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PRESIDENT

December 27, 1985
Fort St. Vrain
Unit No. 1
P-85499

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attn: Mr. H.N. Berkow, Project Director
Standardization and Special
Projects Directorate

Docket No. 50-267

SUBJECT: Fort St. Vrain Equipment Qualification

REFERENCE: 1) NRC Letter Dated November 5, 1985, Butcher to Lee, (G-85452)

Dear Mr. Berkow:

Reference 1) submitted requests for additional information needed by the NRC staff to determine if the Fort St. Vrain (FSV) Environmental Qualification (EQ) Program is in compliance with 10CFR50.49. Attachment 1 to this letter provides responses to those requests. Attachment 2 to this letter provides the System Description. Attachment 3 presents the temperature profiles used in the FSV EQ Program.

The Technical Specification changes and Safety Evaluation associated with the SLRDIS will be submitted in the near future under a separate cover letter, P-85456. If you have any questions on this subject, please contact Mr. M.H. Holmes at (303) 480-6960.

Very truly yours,

R. F. Walker

R.F. Walker,
President



A048 Add: W. Johnston (EB)
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Attachments

# LIST OF ATTACHMENTS AND SUMMARY DESCRIPTION

SUMMARY DESCRIPTION FOR OPERATION FOLLOWING
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## SUMMARY DESCRIPTION FOR THE MODE OF OPERATION FOLLOWING COMPLETION OF FSV EQ PROGRAM

The following is a summary of PSC's intentions with regard to the mode of operation of Fort St. Vrain (FSV) up to 100% power after completion of the FSV EQ program.

In the event of a high energy line break (HELB) and the resulting actuation of the Steam Line Rupture Detection Isolation System (SLRDIS), the fire water forced circulation mode of reactor shutdown will be utilized. At the completion of the FSV EQ program, all electrical equipment, instruments and valves required to perform this will be fully qualified according to 10CFR50.49 guidelines.

In the safe shutdown mode, forced circulation is accomplished by supplying fire water to drive at least one helium circulator pelton water drive and to one steam generator economizer/evaporator/super heater (EES) or reheat section to remove the decay heat. This mode of safe shutdown is in accordance with the current licensing basis as described in the FSV FSAR, Section 1.4.5.

The following major systems are used to accomplish safe shutdown using forced circulation:

- Reserve shutdown system
- Fuel storage facility cooling
- Helium circulator bearing water
- Helium circulator pelton drive
- Helium circulator brake and seal
- Steam generator EES section
- Steam generator reheat section
- Piping systems to provide cooling water to the steam generators
- Circulating water make-up
- Service water
- Fire water pumps
- Liner cooling system
- Control room HVAC
- Instrument air
- Hydraulic power to valves
- Standby diesel generators

The required portions of the above systems with the exception of the helium circulator brake and seal system are currently classified as safe shutdown. In addition to the above systems, the equipment required for the new SLRDIS system will be qualified along with the reactor building louver system. The louvers are not specifically required for the design basis events as part of the FSV EQ program but the system will be qualified to maintain building integrity. All of the components in these systems that are required to operate are included on FSV's EQ Master Equipment List and will be environmentally qualified.

Actuation of the SLRDIS System brings the plant into a loss of forced circulation (LOFC) situation. Based on the analysis presented in FSAR section 14.4.2.2, restoration of forced circulation cooling to the core can be delayed up to 1 1/2 hours with no fuel failure. Depending upon the nature of the design basis event which results in a harsh environment, plant operators will re-establish forced circulation utilizing all available equipment in accordance with plant procedures. Assuming all non-qualified equipment fails, forced circulation cooling will be established utilizing safe shutdown cooling equipment per Section 1.4.5. of the FSAR. In order to initiate forced circulation with fire water, certain manual actions are required. Based on latest building temperature profiles, the operator can gain access to the plant within 1 hour (protective clothing may be required). As described in a letter P-85460, dated December 10, 1985 from Walker to Berkow, PSC has obtained cool suits which permit access to the building under harsh environment condition. The necessary manual actions will be identified and documented in appropriate procedures. A walkdown will be made to physically locate valves or equipment which require manual action as well as to verify that the operator can perform the manual action within the specified time frame.

