating license and to the regulations set forth in the Code of Federal Regulations as a

matter of law. OPPD has committed to adhere to the intent of NRC guidelines, and industry codes and standards, although deviations from these guidelines, codes, and standards may be made in system modifications in accordance with the requirements of 10 CFR 50.59.

At FCS, the control room envelope (CRE) is the plant area that encompasses the control room and adjacent plant areas. The structures that make up the CRE are designed to limit the inleakage of radioactive and hazardous materials from areas external to the CRE. A control room ventilation system (CRVS) provides the control room with isolation, pressurization, heating, ventilation, air conditioning (HVAC), filtration, and monitoring. As such, the CRVS provides assurance that the control room operators can remain in the control room and take actions to operate the plant under all normal and accident conditions.

Licensing and design basis information is found in the FCS Updated Safety Analysis Report (USAR), Sections 9.10, 14, and Appendix G. Nuclear Energy Institute (NEI) guidance contained in NEI 99-03, Revision 1, (Reference 7) was used to verify the bases pertaining to applicable regulatory requirements. The USAR describes the CRE, its systems and functional requirements. The physical as-built condition of the control room as represented by the USAR was verified. No discrepancies between the description of the CRE in the USAR and the HVAC systems controlling the airflow within the CRE and in adjacent areas were found. All modifications to the CRE and HVAC systems have been incorporated in the USAR and design basis documents.

Regulatory Guides (RGs) 1.196 (Reference 8) and 1.197 (Reference 9) have additional requirements that will require OPPD to revise the FCS Technical Specifications (TSs) and implement a new standing order (SO). OPPD will submit a license amendment request (LAR) following the NRC's "best and final" offer on TSTF-448, Revision 1 (Reference 11). The LAR will modify the FCS TSs to invoke the SO containing control room (CR) inleakage limits and testing requirements. The standing order is SO-G-115 currently in draft and titled "FCS Control Room Integrity Program." NRC approval of the LAR, which is also drafted, must be obtained before SO-G-115 can be implemented for use at FCS. The LAR which is closely modeled after the guidance contained in TSTF-448, Revision 1, will bring the applicable FCS TSs more closely into alignment with Combustion Engineering (CE) ISTS.

Administrative Controls

Many of the processes are already in place to maintain and monitor the performance of the CRE, the HVAC systems controlling the airflow within the envelope and in adjacent areas. The draft of SO-G-115 contains administrative controls to ensure continued compliance with CR design and license bases. Existing control programs applicable to control room integrity have been consolidated into the draft SO, which also contains a new requirement for CRE barrier breach control. The draft SO documents previously completed engineering analyses (EA) including the operator dose analysis (Reference 3), the chemical and toxic gas analysis (EA-FC-94-012, "Fort Calhoun Toxic Gas Analysis"), and the effects of smoke (EA-FC-01-013, "Effects of Secondary Environment resulting from a Fire Event").

In several instances, the draft of SO-G-115 proposes a longer interval between program assessment frequencies than contained in RG 1.196 and RG 1.197. FCS CR unfiltered inleakage results (see below) were extremely low despite over 26 years of operation at the time American Society for Testing and Materials (ASTM)-E741 tracer gas testing was conducted in 1999. These test results validate the FCS CR design, maintenance, and surveillance testing with regard to control room habitability. Therefore, in accordance with NEI-99-03, Revision 1 (Reference 7), Step 4.3.4, OPPD's position is that longer intervals between assessments than specified in RG 1.196/1.197 are appropriate for FCS. Additional detail and justification will be provided in the LAR concerning the specific exceptions to RG 1.196/1.197.

ASTM E741-93 Testing:

ASTM E741-93, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution" was performed on the FCS CRE in December 1999. For testing purposes, the CRE consisted of the CR, the mechanical equipment room, the computer room, and the mezzanine of the control room. Air inleakage into the CRE was measured with the CRVS operating in the pressurization mode. All adjacent spaces were aligned with normal HVAC operation to reflect actual conditions. Outside airflow was lower than normal airflow rates to reflect a worst case bounding condition.

