SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. DPR-34
PUBLIC SERVICE COMPANY OF COLORADO
FORT ST. VRAIN NUCLEAR GENERATING STATION
DOCKET NO. 50-267

INTRODUCTION

In late January, 1975, when the Fort St. Vrain reactor was taken critical following an extended shutdown, a reactivity discrepancy was noted. Investigation revealed that water had entered the lower part of the reactor vessel while the plant was shut down and that high moisture in the coolant had contributed to the discrepancy. In late February, 1976, while the reactor was shut down, water again entered the vessel, but the amount was minor. As a result of these occurrences, the Public Service Company of Colorado (PSCo) has conducted a comprehensive evaluation of the cause of the water ingress, modifications necessary to minimize the potential for recurrence, an assessment of potential damage to reactor components, and modifications necessary to improve performance of the moisture monitoring systems. In addition, a number of changes to the Technical Specifications have been proposed.[6][7][13][23] The NRC staff has completed its evaluation of these events and concluded that resumption of operation under the conditions proposed, and with modifications as discussed herein, is acceptable. The basic documents which have been considered in this evaluation, are listed in the enclosed list of references.

BACKGROUND

On January 21, 1975, the Fort St. Vrain reactor was taken critical for training purposes and a negative reactivity discrepancy of between .004 and .007$\Delta\rho$ was noted. The discrepancy was within the Technical Specifications limit of 0.012$\Delta\rho$, and it was attributed at the time to higher than expected core temperature, since the helium circulators were not providing forced convection flow. On the following day the reactor was taken critical,

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again with circulators idle. A negative reactivity discrepancy of about .008$\Delta\rho$ was noted. On January 23, 1975 two circulators were operated to establish a known core temperature and the reactor was taken critical. The reactivity discrepancy was still negative, but as time passed the discrepancy turned positive, reaching a value of +0.008$\Delta\rho$. Since the total change was in excess of Technical Specification limits on unexplained reactivity changes, the reactor was scrammed.

Through investigation it was determined that the negative reactivity discrepancies noted on January 21 and 22 were caused by the presence of reserve shutdown material which had been inadvertently discharged from one of 37 hoppers into the core during maintenance. The positive reactivity discrepancy of January 23 was caused by the moderating effect of increasing moisture adsorption on core graphite as the helium circulators transported moist helium from a water source in the lower part of the reactor vessel. Subsequent investigation revealed that about 4250 gallons of water had entered the reactor vessel via one or more of the non-operating helium circulators in late December 1974 or January 1975 while the reactor was shut down with circulator water drain lines closed to facilitate maintenance operations.

During the period of January 21 through January 23, neither the dewpoint monitors in the plant protection system (DPMMs) nor the "Analytical" moisture monitors in a separate system gave an indication of abnormally high moisture levels in the reactor coolant. The DPMMs were set to trip at a selected dewpoint level, but trip did not occur. When two of the DPMMs were switched from their "trip" mode to the "indicate" mode to give direct readout of dewpoint levels, a discrepancy between the Analytical instruments and the DPMMs was apparent, and a sample of coolant was taken for analysis in a gas chromatograph. This analysis indicated a moisture level of over 10,000 ppmv, several thousand ppmv over the levels shown by either the DPMMs or the Analytical instruments. This confirmed the presence of abnormally high moisture in the reactor vessel and initiated the extensive investigation cited previously.

Dry-out of the reactor system was accomplished via the helium purification system, initially with helium coolant at positive pressures, with final dry-out being accomplished by drawing a vacuum of less than 10mm Hg. By mid-March 1975, the reserve shutdown material had been removed from the core and moisture levels in the reactor coolant had returned to a normal operating range. A critical run on March 17, 1975 confirmed that the reactivity discrepancy was no longer present.

On February 25, 1976, while the reactor was shut down and depressurized, water again entered the reactor vessel via the helium circulators. This second occurrence was detected in time to limit the ingress to about 200 gallons. Subsequent dry-out has been sucessfully accomplished.

The NRC evaluation of these events, the remedial measures that have been taken and the associated Technical Specification changes that are required is presented in following sections.

EVALUATION

Water Ingress

Review of plant records and tests conducted by PSCo have eliminated the steam generators, the liner cooling system for the prestressed concrete reactor vessel (PCRV), the helium transfer compressor and the helium purification system as possible sources of the water in the reactor vessel. The helium circulators were then singled out as the most probable source.

During February 1975, Gene[illegible]tomic conducted tests at its San Diego Laboratories using a spare [illegible]rculator to determine under what conditions during circulator shutdown and startup bearing water in-leakage could occur. These tests clearly show that circulator drain valves would have to be abnormally restricted or that an unusual pressure imbalance would have to occur for water to rise up the shaft and into the PCRV during circulator startup and operation. Tests with the circulator shutdown and the mechanical seal set also showed that with drain valves closed sufficient pressure could be developed to lift the seal and admit water to the reactor vessel if a sufficiently high pressure source of water were admitted to the circulator cavity.

