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2.C.2.1
FYR 81-152

November 18, 1981

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Reference: (a) License No. DPR-3 (Docket No. 50-29)

Subject: SEP Topic Assessment Completion

Dear Sir:

Enclosed please find our assessments of the following topics:

III-7.D	Containment Structural Integrity Test
IX-1	Fuel Storage
IX-5	Ventilation Systems
XV-19	Loss of Coolant Accidents

We trust this information is satisfactory; however, if you have any questions, please contact us.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. A. Kay

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Senior Engineer - Licensing

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SEP TOPIC III - 7.D

Containment Structural Integrity Test

Introduction

The original structured integrity test performed on the Yankee Nuclear Power Station vapor container has been reviewed against current regulatory criteria. This review demonstrated that the original structural integrity test is equal to or more conservative than current requirements.

Current Criteria

The current criteria pertaining to containment structural integrity tests are

- NRC Standard Review Plan Section 3.8.2
- ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE

Containment Original Code of Construction

The containment at the Yankee Nuclear Power Station is a Steel Sphere, called the vapor container.

The vapor container was designed, fabricated and tested in accordance with the rules of the ASME Code, Section VIII, which was the forerunner of Subsection NE of Section III of the ASME Code.

The material of fabrication was SA-201 Gr B steel, processed to the requirements of SA-300. This is comparable to current grades of SA-516.

All butt weld joints in the shell were fully radiographed. In addition, for conservatism a 90 percent weld efficiency factor was used in the analysis of the vapor container.

Original Structural Integrity Test

Following fabrication the completed vapor container was soap bubble tested at all welds and pneumatically tested at 1.25 times design pressure (40 psig test pressure). This exceeds the current Code test requirement of 1.1 times design pressure.

The test was conducted in accordance with the requirements of Section VIII of the ASME Code and witnessed by an authorized inspection agency.

Following completion of the test the unfired pressure vessel "U" stamp was affixed. A rubbing of the nameplate is attached as Figure 1.

Current Structural Integrity Test Requirements

The minimum test pressure required by Subsection NE of the ASME Code is 1.1 times design pressure. NRC Standard Review Plan 3.8.2 requires that the structural integrity test be performed in accordance with SRP 3.8.2.

Conclusion

It is concluded that the original structural integrity test complies with current regulatory criteria. Therefore SEP Topic III - 7.D is satisfactorily resolved.


HORTONSPHERE
NUCLEAR CONTAINMENT VESSEL
7-6120
1959
37.5
AMBIENT+249°
U
125
0
1,022,600
1.25" 1.00" .938" .888" .881" .375"
HARTFORD
HSB 990
G 990
CHICAGO BRIDGE & IRON COMPANY

F I G. 1

SYSTEMATIC EVALUATION PROGRAM

TOPIC IX-1 FUEL STORAGE

1.0 INTRODUCTION

The objective of this topic is to assure that fuel is stored safely with respect to criticality ($K_{eff} < 0.95$), cooling capability (outlet temperature $< 150^{o}F$), shielding, and structural capability.

2.0 CRITERIA

This topic has been reviewed for compliance to the requirements of ANSI N210-1976/ANS 57.2, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations.

3.0 DISCUSSION

The spent fuel storage pool and racks at the Yankee Nuclear Power Station are currently being reviewed as part of Proposed Change No. 158 to the Technical Specifications. NRC review and approval of that proposed change and its supplements resolves the SEP-related concerns of this topic.

4.0 REFERENCES

1. USNRC Letter to YAEC, dated March 7, 1978

2. USNRC Letter to YAEC, dated January 22, 1981

SYSTEMATIC EVALUATION PROGRAM
TOPIC IX-5 VENTILATION SYSTEMS
YANKEE ROWE

I. INTRODUCTION

To assure that the ventilation systems have the capability to provide a safe environment for plant personnel and for engineered safety features, it is necessary to review the design and operation of these systems. For example, the function of the spent fuel pool area ventilation system is to provide ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational transients, and following postulated fuel handling accidents. The function of the engineered safety feature ventilation system is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design basis accidents.

II. REVIEW CRITERIA

The current criteria and guidelines used to determine if the plant systems meet the topic safety objective are those provided in Standard Review Plant (SRP) Sections 9.4.1, "Control Room Area Ventilation System", 9.4.2, "Spent Fuel Pool Area Ventilation System", 9.4.3, "Auxiliary and Radwaste Area Ventilation System", 9.4.4, "Turbine Area Ventilation System" and 9.4.5, "Engineered Safety Feature Ventilation System".

