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PUBLIC SERVICE COMPANY OF OKLAHOMA
A CENTRAL AND SOUTH WEST COMPANY

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PSO

Public Service Company of Oklahoma
Black Fox Station Units 1 and 2
Docket Nos. 50-556 & 50-557
Responses to Generic Issues

December 11, 1981
File: 6212.125.3500.21L

Mr. Robert L. Tedesco
Assistant Director for Licensing
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Tedesco:

Enclosed you will find Public Service Company of Oklahoma's response to your letter of December 3, 1981, to Mr. G. W. Muench. In that letter you requested that Public Service Company provide updated information on ten (10) generic issues applicable to Black Fox Station, Units 1 and 2 (Docket Nos. 50-556 and 50-557). These issues were previously discussed with the NRC Staff in a meeting on this subject November 6, 1981.

In submitting the attached information, Public Service Company is cooperating with the NRC Staff to help them in their efforts to update the Generic Safety Issues Appendix to the Black Fox Station Safety Evaluation Report. This submittal does not contain new information, but rather directs the Staff to information already on the Black Fox Station docket and points out how it relates to these ten issues. Hence, we believe that by referencing existing information on the Black Fox Station docket, the Staff can update the generic issues for the SER supplement.

Sincerely,

John C. Zink

John C. Zink
Manager, Nuclear Licensing

JCZ:bjr
Enclosure
cc:BFS Service List

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1. BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle (A-10)

Description:

Generic Technical Activity (GTA) A-10 is concerned with cracking found in feedwater nozzles at several operating BWR's. The cracks have been discovered in the nozzle blend radius and bore region. The crack growth is slow but accelerates with increasing depth. It is possible that the cracks will present a repair problem if ASME code limits for nozzle reinforcement are exceeded during crack removal by grinding. Similar cracking has also been discovered on BWR control rod drive (CRD) return line nozzles.

Resolution:

Feedwater Nozzle Cracking-
Issuance of NUREG-0619 resolves Generic Technical Activity A-10. The NRC staff concluded that the GE triple sleeve sparger modification, when combined with the removal of stainless steel cladding, appropriate feedwater system modifications, and appropriate operating procedures will provide a substantial and acceptable improvement over previous designs. As described in the BFS PSAR Amendment 15, BFS will utilize the GE triple-sleeve sparger design as described in NEDE 21821-02, and NEDE-21821-A. The BFS reactor pressure vessels are not clad in the nozzle area. According to the NRC Safety Evaluation Report, included as Appendix C to NUREG-0619, the triple-sleeve sparger design may be used without further justification beyond that given by GE. As for Ultrasonic Testing and Inspections, the Staff conclusions in Sections 6 and 7 of Appendix C will be used as guidance in developing inspection and testing procedures for BFS. New testing techniques will be examined for applicability as they are developed.

Control Rod Drive Return Line Nozzle -
The control rod drive return has been deleted and no CRD return line nozzle has been provided on the BFS reactor pressure vessels, as permitted by Part II, Section 8.1(4) of NUREG-0619. This design change information is also included as part of Amendment 15 to the BFS PSAR.

2. Residual Heat Removal Requirements (A-31)

Description:
Task A-31 investigated the ability of the Residual Heat Removal (RHR) system to adequately bring the plant to a cold shutdown condition. This task was resolved in 1978. Requirements that were developed were reflected in a revised Standard Review Plan (SRP) Section 5.4.7 and BTP 5-1.

Resolution:
The NRC Staff concluded in the BFS SER June, 1977, (Section 5.4.5) that BFS design will conform to Criteria 19 and 34 of the General Design. Criteria, i.e., "the plant will have Seismic Category I systems capable of bringing the plant to cold shutdown within approximately 24 hours...."

Standard Review Plan 5.4.7 and Branch Technical Position 5-1 require that BFS be designed in conformance with GDC 2, 5, 19 and 34. BFS design is presented in Sections 3.1.2 of the BFS PSAR and Sections 3.1.2, 5.5.7.2 and 15.1.27 of GESSAR.

3. Control of Heavy Loads Near Spent Fuel (A-36)

Description:

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in PWR's and BWR's. If a heavy object, e.g., a spent fuel shipping cask or shielding block, were to fall or tip on to spent fuel in the storage pool or the reactor core during refueling and damage the fuel, there would be a release of radioactivity to the environment and a potential for radiation over-exposures to inplant personnel. If the dropped object is large, and the damaged fuel contained a large amount of underdecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines. These requirements are currently considered in the licensing review. However, with the advent of increased and longer term storage of spent fuel assemblies in the spent fuel pools, there is a need to systematically review NRC requirements, facility designs, and technical specifications requiring the movement of heavy loads to assess safety margins and to improve those margins where warranted.