ATTACHMENT 1

to P-85499

EQ BRANCH REQUEST NO. 1

Provide a description of the aging studies being performed and any interaction between equipment degradation and harsh environment accident conditions. (i.e., clarify the following statements: "The accelerated aging that occurs during the 4 minute isolation is very much different from that which occurs during a faster isolation time." "These lower temperature profiles will have a favorable impact in the areas of aging.")

PSC RESPONSE

FSV is committed to qualifying as a minimum all safe-shutdown equipment to IE Bulletin 79-01B. IE 79-01B specifically does not call for the establishment of qualified life by means of an accelerated aging program. Aging is addressed by analysis in conjunction with established material properties, relevant manufacturers data and testing performed on similar equipment.

The aging studies are summarized as follows:

a) Using regression line data and the Arrhenius Method, the end-of-life for age susceptible materials at ambient temperatures is calculated.

b) The thermal degradation that occurs during the high energy line break accident (HELB) and the post accident operability time is converted to an equivalent aging time at ambient temperatures.

c) The equivalent aging time calculated in b) is subtracted from the end-of-life determined in a), resulting in the replacement interval for the age susceptible material.

This approach is being utilized for all equipment being qualified to the requirements of the DOR Guidelines.

PSC believes there has been a misunderstanding regarding the terminology we have used in the past to describe our qualification efforts. We realize that typically aging refers to the process by which equipment is brought to its end of life state prior to HELB testing. By our use of the term "aging" we did not wish to infer that we were in any manner utilizing our existing DBE tests to account for the aging that occurs prior to the accident.

Since aging is a process of degradation due to thermal effects, we have in the past referred to all thermal degradation as "aging." The increased rate of degradation that occurs due to the HELB has in the past been referred to as "accelerated aging."

With this in mind, the two statements may be clarified.

"The accelerated aging that occurs during the 4 minute isolation is very much different from that which occurs during a faster isolation time."

The faster isolation time will result in a less severe accident profile since the shorter isolation time will yield smaller amounts of heat generated in the form of steam. Thus, the peak accident temperature will be less, and the subsequent bulk building temperature will return to ambient room temperature at a faster rate than it would for a 4 minute isolation time. By converting the different accident profiles to an equivalent aging time using the Arrhenius Methodology, it can be demonstrated that the faster isolation time results in a lower amount of thermal stress.

"These lower temperature profiles will have a favorable impact in the areas of aging."

The lower accident temperature profiles will have a favorable impact in regard to HELB qualification since the HELB test profiles contain a much higher degree of temperature margin versus the actual plant requirements.

EQ BRANCH REQUEST NO. 2

Provide a description of the operability studies being performed, specifically how operability is being demonstrated when test duration is less than required equipment post-accident operating time.

PSC RESPONSE

Qualification testing performed by the nuclear industry for operability during the accident simulation typically does not reflect real time vs. temperature profiles but rather accelerates the test conditions at elevated temperatures to a shorter than actual duration. A thermal equivalency analysis is then performed to verify tested conditions envelope plant requirements. In addition, operability of the equipment is verified to ensure the equipment performs its required safety functions.

A similar approach is being utilized fo. the FSV EQ program. The following summarizes this approach:

1) The post accident operability time has conservatively been assumed to be 30 days for all safe shutdown equipment.

2) The plant accident profile is compared with the test profile to ensure that the peak accident temperature including margin, is enveloped by the test profile.

3) Using the Arrhenius methodology, the equivalent thermal degradation that occurs during the post-peak accident profile is compared with the thermal degradation occurring during the same time period of the test profile.

4) Material properties are reviewed to ensure that no known phase changes occur in the temperature ranges in which the Arrhenius methodology is being used.

This approach of evaluating operability times will be performed for two separate scenarios: complete offset ruptures resulting in the highest peak temperatures, and fractional line breaks which result in lower peak temperatures for an extended time. PSC does not believe it is reasonable to assume simultaneous small and large line breaks, therefore, the two scenarios have not been enveloped by a single curve for operability time evaluation.