Air leakage rates within the CRE with the CRVS operating in the pressurization mode were inferred using a NCS Corporation/Lagus Applied Technology (LAT), Inc., NCS/LAT Procedure 1204A Revision 2 "Constant Injection Tracer Ventilation Test." These procedures are based on the methodology described in ASTM Standard E741-93. Tracer gas flow rate measurements of makeup airflow rates were performed using NCS/LAT Procedure 1215 Revision 1 "Tracer Gas Flow Rate Determination Test." This procedure is based on ASTM Standard E2029-99, "Standard Test Method for Volumetric and Mass Flow Rate Measurement in a Duct Using Tracer Gas Dilution."

In addition, two tracer gas tests were performed to measure the amount of by-pass leakage across the normal mode isolation dampers with the CRVS in the pressurization mode. Damper by-pass leakage rates were inferred using NCS/LAT Procedure 1214 Revision 1 "Damper Leakage Test." Sulfur hexafluoride (SF_6) was used as the tracer gas in the measurement procedures. In all of the testing preformed at Fort Calhoun Station, SF_6 concentrations were determined using measurement specific analyzers optimized for detection of SF_6.

Air inleakage into the CR was measured using multiple combinations of CRVS components and a maximum of 38 SCFM inleakage (sum of filtered and unfiltered inleakage) was found. The unfiltered air inleakage rate measured less than 8 SCFM, while filtered air inleakage measured approximately 30 SCFM. Damper by-pass leakage rates were undetectable. The outside airflow measured at a range from 822 SCFM to 851 SCFM (measured with the CRVS in filtered mode). The control room is pressurized in the filtered mode.

As demonstrated in the test, a decrease in control room pressure towards atmospheric (with differential pressures close to 1/8 inches water gauge) is expected to have a minimal effect on unfiltered air inleakage.

Radiological Analysis:

On February 7, 2001, OPPD submitted a LAR (Reference 4) to replace the FCS accident source term used in the design basis radiological analyses for control room habitability with an alternative source term (AST) pursuant to 10 CFR 50.67, "Accident Source Term." The site boundary and control room dose analysis summary (Reference 3) are contained in Attachment E of Reference 4.

The NRC issued Amendment 201 (Reference 6) on December 5, 2001 approving the LAR. Implementation of the AST for site boundary and control room dose analysis is complete reflecting the guidance provided in RG 1.183 (Reference 1). The radiological consequences analyses, summarized in the dose analysis summary (Reference 3), utilizes a conservative unfiltered inleakage value based on the ASTM E741-93 tracer gas testing described above and documented in Reference 2. The radiological consequences analysis concluded that no changes to license or design basis documents were required.

Toxic Gas Analysis

Fort Calhoun Station evaluated its susceptibility to toxic gas events in accordance with RG 1.78, "Evaluating The Habitability Of A Nuclear Power Plant Control Room During A Postulated Hazardous Chemical Release," and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." EA-FC-94-012 documents the calculated probability of core damage due to storage and transportation of toxic materials near the Fort Calhoun Station per RG 1.78.

With the exception of ammonia, the probability risk analysis (PRA) conducted for EA-FC-94-012 supported the deletion of FCS TS monitoring requirements for toxic gases. Accordingly, in Amendment 183, all toxic gas monitoring requirements except for ammonia were deleted from the FCS TSs. The FCS TSs retain Specification 2.22 for monitoring ammonia as a defense-in-depth measure due to the large amount of ammonia stored near the plant.

As stated previously, a conservative unfiltered inleakage value of 38 SCFM was used in the operator radiological dose consequences analysis using the alternate source term. The toxic gas analysis was not modified due to the tracer gas testing since the response to the analysis is to go into the recirculation mode with no outside airflow. This differs from the radiological accident mitigating action, which is to operate in filtered mode. There is no change to the outside airflow and thus no effect on the toxic gas monitors ability to sense and alarm. Operators respond to a toxic gas alarm by donning breathing apparatus to protect them from potential unfiltered toxic gas inleakage expected to be minimal at atmospheric pressure.

Fire/Smoke Analysis

FCS performed an engineering analysis (EA-FC-01-013) of secondary environment (smoke) effects resulting from a fire. In part, the EA addressed the potential for smoke propagation to affect redundant safe shutdown components in adjacent fire areas. The EA also evaluated the feasibility of performing safe shutdown manual actions (including CR access and egress) necessary to achieve and maintain safe shutdown in a secondary environment (smoke).