Resolution:

The Black Fox Station spent fuel storage, spent fuel cask handling, and fuel handling systems are described in Section 9.1 of the PSAR. These systems were reviewed by the NRC staff and judged acceptable with respect to the control of heavy loads near spent fuel, as documented in Section 9.2 of the BFS SER.

4. Pipe Cracks in Boiling Water Reactors (A-42)

Description:

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in boiling water reactors since the mid-1960's. These cracks have occurred mainly in Type 304 stainless steel that is being used in most operating BWRs. The major cause of this problem has been determined to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components. These components have been made susceptible to this failure mode by being "sensitized" in the narrow heat-affected zone during the welding process.

Resolution:

The NRC Staff resolution of Generic Technical Activity A-42 "Pipe Cracks in Boiling Water Reactors" is presented in NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." Section II provides guidance to the methods to minimize susceptibility to intergranular stress corrosion cracking for BWR ASME Code Class 1 and 2 piping systems. Black Fox Station complies with these methods in the following manner:

a. All ASME III Code Class 1 and 2 piping having a sustained service temperature above 200 degrees F will be carbon (ferritic) steel or type 304L or 316L austenitic stainless steel or "Nuclear Grade" austenitic stainless steel. (PSAR (GESSAR) Section 5.2.3.2, 5.2.3.2.1.3 and PSAR Table 1.9-1, p.1.9-90.) PSO has selected "Nuclear Grade" austenitic stainless steel for recirculation system piping to mitigate intergranular stress corrosion cracking. (Affidavit of John B. West before the Atomic Safety and Licensing Board, November 5, 1981.)

b. Austenitic stainless steels will be procured in the solution-annealed condition and tested in accordance with ASTM A-262 to assure that they were properly annealed and are not susceptible to intergranular stress corrosion cracking. (PSAR (GESSAR) 5.2.3.2 and PSAR Table 1.9-1.)

c. Valves and fittings will be of a material which is compatible with the piping.

d. Shop fabricated Reactor Recirculation System piping will be solution annealed after welding. (PSAR (GESSAR) Section 5.2.5, 5.2.3.2.1.3 and Table 1.9-1.)

e. Field welded joints in austenitic stainless steel which cannot be solution heat treated will be made using heat input controls, and filler metal will be selected which will produce a weld having a five percent minimum ferrite content. (PSAR Table 1.9-1 pp. 1.9-6.)

5. Containment Emergency Sump Reliability (A-43)

Description:

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of the containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reached a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggrevated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools, and will also be addressed.

Resolution:

The concern that insulation failure could result in debris blocking the ECCS system and RCIC system pump suction has been recognized in the Black Fox Station design. Measures to mitigate the blockage problem are discussed in the BFS PSAR (GESSAR) Section 6.3.2.2.7. Additional details are:

a. All ECCS pumps suction lines and RCIC pumps suction lines will be provided with tee-type strainers. (PSAR) (GESSAR) Section 5.2.3.3).
b. Each strainer has a minimum inlet area of at least 200 percent of that required to satisfy the pump. (PSAR (GESSAR) Section 6.3.2.2.7).
c. The strainers are located 5 feet 7 inches above the bottom of the pool. Approach velocities are low to prevent plugging. (PSAR (GESSAR) Section 6.3.2.2.7).
d. The various pump strainers are diversely located in the suppression pool to minimize the chance of disabling multiple systems from local debris.

In addition, ECCS and RCIC equipment specifications will include requirements to provide the capability for the equipment to function in the presence of particles which can pass through the suction screens. (BFS PSAR Section 6.3.2.2.7.)

BFS ECCS and RCIC suction strainers are designed to have a submergence of greater than 8 feet with respect to the centerline of the ECCS pump suction piping. This submergence, coupled with a low intake velocit , is designed to preclude vortex formation. (PSAR (GESSAR) Section 6.3.2 2.7, PSAR Figure 6.2-32).

6. Station Blackout (A-44)

Description:

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such, it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely but possible loss of all A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

Resolution:

The design of the BFS electrical system assures that there will be a source of electrical power for safe shutdown of the reactor. Should there be a loss of both offsite and onsite alternating current power BFS may use a combination of the safety/relief valves and the RCIC system to remove the decay heat without reliance on alternating current power. This allows time for restoration of alternating current power from either offsite or onsite sources.