EQ BRANCH REQUEST NO. 3

Provide sample files (at least 3) which demonstrate how the above are being factored into the FSV EQ program.

PSC RESPONSE

PSC has previously submitted three (3) sample qualification packages to the Equipment Qualification Branch for staff review. NRC comments will be considered in preparation of FSV's files. Three updated files will be submitted under a separate cover letter when they are complete. PSC suggests a future meeting to discuss these files. As PSC proceeds with the FSV EQ program, we would suggest that additional files be forwarded for your review so that any problem areas can be identified early in the program.

EQ BRANCH REQUEST NO. 4

Provide assurance that the equipment within the scope of 10 CFR 50.49 is being qualified to the most limiting environment resulting from a spectrum of break sizes.

PSC RESPONSE

Attachment 3 to this letter describes the spectrum of break sizes analyzed in the FSV EQ Program. As stated in Attachment 3, the temperature profiles used for equipment qualification will represent the most limiting environment.

EQ BRANCH REQUEST NO. 5

Provide assurance that equipment accessibility is possible within the required time frame in the most limiting environment resulting from a spectrum of break sizes.

PSC RESPONSE

Attachment 3 to this letter describes the spectrum of break sizes analyzed in the FSV EQ Program. As stated in Attachment 3, the temperature profiles used to demonstrate equipment accessibility do represent the most limiting environment. As discussed in the recent PSC letter P-85460, human access can be accomplished when temperatures reach levels equal to the maximum temperatures achieved at one hour following the steam line breaks analyzed in Attachment 3 to this letter. Using the cool suits already purchased by PSC and described in P-85460, access into hot, moist areas with temperatures in the 180 degree Fahrenheit range is possible with the assistance of Scott Air-Pak breathing apparatus.

EQ BRANCH REQUEST NO. 6

Provide assurance that all design basis events. as defined in 10 CFR 50.49, have been considered in the determination of harsh environments.

PSC RESPONSE

PSC hereby provides assurance of the confirmation as documented in PSC response to NRC concern No. 8 (PSC letter P-85112, dated March 28, 1985, Warembourg to Johnson) that all design basis events have been considered in determination of harsh environments per 10 CFR 50.49.

Supplemental to those identified in the above response, the maximum credible accident (MCA) has also been considered. The MCA, as discussed in FSAR Section 14.8, is the result of a multiple failure involving the helium purification system regeneration piping. New analyses performed by GA Technologies, Inc. using the CONTEMPT-G code result in an average reactor building temperature rise of about 4 degrees Fahrenheit above the analyzed ambient temperature. This is not considered to be a harsh environment.

A rapid depressurization of the PCRV (Design Basis Accident No. 2) analyzed in FSAR Section 14.11, is the result of a sudden failure of both the primary and secondary closures of a PCRV penetration. Although the resultant peak building temperatures are higher than the peak average building temperature from the steam line rupture, the heat transfer coefficient of helium is much lower than that of steam. Because the duration of the DBA #2 accident is very short (typically 2 minutes), the equipment surface temperatures are less than those experienced for a steam line rupture. Therefore, the harsh environment to which the equipment will be qualified is bounded by the high energy line breaks.

AUXILIARY SYSTEMS BRANCH REQUEST NO. 1

Provide a detailed description of the steam line rupture detection and isolation system (SLRDIS). This discussion should include the systems design basis including its capability to assure environments within acceptable limits following steam line breaks concurrent with a single failure.

PSC RESPONSE

Please refer to Attachment 2 for the Steam Line Rupture Detection/Isolation System (SLRDIS) System Description.

The detection/trip system has been designed to meet the single failure requirements of IEEE 379-1977. It will be able to withstand a single failure and still provide its trip function.

During the generation of the steam line rupture temperature curves, a single active failure of a valve operator was assumed and is factored into the curves. Please refer to Attachment 3.

AUXILIARY SYSTEMS BRANCH REQUEST NO. 2

Confirm that previous pipe break analyses have addressed equipment qualification concerns for failures in systems other than the steam lines, e.g., main feedwater. Verify that these analyses have addressed protection from flooding.

PSC RESPONSE

As discussed in Attachment 3 to this letter, the pipe break analyses and resulting temperature profiles did address failures in systems other than steam lines.