The EA concluded that in a credible fire scenario, the secondary environment (smoke) effects will not prevent safe shutdown of the plant from either the control room or alternate shutdown panel, and that no license or design basis changes were necessary. However, the analysis did provide several recommendations for enhancements to handle smoke propagation more effectively.

The recommendations are being implemented through procedural and training enhancements and the procurement of additional manual smoke removal equipment for fire brigade personnel.

TS Verification of CRE Integrity

The current surveillance requirement (TS 3.1, Table 3-3, Item 10b), which is performed on a refueling outage frequency records the ΔP across the boundary of the control room to the adjacent spaces and the atmosphere. This surveillance will be retained when the FCS technical specifications are revised as it verifies that no significant degradation of the CR boundary has occurred. As stated above, in December 1999, with the control room pressurized and operating in filtered mode, unfiltered inleakage was measured with tracer gas testing. Despite over 26 years of commercial operation at the time, unfiltered inleakage was very low at 8 SCFM. Thus, FCS is confident that the ΔP surveillance in combination with visual inspections is sufficient to ensure that CR unfiltered inleakage will remain bounded by the radiological consequences analysis. The ΔP surveillance and visual inspections will ensure that no significant degradation of the CRE occurs during the period between the ASTM E741 tracer gas tests that will be required by SO-G-115.

As stated above, the FCS Technical Specifications will be updated to reflect the changes to NUREG-1432, Revision 2 (Reference 5) proposed by TSTF-448, Revision 1 (Reference 11). To meet applicable regulatory requirements, the draft LAR includes a new surveillance requirement (SR) to verify that control room inleakage is within limits established in accordance with SO-G-115. Current and proposed SRs for the control room ventilation system are comprehensive and verify control room habitability. TSTF-448, Revision 1, in combination with NUREG-1432, Revision 2, was used as a model for most of the changes in the draft LAR. Many of the TS revisions contained in the draft LAR were taken directly from TSTF-448, Revision 1 or from the improved standard technical specifications (ISTS) of NUREG-1432, Revision 2. Therefore, the draft LAR when submitted and approved will bring the FCS TS more fully into alignment with ISTS.

2. If you currently use compensatory measures to demonstrate control room habitability, describe the compensatory measures at your facility and the corrective actions needed to retire these compensatory measures.

Response:

No compensatory measures are in effect at this time.

3. If you believe that your facility is not required to meet either the GDC, the draft GDC, or the "Principal Design Criteria" regarding control room habitability, in addition to responding to 1 and 2 above, provide documentation (e.g., Preliminary Safety Analysis Report, Final Safety Analysis Report sections, or correspondence) of the basis for this conclusion and identify your actual requirements.

Response:

As stated in response to question 1, OPPD submitted a Final Safety Analysis Report (FSAR) in support of its application for an operating license for Fort Calhoun Station on November 28, 1969. The AEC issued an operating license on May 24, 1973, pursuant to the Atomic Safety and Licensing Board's Order of April 5, 1973.

With the implementation of an AST, OPPD is required to meet 10 CFR 50, Appendix A, GDC 19. With respect to the other GDC listed in question 1, OPPD is required to adhere to the draft GDC published for comment in the Federal Register prior to issuance of operating license and regulations set forth in the Code of Federal Regulations as a matter of law. OPPD has committed to adhere to the intent of NRC guidelines, industry codes and standards although deviations from these guidelines, codes, and standards may be made in system modifications in accordance with the requirements of 10CFR50.59.

With respect to 10 CFR 50, Appendix A, GDC 1, 3, 4, and 5, the following table lists the corresponding FCS USAR Appendix G criteria contained in Attachment 2.

10 CFR 50, APPENDIX A		**FCS USAR, APPENDIX G**			
No.	**Title**	**No.**	**Title**	**No.**	**Title**
1	*Quality Standards and Records*	1	*Quality Standards*	5	*Records Requirement*
3	*Fire Protection*	3	*Fire Protection*		
4	*Environmental and Dynamic Effects Design Bases*	23	*Protection Against Multiple Disability for Protection Systems*	40	*Missile Protection*
5	*Sharing of Systems, and Components*	4	*Sharing of Systems*		

Fort Calhoun Station
Updated Safety Analysis Report
Appendix G Criterion

APPENDIX G

RESPONSES TO 70 CRITERIA

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CRITERION 1 - QUALITY STANDARDS

Those systems and components of reactor facilities which are essential to the prevention of accidents which could effect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

This criterion is met. Components of the engineered safeguards systems are designed and fabricated in accordance with established codes and/or standards as required to assure that their quality is in keeping with the safety function of the component.