III. RELATED SAFETY TOPICS AND INTERFACES

The scope of review for this topic was limited to avoid duplication of effort since some aspects of the review were performed under related topics. The related topics and the subject matter are identified below. Each of the related topic reports contains the acceptance criteria and review guidance for its subject matter.

II-2.A	Severe Weather Phenomena
III-1	Classification of Structures, Components and Systems (Seismic and Quality)
III-6	Seismic Design Considerations
III-12	Environmental Qualification of Safety Related Equipment
VI-4	Containment Isolation System
VI-7.C.1	Independence of On-site Power
VI-8	Control Room Habitability
VI-3	Systems Required for Safe Shutdown
IX-3	Station Service and Cooling Water Systems
IX-6	Fire Protection
XV-20	Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment)

TMI III.D.3.4 Control Room Habitability

IV. EVALUATION

The systems reviewed under this topic are the Control Room Area Ventilation System, Pump and Switchgear Rooms Ventilation System, Primary Auxiliary Building Ventilation System, Waste Disposal Building Ventilation System, Spent Fuel Pool Ventilation System, Diesel Generator Building Ventilation System, Turbine Area Ventilation System, and the Heating Boiler Room Ventilation System.

A. Control Room Area Ventilation System

The function of the Control Room Area Ventilation System (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to assure the operability of control room components during normal operating, anticipated operational transient and design basis accident conditions.

As a result of TMI, this system is being reviewed generically (TMI Item III.D.3.4, Control Room Habitability) to assure compliance with Criterion 19, "Control Room" of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50. Therefore, the CRAVS was not reviewed under this topic.

B. Pump and Switchgear Rooms Ventilation System

The Pump and Switchgear Rooms Ventilation System (PSRVS) services the boiler feed pump area on the ground floor level of the turbine building, and the switchgear room located directly under the control room. The switchgear room contains the Nos. 1 and 2 battery rooms, the 2400 volt switchgear, and a substantial amount of other plant electrical equipment.

The switchgear room is considered to be essential, because it houses to No. 1 and 2 batteries, the battery charges and inverter, vital instrumentation power supplies, emergency MCC 1, and the vapor container hydrogen analyzer and sample station, all of which are important to safety. Air is supplied by two supply fans, FN-13-1 and 2, and exhausted into the turbine building through fire dampers. A separate exhaust fan, FN-21, exhausts air from the two battery rooms and is ducted outside of the turbine building. Air is drawn into the battery rooms from the switchgear room through filtered openings in the battery room doors. A failure of the battery room ventilation system is alarmed in the main control room.

All three fans are tripped by the fire detection system just prior to activating the halon supression system in the switchgear room. The two supply fans are also tripped by the fire detection system if smoke is detected in the supply duct. All the exhaust dampers from the switchgear room and battery rooms are tripped by the fire detection system by electrically tripped thermal links. These thermal links are also heat sensitive.

The two supply fans are also tripped at 1700 psi pressurizer pressure in the event of a steam line break which may cause steam to be drawn into the switchgear room by the fans.

The two supply fans are not powered from emergency power sources. However, in the event of a loss of off-site power, most equipment in the switchgear room is denergized. The remaining heat loads are quite small, and past experience shows that it would take several hours to raise the room temperature to where equipment operation may be affected. If off-site power could not be restored, the fans could be powered from on-site emergency sources.

The battery room exhaust fan is not powered from an emergency power source either. This fan is required to prevent the buildup of hydrogen gas in the battery room during battery charging. The buildup of hydrogen must be evaluated with a loss of ventilation. The evaluation is in progress as part of a modification to replace the existing batteries.

C. Primary Auxiliary Building Ventilation System

The Primary Auxiliary Building Ventilation System (PABVS) is segregated into two distinct sections; the radioactively clean portion and the potentially contaminated portion.

The radioactively clean portion of the Primary Auxiliary Building (PAB) contains the emergency feedwater pumps, component cooling pumps, some containment isolation system equipment, and MCC-4, making the area essential. The area is ventilated by a roof type exhaust fan, RF-11, which is mainly required only during the summer months. Air is drawn in through windows in the upper level PAB, and through flood control one-way dampers in the lower level PAB. The upper level PAB also contains steam generator blowdown lines. In the event of a steam line break, ventilation louvers in the upper level PAB will open to vent the building and prevent the formation of a harsh environment in the lower level PAB, where the majority of the essential equipment is located.