Flooding caused by a line rupture is being analyzed separately for all cases. The analysis for the reactor building concludes that no flooding problem exists since the sump is large enough to contain any postulated leak. Required electrical equipment located in the sump is being moved. The turbine building sump however is not large enough and overflow would result. This is a result of an estimated 51,200 gallons being released during the first 6 minutes of a condensate line rupture with a continuing flow of approximately 6400 gpm after that. The service water sump return pumps are also assumed to fail since they are not qualified, adding another 5000 gpm maximum to the building. This combined flow results in overflow of the sump in about 10 minutes.

The turbine building is basically an open building, and an estimated 1000 gpm would leak from the building with all the doors closed. FSAR Section I.6.1 lists the allowable flood level in the turbine building to be 11 inches. This level would be reached in about 13 minutes after the condenser pit overflowed.

Within this total elapsed time of 23 minutes from the initiation of the leak until the allowable flood level is reached, an operator could open a door to avoid flooding of the building. Specifically, the east double doors could be easily accessed from outside the building and the doors swing outward. This 6 foot wide flow area would easily prevent flooding of the building. Also, the resulting building temperature due to a condensate rupture does not exceed 200 degrees Fahrenheit thus it is highly unlikely that the service water return pumps would fail, thus greatly reducing the inflow to the building.

AUXILIARY SYSTEMS BRANCH REQUEST NO. 3

Provide the results of an analysis of a spectrum of postulated breaks in the main steam, and hot and cold reheat lines in the turbine and reactor buildings. Include the resulting temperature profiles. Confirm that small breaks, i.e., those less than a full double ended break can be detected and isolated by the SLRDIS prior to exceeding the equipment qualification envelope or unacceptably preventing access for required manual actions to achieve shutdown.

In addition, provide response to the attached information request sheet in order to permit us to perform an independent calculation to verify your temperature profiles.

PSC RESPONSE

Attachment 3 to this letter provides the analysis results resulting from a spectrum of steam line break sizes. Attachment 3 identifies the steam line break sizes which the SLRDIS will detect and isolate and includes the composite temperature profiles. These composite profiles will be used in the FSV EQ Program to demonstrate equipment qualification and access required for manual actions to achieve shutdown. See the previous response to the EQ Branch Request No. 5 for a discussion of building access.

To permit independent calculations of temperature profiles, PSC has selected a large break scenario and a small break scenario for the reactor building which yield the peak temperatures overall and the maximum temperature at one hour. The large break scenario which gives the peak overall temperature for the turbine building was also selected. The requested information for those three scenarios is being assembled and will be forwarded under a separate cover letter in the near future.

AUXILIARY SYSTEMS BRANCH REQUEST NO. 4

Provide information on the capability of the SLRDIS temperature sensors to adequately detect elevated temperatures in the areas of concern.

Verify that the sensitivity of these sensors is sufficient to provide proper indication/actuation in the event of localized temperature effects following steam line breaks.

Include any available manufacturer's test data and/or performance information on similar detectors in comparable applications.

PSC RESPONSE

The thermistor cable temperature sensors are coaxial in design. A 20 AWG nickel center conductor is surrounded by a powder ceramic semi-conductor material which is then covered by an Inconel jacket. The outside diameter of the cable is .09" and weighs 6 grams per linear foot. The small mass of the cable enables it to respond quickly to changes in temperature. The thermistor material has the characteristic of exponentially decreasing resistance with higher temperature (negative coefficient of resistance). It is this change of resistance between the center conductor and the outside sheath that is monitored by the control panel. A change in temperature can be monitored anywhere along the length of the cable. The thermistor cable can withstand temperature extremes from -50 degrees Fahrenheit to 2000 degrees Fahrenheit. Because the primary parameter is resistance, the coaxial thermistor cable is able to monitor open circuit, short circuit, pre-selected temperatures and rate-of-rise.