The following codes, standards, and procedures judged to be required to assure such quality will be applied. It is not intended, however, to limit quality standards requirements to this list.

High Pressure Injection, Low Pressure Injection, and Containment Spray Pumps

a) Pressure containing material have been tested and examined per ASME Code, Section VIII. Castings have been liquid penetrant inspected in accordance with Appendix VIII of Section VIII of ASME B&PV Code. Acceptance standards are in accordance with USAS B31.1 Case N-10.

b) Butt welds have been fully radiographed in accordance with ASME Code, Section VIII, paragraph UW-51.

c) Fillet welds have been liquid penetrant inspected in accordance with ASME B&PV Code Section VIII, Appendix VIII.

d) The pump supplier submitted certified mill test reports of pressure containing materials.

e) Pressure containing parts were hydrostatically tested to 1½ times design pressure by the supplier in accordance with ASME Code, Section VIII.

f) The pumps will undergo periodic leak testing after installation.

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g) At least one pump of each type has been hydraulic-performance tested for capacity and head, in accordance with the requirements of the Hydraulics Institute and the Power Test Code PTC-8.2. One pump of each type has been tested for operation at stated NPSH and stated thermal transient conditions.

h) Special consideration was given the design of the pump seals to provide a high degree of assurance of their proper operation, including compatibility of seal materials with water chemistry conditions and minimum dependence on externally supplied cooling water.

i) Pump drive motors conform to NEMA standards. The pump supplier provided motor test data for at least one pump of each type.

Stored Energy Tanks

ASME Code, Section III, Class C

Safety Injection and Containment Spray System Motor Operated Valves and Control Valves

a) The design criteria for pressure containing parts is in accordance with USASI B16.5. Castings were radiographed in accordance with ASTM-E-71-64 or ASTM-E-186-65T as applicable. Radiographs were made in accordance with ASTM-E-94-62.

b) Pressure containing materials were tested and examined per ASME Code, Section VIII.

c) Radiographic inspection of pressure containing welds was performed in accordance with the requirements of ASME Code, Section VIII, ASTM-E-99-63 and ASTM-E-94-62.

d) Certified mill test reports of pressure containing materials were provided by the supplier.

e) Valve motors are in accordance with NEMA standards. Particular attention was given to the design of the valve motors to ensure their proper operation in a post-accident environment.

f) Pressure containing parts were hydrostatically tested in accordance with USASI B16.5.

g) Periodic leak testing will be performed on the valves after installation.

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Shutdown Heat Exchangers

a) Pressure containing materials were tested and examined per ASME Code, Section III, Class A (on tube side) and Class C (on shell side).

b) Heat transfer design and physical design are in accordance with TEMA standards.

c) Certified mill test reports of pressure containing materials were provided by the supplier.

d) Radiographic inspection of pressure containing welds was performed in accordance with the requirements of ASME Code, Section VIII, ASTM-E-99-63 and ASTM-E-94-62.

e) Pressure containing parts were hydrostatically tested in accordance with ASME Code, Section III.

Omaha Public Power District reviewed tests and inspections during material procurement and fabrication of the components to assure conformance with the quality control techniques of the applicable codes and standards. Records of all test and inspection results will be maintained.

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CRITERION 3 - FIRE PROTECTION

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room and components of engineered safety features.

This criterion is met. The reactor facility is designed to minimize the probability of such events as fires and explosions and to minimize potential effects of such events to safety. Noncombustible fire resistant materials are used whenever practical throughout the facility. The facility is provided with a fire protection system which includes detectors, alarms, water supply, secondary water supply, deluge systems, sprinklers, hose lines and portable extinguishers. For further information the fire protection system is described in Section 9-11 of the Fort Calhoun Station - Unit No. 1 Final Safety Analysis Report.

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CRITERION 4 - SHARING OF SYSTEMS

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

This criterion is met. The design of Ft. Calhoun Station Unit No. 1 is not based on sharing of systems and components with a future reactor facility.