A loss of offsite alternating current power at BFS involves a loss of two preferred power sources to each division ESF bus. One circuit connecting the preferred power source to the Division 1, Division 2, and Division 3 ESF buses for each unit is from the 345 kV system through the main generator transformer bank, the main auxiliary transformers and the plant normal auxiliary power distribution system. The normal preferred power system configuration includes a second completely independent circuit connecting the Division 1, Division 2, and Division 3 ESF buses for each unit to the 138 kV system through the reserve auxiliary transformer and an independent distribution system. (PSAR Sections 8.1 and 8.2.).

If offsite alternating current power is lost, three diesel-generators and their associated distribution systems will deliver emergency power to safety related equipment. The normal preferred power source is continuously monitored at each ESF 4.16 kV bus by voltage relays; and, if a bus undervoltage is detected, the normal preferred source circuit is automatically disconnected, and the diesel generator is automatically started and brought to rated speed and voltage. If the normal preferred source is disconnected from a 4.16 kV ESF bus and the alternate preferred source is proven available by the voltage relays, the ESF bus will be automatically connected to the alternate preferred source approximately 2 seconds after bus undervoltage is detected; the diesel generators remain running at no load until operator action is taken to shut them down. If the alternate AC preferred source is not proved available, the diesel-generator will be automatically connected to its respective 4.16 kV ESF bus when the diesel-generator reaches rated speed and voltage. The standby power source for each engineered safety feature bus is the diesel generator connected exclusively to that bus. There is one independent and separate diesel generator for each of the three ESF divisions. (PSAR Section 8.3.)

The Class 1E DC System is comprised of four independent (Division 1-4) 125 volt DC systems. Each division is physically separated to assure that no single credible event will prevent the operation of the required number of redundant fuctions. The function of the Class 1E subsystem of the DC System is to furnish highly reliable 125-volt DC power for control, to power loads, and to power instrumentation for equipment that limits the release of fission products and maintains safe plant conditions. Each division of the DC System is comprised of a primary DC supply (charger) powering distribution equipment, with the backup DC supply (battery) "floating" on the bus during normal conditions. Each battery charger is fed from the Standby AC Power Supply System. (PSAR Section 8.3.)

Maintenance and testing programs will be implemented in accordance with detailed design and individual equipment qualification test results. The design accommodates these programs to assure the readiness of these systems to deliver the performance required. (PSAR Sections 8.3.1 and 8.3.2.)

7. Shutdown Decay Heat Removal Requirements (A-45)

Description:

Task A-45 will investigate the need for possible design requirements to improve the reliability of decay heat removal systems in BWR's and PWR's.

Resolution:

Black Fox Station has several ways for removing decay heat. The normal method is via the steam lines to the main condenser. The condensate is returned to the reactor by the Condensate and Feedwater Systems. In the event the Feedwater System is not available, the RCIC and/or the HPCS Systems can supply the required water from the condensate storage tanks. (PSAR (GESSAR) Section 10.4, 5.5.6 and 6.3.2.)

If the condenser is not available, decay heat can be removed by operating the RHR in the steam condensing mode and returning the condensate via the RCIC System. Alternately, the heat may be removed without depressurizing the reactor by cycling the safety-relief valves (SRV) which discharge to the suppression pool, and returning suppression pool water to the reactor vessel using the RCIC or HPCS. (PSAR Sections 5.5.7 and 6.3.3.)

If the RCIC and HPCS are not available, decay heat may be removed by depressurizing the reactor using the safety-relief valves in the ADS mode and controlling reactor water inventory by using the RHR or LPCS. (PSAR Section 6.3.2.)

The RHR, LPCS, and HPCS are safety related systems which use motor driven pumps powered by on-site safety related diesel generators. The RCIC uses a turbine driven pump with control power supplied to the RCIC from a safety related on-site DC power source. (PSAR Sections 5.5.7 and 6.3.2.)

All decay heat removal modes which use the suppression pool as an intermediate heat sink, require the eventual operation of the RHR in the suppression pool cooling mode.

8. Seismic Qualification of Equipment In Operating Plants (A-46)

Description:

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this "Unresolved Safety Issue" is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

Resolution:

This unresolved safety issue is primarily directed to the programs for seismic qualification of equipment in operating plants. Black Fox Station is designed to current seismic criteria. The seismic design and requirements for mechanical and electrical equipment and instrumentation are presented in Chapters 3 and 7 of the BFS PSAR.

Seismic classification of structures, systems and components important to safety is discussed in Section 3.2 of GESSAR for the NSSS and in Section 3.2 of the BFS PSAR for the Balance of Plant. Seismic design of BFS is addressed in Sections 3.2.1, 3.9 (Mechanical Equipment), 3.10 (Electrical and Instrumentation and Controls) and 3.8 (Structures).