Being all solid state, the sensors have only two failure modes - open circuit and short circuit. These conditions can be caused only by mechanical damage and are minimized by proper mounting. These two failure modes are continually monitored by the control panel. The thermistor cable has been exposed to radiation levels as high as 150 megarads, with no degradation in performance. This represents a radiation dose level many orders of magnitude higher than the design basis of Fort St. Vrain. The thermistor cable can continue to monitor temperature levels after generating pre-alarm and alarm signals and is the only thermistor type sensor approved by the Factory Mutual Research Corporation for fire protection.

The coaxial-type thermistor sensors have been specified and utilized in numerous applications such as on reactor coolant pumps and charcoal filters in nuclear power plants. Other uses include power cable trays, coal conveyers, cooling towers, power transformers and offshore oil platforms.

Over 39 domestic nuclear plants and 10 foreign nuclear plants are utilizing the thermistor cable as sensing elements in their fire detection and deluge control systems. The Commanche Peak Station is using the thermistor cable inside containment for heat detection below cable tray level. At the Davis Besse Nuclear Power Station, coaxial thermistor cable is used to monitor the temperature of the Hot Leg involved in Reactor Water Level System.

The sensors and control equipment have been tested and certified to meet IEEE Standards 323 and 344. The vendor's Quality Assurance program complies with 10 CFR 50 Appendix B.

The SLRDIS system control panel is designed to 'pre-alarm' at 160 degrees Fahrenheit (analysis value) and 'alarm' (send tripping signal) at 210 degrees Fahrenheit (analysis value). The control panel is programmed to respond to the corresponding equivalent resistances of these temperatures.

The thermistor cable temperature sensors change resistance, as previously stated, inversely and exponentially to temperature change. The detection panel generates a signal to the logic panel when the programmed level of resistance is recognized. Because of the exponential curve of resistance versus temperature, the detection panel is very selective, i.e., the elevated temperature to be detected presents a resistance much less in value than a temperature reading relatively close in magnitude. 150 degrees Fahrenheit for example presents a resistance of 115,000 ohms while 200 degrees Fahrenheit presents 30,000 ohms, and 300 degrees Fahrenheit equals 3500 ohms.

A test is currently being documented by Factory Mutual Research Corporation (FMRC) on the response time of the thermistor cable to rapidly changing elevated temperatures. The FMRC plunge tests are recognized as one of the most accurate methods of determining thermal response. The tests were conducted on December 10, 1985 at Norwood, Massachusettes. The test measured the resistance versus time response for nine combinations of gas temperature and velocity and will produce data for the environment specified at Fort St. Vrain. The preliminary statement from the vendor is that the results are satisfactory.

It should be noted that the temperature environment for a steam line rupture at Fort St. Vrain assumes a Bulk Temperature Model. The reactor building and turbine building will therefore see uniform temperature environments. No 'localized temperature effects' are considered. This means that the entire length of thermistor cable is assumed to sense the same temperature. Inasmuch as certain thermistor cables are closer to higher energy steam lines than others and that unequal heating will occur along the cable, the actual response of the cables will probably be faster than the simulated test results would indicate.

A combined 40 year accelerated aging and 20 megarad radiation exposure test shows little difference between the initial and final readings of resistance versus temperature on the thermistor cable. Heating and cooling does not affect the sensor's 1% repeatability.

Available Manufacturer's Test data includes the following:

1) Resistance vs Temperature curve of the 9090-13 thermistor cable (50' lengths used at Fort St. Vrain) Dwg. No. 280023 Rev. A

2) Sensor Center Conductor to Case Resistance vs Temperature - Initial and Final Curves after Accelerated Aging & Radiation Tests

3) Functional Test of Alison Control Panel After Seismic Test - dated 11/18/85

4) Equipment Qualification Package

Manufacturer's Test data still to be received:

1) Seismic Test of Control Rack & Sensor Assemblies performed by Wyle Labs, 11/9/85. (witnessed by PSC - all tests passed)

2) Response Time Test of Thermistor Cable performed by Factor Mutual Research Corporation, 12/10/85

Comparable Application Data:

1) Qualification Test Report of 9090-13 Sensor Assembly (Doc. NO. ETR101), dated 2/7/84 for Application at Davis-Besse

The above information is available for review at your request. The results for the response time test (item #2 above) will be submitted to the NRC staff when it is received.

ATTACHMENT 2

to P-85499