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CRITERION 5 - RECORDS REQUIREMENT

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

This criterion is met. The Omaha Public Power District is the owner and operator of the Fort Calhoun nuclear facility. The Omaha Public Power District will maintain records of the design, fabrication, and construction of essential components of Fort Calhoun Station throughout the life of the reactor.

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CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of protection function.

This criterion is met. The protection systems are designed to provide the maximum practical degree of redundancy of channels and power sources. Physical separation of the multiple detectors and signal leads for these systems minimizes the vulnerability to a single accident of an individual protective system. Any accident or adverse condition which causes a loss of power to a protective system will result in automatic reactor shutdown.

The protective systems are designed to operate in the environment of normal plant operation and to terminate any transient condition before a measured variable reaches a level where the reactor or the respective instrumentation is damaged. Therefore, the equipment is not normally subject to adverse conditions which would result in failure. Because of the redundancy provided, loss of a single channel will not result in a loss of protection from accident(s) which is afforded by that protective system.

Components located inside the containment building which are required to operate following a DBA will withstand the post DBA environment for the required time period.

A major fire in the control room could result in extensive damage to instrumentation, controls, and protective circuitry and would almost certainly cause an automatic reactor shutdown. In the event that an automatic shutdown does not occur and the operator does not manually trip the reactor, the reactor can be safely shutdown from outside the control room.

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CRITERION 40 - MISSILE PROTECTION

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

This criterion is met. The high-pressure equipment in the reactor coolant system is surrounded by reinforced concrete and steel structures designed to stop all credible missiles and withstand the forces generated in a loss-of-coolant accident for break sizes up to and including the double-ended rupture of the reactor coolant pipe. The containment liner, the reactor coolant loops, the steam and feedwater piping, the auxiliary cooling piping and the containment cooling system are protected from missiles generated within the containment building. Barriers are provided where the use of radiation shielding and/or support structures for missile shielding is not feasible.

Two of the containment air recirculation and cooling units are located on the operating floor, and two on a concrete platform above the first pair. They are protected from missiles by the walls of the reactor coolant equipment compartments and by the missile shield placed over the reactor. Auxiliary coolant enters the air handling unit from below the operating floor, so that it is remote from any missiles.

The most critical plant missile external to the containment building has been determined to be a turbine last stage wheel fragment. Analyses show this missile will not perforate or impair the structural integrity of the containment building.

The emergency core cooling system will be designed to prevent loss of design capability during the emergency of a pipe rupture or earthquake. Piping connecting vessels will be engineered to restrict movement to certain maximum values during these emergencies. The piping system will be designed to accept these emergency imposed movements and still remain within code allowable limits for stress. Flexibility calculations will be according to the Code for Nuclear Piping USASI B31.7.

OPPD GL 2003-01 Response Commitments

NO.	COMMITMENT	DUE DATE
1	Submit CRH LAR, to include specific exceptions to RG 1.196/1.197 and justification for the exceptions (AR 33774).	Within 120 days of the NRC's "best and final" offer on TSTF-448, Revision 1
2	Finalize SO-G-115, "FCS Control Room Integrity Program," (AR 33774).	Prior to Submittal of CRH LAR
3	Implement CRH License Amendment & SO-G-115.	Following NRC approval of CRH LAR & within implementation period requested in CRH LAR

References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, December 1999
2. NCS Corporation Report, "FCS Control Room Envelope Inleakage Testing," February 2000
3. Stone & Webster Report, "Implementation of Alternative Source Terms, Site Boundary and Control Room Dose Analysis for Fort Calhoun Station," January 2001
4. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), "Application for Amendment," dated February 7, 2001, (LIC-01-0010)
5. NUREG-1432, Revision 2, "Standard Technical Specifications Combustion Engineering Plants, April 2001
6. Letter from NRC (A. B. Wang) to OPPD (S. K. Gambhir), "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment (TAC NO. MB1221)," dated December 5, 2001 (NRC-01-112)
7. Nuclear Energy Institute (NEI) 99-03, Revision 1, "Control Room Habitability Guidance," March 2003
8. Regulatory Guide 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," May 2003
9. Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," May 2003
10. NRC Generic Letter 2003-01, "Control Room Habitability," dated June 12, 2003 (NRC-03-109)
11. Letter from Technical Specifications Task Force (TSTF) to NRC (W. D. Beckner) dated August 19, 2003, TSTF-448, Revision 1, "Control Room Habitability"