Compliance with the requirements of IEEE-344-1975 are discussed in Section 3.10.1.3 of the BFS PSAR. The NRC Staff has concluded in the BFS SER Section 3.10 that the seismic qualification program described by the applicant is consistent with IEEE-344-1975 and with Regulatory Guide 1.100, and is an acceptable basis for satisfying the applicable requirements of GDC-2.

The NRC Staff has reviewed the procedure proposed for dynamic testing and analysis to confirm the adequacy of seismic category 1 mechanical equipment, including their supports, to function during and after an earthquake of magnitude up to and including the safe shutdown earthquake. The NRC Staff concluded that implementation of these procedures constitutes an acceptable basis for satisfying the requirements of GDC-2 and 14. (BFS SER 3.9.12)

PSO has proposed the use of absolute seismic methodology for combining dynamic response for the design of structures and the square-root-sum of squares (SRSS) methodology for all plant systems and components when the Newmark-Kennedy criteria are satisfied. The NRC Staff found the use of absolute sums methodology for structures satisfactory, but concluded the use of SRSS for systems and components did not meet the more restricted use of SRSS proposed in NUREG-0484. Subsequent to the close of the BFS Safety Hearing record, the NRC Staff issued Revision 1 to NUREG-0484 which significantly extended the staff acceptance of the SRSS methodology for combining responses. The use of SRSS methodology, however, remains an open issue before the Atomic Safety and Licensing Board.

9. Safety Implications of Control Systems (A-47)

Description:

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system, and the effects of the control system falures, may differ from plant to plant. Therefore, it is not possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that will be used for plant-specific reviews. A specific subtask of this issue will be to study the reactor overfill transient in Boiling Water Reactors to determine the need for preventive and/or mitigating design measures to accommodate transient.

Resolution:

The Black Fox Station control and safety systems have been designed with the goal of ensuring that control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident". This has been accomplished by either providing independence between safety and nonsafety systems or providing isolating devices to preclude the propagation of nonsafety system equipment faults such that operation of the safety system equipment is not impaired.

A systematic evaluation of the control system design, such as contemplated for this "Unresolved Safety Issue", has not been performed to dtermine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, as described in Chapter 15 of the Black Fox Station PSAR, a wide range of bounding transients and accidents has been analyzed to assure that the postulated events would be adequately mitigated by the safety systems.

In recognition of the on-going concern pertaining to the reactor overfill transient, the Black Fox Station design incorporates a commercial grade trip of the RCIC, HPCS, and feedwater systems to prevent the occurrence of this transient.

In addition, as described in Addendum II to the Black Fox Station PSAR, a Black Fox Station Reliability Analysis Program will be developed with objectives to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the station. The program will incorporate environmental effects, system interactions, human error and performance data, interdependence of support systems and system unavailabilities in the event tree/fault tree analysis. (PSAR Addendum II, pp. 1-6)

Changes in the design of control systems can be accommodated prior to the issuance of the operating license since instrumentation design is normally completed in the latter stages of plant construction.

10. Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

Description:

Following a loss-of-coolant accident in a light water reactor plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of an accident, 10 CFR Section 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and Criterion 41 of the General Design Criteria, "Containment Atmosphere Cleanup," in Appendix A to 10 CRR Part 50, require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

The regulation, 10 CFR Section 50.44, requires that the combustible gas control system provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system such that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

The accident at TMI-2 on March 28, 1979 resulted in hydrogen generation well in excess of the amounts specified in 10 CFR Section 50.44. As a result of this knowledge it became apparent to NRC that specific design measures are needed for handling larger hydrogen releases, particularly for smaller, low-pressure containments. As a result, the Commission determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be taken into account in plant design. An advance notice of this rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980.

Resolution:

The BFS Containment Combustible Gas Control System designed to meet the requirements of 10 CFR 50.44 is described in the BFS PSAR Section 6.2.5. This system was reviewed and found acceptable by the NRC Staff (BFS SER Section 6.2.5).

In response to the NRC requirements arising from the TMI accident, PSO submitted Amendment 19 to the BFS PSAR. Amendment 19 describes the PSO Hydrogen Control Program, including preliminary design parameters, a preliminary design of a hydrogen control system utilizing distributed igniters and committing to a continuing evaluation of the hydrogen control system with final evaluation and selection of the system to be completed and submitted for NRC review within two years of the issuance of a construction permit. (PSAR Addendum II, pp. 279-392.)