The potentially contaminated portion of the PAB, the cubicle area, contains mostly primary plant support equipment required for manual plant operation. The charging pumps, shutdown cooling pump are the major electrical components that would be required post-accident and may be effected by a loss of ventilation. This area would therefore be considered essential.

Filtered outdoor air is supplied by a unit ventilator, UV-5, and distributed by ducting to the various equipment cubicles. Air is exhausted from the various equipment cubicles through a high efficiency filter unit consisting of prefilters, charcoal and HEPA filter, and an exhaust fan, FN-19, to the primary vent stack. The vapor container purge system filter assembly and fan can function as a backup for the PAB exhaust system.

None of the ventilation equipment in the PABVS system is powered by emergency sources. However, with a loss of off-site power, none of the electrical equipment being serviced by the PABVS has power available either. The loads that are necessary can be fed by on-site emergency power when and if required. The same is true for the ventilation equipment. However, it will not be required in the same time frame since the heat generated by the small amount of equipment running will initially be dissipated within the building. With the postulatd loss of off-site power, the building heat load is greatly reduced, reducing the building ventilation requirements.

D. Waste Disposal Building Ventilation System

The Waste Disposal Building Ventilation System (WDBVS) services all equipment located in the waste disposal building. This includes equipment for the processing of gaseous and liquid wastes. This equipment is not required post-accident or for safe shutdown, therefore, the WDBVS is not considered to be an essential system.

E. Spent Fuel Pool Ventilation System

The Spent Fuel Pool Ventilation System (SFPVS) services the spent fuel pool area which contains the spent fuel pit and its cooling system. This equipment is not required post-accident or for safe shutdown. A failure of the ventilation system will not result in the potential for off-site exposures comparable to the guidelines of 10 CFR Part 100. Also, SEP Topic XV-20, Radiological Consequences of Fuel Damaging Accidents (Reference 1) has determined that the radiological consequences of a fuel handling accident are well within the guidelines of 10 CFR Part 100. Therefore, the SFPVS is not an essential system.

F. Diesel Generator Building Ventilation System

The Diesel Generator Building Ventilation System (DGBVS) services the diesel generator building which contains the emergency diesel generators and switchgear, the No. 3 battery and charger, and the safety injection system pumps. This equipment is required post-accident and for safe shutdown, requiring the DGBVS system to be classified as essential.

The three diesel generators are located in individual cubicles. The ventilation system for each cubicle is separate and independent to that cubicle. Air is supplied to the cubicle by an intake hood on the roof with a motor operated damper that is actuated by the diesel starting circuit. Air is exhausted outside through an exhaust louver on the front of the cubicle. The exhaust ducting also includes a motor operated damper that is also actuated by the diesel starting circuit. The diesels are air cooled, and the diesel cooling fan also serves as the cubicle exhaust fan. It is driven directly off the diesel.

The areas of the diesel generator building containing the emergency switchgear, No. 3 battery and the safety injection pumps are ventilated by a separate system. Air is supplied to the room by four motor operated dampers and one ventilating unit, VU-1, with its own motor operated damper. Air is exhausted from the room to the outside by two roof exhaust and vent fans, PRV-1 and 2. All of this equipment is actuated by temperature switches on high temperature in the room. PRV-1 and 2, and VU-1 will start automatically on high temperature in the room. The motor operated dampers are also actuated on high temperature.

All of the electrical ventilation equipment of the DGBVS system is powered by on-site emergency power sources. The No. 3 battery is located in a screened-in cage which allows complete mixing with the atmosphere of the diesel generator building. Any hydrogen generated in the battery space will be disbursed throughout the building. Past experience has demonstrated that due to the size and construction of the building, the hydrogen generated will be adequately disbursed without ventilation.

G. Turbine Area Ventilation System

The Turbine Area Ventilation System (TAVS) services the turbine and condenser rooms. The equipment in these areas is not required post-accident or for safe shutdown. Therefore, the TAVS system is not considered to be an essential system.

H. Heating Boiler Room Ventilation System

The Heating Boiler Room Ventilation System (HBRVS) services the heating boiler room. The only piece of equipment located in this room which may be required post-accident or for safe shutdown is the steam driven emergency feed pump. The steam driven emergency feed pump is manually operated, and its operation is not effected by a lack of ventilation. Ventilation would be required for the operator only. The heat load of this pump is so low that adequate ventilation can be achieved by opening the doors to the room. Therefore, the HBRVS system is not an essential system.

V. SUMMARY

The ventilation systems reviewed meet the SRP criteria. The only area requiring further review is the ventilation of battery Room 1 and 2, the concern being hydrogen buildup. This review is in progress.

VI. REFERENCES

1. Letter USNRC to YAEC, dated September 6, 1981.
2. Yankee Rowe Final Hazards Summary Report, Revision 1, dated July 1, 1980.

SEP/XV-19 LOCA

Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

GENERAL

A loss of reactor coolant will result from a rupture of the primary coolant system piping. The size of the rupture can range in size from a small leak which can be controlled by charging flow, up to the highly unlikely rupture of the largest pipe in the system. System response following a LOCA depends on the size and location of the break. All LOCA's which have leak rates in excess of charging capability will lead to system depressurization. Severe LOCA's can cause core uncovery and fuel failures. Reactor protection for loss-of-coolant accidents is provided by the Emergency Core Cooling System.

The primary function of the ECCS is to deliver borated water to the reactor vessel to assure adequate core cooling during loss-of-coolant accident situations for a spectrum of break sizes including the double-ended rupture of the largest pipe in the system. ECCS performance for zirconium-fueled cores must be shown to be within the criteria set forth in 10CFR50.46. In brief, these criteria are:

1) Peak cladding temperature less than 2200°F.

2) Maximum cladding oxidation less than 17%.

3) Total hydrogen generation shall not exceed 1% of total zirconium.

4) Maintenance of coolable geometry.

5) Long-term coolability.

Large LOCA's, though highly improbable, place the most demand upon the ECCS in terms of liquid delivery. The ECCS must deliver large amounts of water to recover and cool the core in a relatively short period of time.

Small LOCA's require flow at much higher pressures than the large LOCA's with the flow quantity required being less.

The Yankee Nuclear Power Station Emergency Core Cooling System consists of three high pressure safety injection pumps (HPSI), three low pressure safety injection pumps (LPSI), and an accumulator. The system is arranged such that at high RCS back pressures, the HPSI and LPSI pumps operate in series delivering borated water up to 1550 psig RCS pressure. Below the LPSI pump shutoff pressure of 650 psig, the HPSI and LPSI pumps operate in parallel. LPSI and HPSI pumps discharge flow to separate distribution headers, which in turn distributes the ECC water to the RCS cold legs via common injection nozzles. The accumulator, normally maintained at ambient pressure, is pressurized with nitrogen to 477 psig upon receipt of an SIAS. This constant nitrogen overpressure is maintained during injection until the accumulator is isolated on low level. ECC water from the accumulator enters the ECCS piping downstream of the LPSI pumps.

The way in which the ECCS responds to a LOCA following the receipt of a safety injection actuation signal (SIAS) at 1700 psig RCS pressure is, of course, dependent upon the magnitude of the break. There are two modes in which the ECCS functions. First, in the injection mode, ECC water is delivered to the RCS via the accumulator and the ECCS pumps. In the recirculation mode, spilled and condensed ECC water which has collected in the vapor container sump is recirculated via the ECC pumps back into the RCS. The time at which the switchover from injection to recirculation takes place varies according to break size. For a large LOCA, this occurs very quickly; whereas it will take longer to occur for a very small break (it should be noted that SI accumulator may or may not inject water, depending upon the MCS pressure response).

The injection mode is characterized by a minimum of two ECCS trains operating (2 HPSI and 2 LPSI) with flow reaching the MCS via either the HPSI distribution header or both the HPSI and LPSI distribution headers depending on system pressure. In the recirculation modes, the number of operating ECCS trains is a function of the vapor container back pressure, but a minimum of one HPSI pump flowing is required.

A subset of the recirculation mode is hot leg injection which is required 20-24 hours after a LOCA to prevent boron precipitation. Sufficient ECCS flow to remove decay heat is directed to the No. 4 loop hot leg via the charging header while the same amount of ECC flow is being directed to the cold legs. This assures that sufficient flow to remove decay heat is reaching the MCS via both cold and hot leg injection paths.

The vapor container (containment) is a passive, integral part of the ECCS. In addition to being a pressure boundary and a radiation boundary, it also serves as the ultimate heat sink during the long-term (recirculation) phase of a large LOCA. Once the contents of the SI tank water have been injected, there is adequate level in the vapor container sump to provide suction to the ECC pumps. The ECC pumps return this water back to MCS where it removes decay heat from the core. The ECCS water/vapor spills through the break and is then condensed on the surface of the vapor container which transfers the heat to the atmosphere. The condensate then returns back to the vapor container sump and the cycle is completed.

ANALYSIS

A spectrum of break sizes up to and including a double-ended rupture of the cold leg has been examined for Yankee using an approved evaluation model which conforms to the requirements of Appendix K of 10CFR50. Both large breaks and small breaks have been examined in the analyses summarized below.

Large Break Analysis

The most recent large break LOCA analysis was submitted with the Core XV Reload Analysis, Reference XV-19-1. This analysis was performed using YAEC's approved WREM-Based PWR ECCS Evaluation Model, Reference XV-19-2. Six different cold leg pipe breaks, (three double-ended guillotine breaks, and three double-ended split breaks) with discharge coefficients (C_D) of 1.0, 0.8 and 0.6 were examined. The selection of breaks for this analysis is based on previous evaluations which clearly identified the cold leg split and guillotine breaks as most limiting.

The results of this analysis show that the most limiting break is a double-ended guillotine break at the pump discharge with a discharge coefficient, C_D, of 1.0. For this break, the peak clad temperature was calculated to be 1875°F, well below the 10CFR50.46 limit of 2200°F. Clad oxidation (both peak and total) are also well within limits.

Since the break spectrum analysis is performed beginning-of-life (BOL) with fuel operating at a maximum linear heat generation rate (LHGR) of 9.0 kw/ft, a burnup sensitivity study is performed on the most limiting case in order to raise the LHGR to acceptable values required for full power operation. Maximum allowable LHGR's are determined at various burnup points throughout the cycle with a maximum peak clad temperature of 2094°F occurring for fresh fuel at 25 effective full power days.

SMALL BREAK ANALYSIS

The most recent small break LOCA analysis of the Yankee Nuclear Power Station was submitted as part of the Core XIII reload analysis performed in 1977 and is contained in References XV-19-3 and XV-19-4. Again, this analysis used Yankee's generically-approved WREM-Based PWR Evaluation Model (Reference XV-19-2). Five small breaks occurring in the cold leg were considered with equivalent break diameters of 2.25, 4.0, 5.0, 7.5 and 10.0 inches. These analyses were conducted at BOL with the fuel operating at a peak LHGR of 12.85 kw/ft, representing a total peaking factor of 2.76.

The most limiting break size was found to be the 4-inch ID break for which a peak cladding temperature of 1793°F was calculated. This peak clad temperature is well below the 10CFR50.46 limits and shows that the small breaks are less limiting than the large breaks at Yankee.

Small Break Analyses - Post-TMI

Generic analyses of small break LOCA's were submitted by Westinghouse in response to NRC Bulletins and Orders Task Force requirements. The staff has accepted these analyses as a basis for providing information on plant response and as an aid to developing guidelines for operator action. The staff considers these generic analyses to be representative of the response for Yankee to a postulated small break LOCA; however, the Yankee Nuclear Power Plant would respond somewhat differently to small breaks than more recent PWR's because the high pressure injection pump has a 1550 psi shutoff head, compared with a shutoff head of 2250 psi for the more recent plants. The effect of the lower Yankee HPI shutoff head is that small breaks will result in depressurization to 1500 psi, and full repressurization of the RCS is not possible using the safety grade HPI. Since 1500 psi corresponds to a saturation temperature of 596°F, and the hot leg temperature is 563°F, a depressurization to 1500 psi will not result in voiding.

As a result of the review of these analyses, the staff expressed concern about the applicability of current evaluation models and their application to the expanded scope of small break LOCA analyses now being considered. As part of the TMI Task Action Plan (NUREG-0737), which is beyond the scope of the SEP review, Yankee intends to revise and resubmit the small break LOCA analysis methods for staff approval. Plant specific calculations, using these revised methods, will then be required to show compliance with 10CFR50.46.

CONCLUSION

The loss-of-coolant accidents analyzed for the Yankee Nuclear Power Plant meet the acceptance criteria in Appendix K, 10CFR50.

New small break LOCA analyses using revised evaluation models will be conducted as part of the TMI Task Action Plan and will not be included as part of the SEP review.

XV-19 REFERENCES

1. Attachment D to Proposed Change No. 173, "Core XV Refueling", March 26, 1981. ("Yankee Nuclear Power Station Core XV Performance Analyses", YAEC-1240, March 1981.)

2. YAEC-1160, "Application of Yankee WREM-Based Generic PWR ECCS Evaluation Model to Maine Yankee", July 1978.

3. Proposed Change No. 145, Supplement No. 5, "Yankee Rowe Core XIII ECCS Performance Evaluation", August 1, 1977.

4. Proposed Change No. 145, Supplement No. 7, WYR 77-90, "Additional Yankee Rowe Core XIII Small Break Analysis", September 21, 1977.