During December 1974 and early January 1975, various circulators were in an abnormal configuration occasioned by replacement of Pelton wheels which are used as an emergency backup drive to the normal steam turbine drives. Review of plant records indicated that in early January while two of the circulators were blind-flanged from the bearing water system, the emergency feedwater system was pressurized. Although this system was isolated from the Pelton wheel cavities by closed valves, it is strongly suspected that leakage occurred. All of the water may have entered the reactor vessel during this period. There were also periods of time when the other two circulators were similarly isolated and vulnerable to such inleakage.

During mid-January 1975, difficulty was experienced in putting some of the circulators into operation. Water accumulated in instrument sensing lines, buffer helium alarms occurred and differential pressures were such that bearing water could have entered the vessel. Circulator speeds were occasionally less than anticipated. Review of gas chromatograph records for mid-January also indicated the possible presence of high moisture.

These anomalous indications and circulator startup difficulties are now recognized as symptomatic of water control difficulties, and plant operating procedures have been changed accordingly, to assure that drain lines are kept open when circulators are idle and to assure that abnormal conditions are recognized and corrective actions are taken promptly. In addition, level indicators and Hi Level Alarms have been installed in the Pelton turbine drain cavities of each circulator. Pressure differential instruments have been installed to directly verify that bearing water surge tank pressures are lower than reactor pressure during circulator startup. Drain pots for buffer helium instruments have been relocated to a low point and equipped with level alarms. Setpoint of flow control for the high pressure buffer helium water separator has been relocated to the control room.

The water ingress event of February 25, 1976 occurred while the reactor was shut down and depressurized. Two circulators were being operated in the self-turbining mode and buffer helium was being supplied by high pressure bottles rather than by the helium purification system which is normally used during reactor operation. Power was inadvertently disconnected from a pressure controller, allowing valves in the supply and recirculation lines to fail-open and admit 80 psig helium to the helium dryer outlet. This upset the helium buffer system by preventing buffer helium recirculation and interfering with drain of bearing water to the high pressure separator. As a result, about 200 gallons of water rose up the circulator shafts into the PCRV because of reversed pressure differentials during the ensuing 35-minute period while corrective actions were being attempted. This occurrence is indicative of the complex interactions that can take place in the circulator auxiliaries as the result of single failures. Until further operating experience proves to the contrary, it must be anticipated that water ingress can occur via the circulators.

To reduce the chance of recurrence, PSCo has implemented a change in the control valve in the line from the helium storage system so that loss of controller power will result in valve closure. PSCo has also initiated a Technical Specification change which will require that valves V-23224 and V-23221 be placed in the closed position to isolate the helium storage system from the helium circulator buffer helium system when the reactor is in operation.

The procedural measures and equipment modifications taken to date give increased assurance that substantial amounts of water will not enter the reactor vessel via the helium circulators in the future. The consequences of water ingress via the circulators or other pathways were previously analyzed and found acceptable during operating license review. Improvements to the moisture monitoring systems, as discussed in a following section, give assurance that such ingress will be readily detected during reactor operation.

Effects of Water Ingress on Components

During the periods of January through March 1975 and February and March of 1976, while substantial moisture was in the reactor vessel, temperature and oxygen levels were too low to allow significant amounts of corrosion to metallic materials or reaction between water and graphite to cause dimensional changes or loss in strength. Tests have demonstrated that the properties of fibrous insulation on the reactor vessel walls are not affected by soaking and subsequent dryout. The absence of significant corrosive action on metallic components has been verified by direct examination of a circulator and its penetration area and of several orifice assemblies and control rods, their drives and housings. Control rod and orifice assemblies have been tested and found to function normally. We conclude that none of these materials and primary system components were adversely affected.

Following the ingress event of 1975, several control rod position potentiometers were replaced because they appeared to be marginal owing to local electrolytic corrosion that occurred during the period of high moisture. Following the 1976 ingress, power to the control rod drive and orifice assemblies was removed to prevent such corrosion, and subsequent tests have demonstrated that position instrumentation functions normally. To provide further verification that deterioration of potentiometer circuitry has not occurred, PSCo will perform insulation resistance tests prior to startup. Reserve shutdown material removed from the core was found to be unaffected by the moisture and it was re-used. During seal modifications to control rod drive assemblies in September 1975, a white crystalline deposit was found on some of the boron carbide balls in the reserve shutdown hopper associated with the mechanism undergoing modification. Inspection of several other hoppers revealed a similar deposit. Analysis has shown that this deposit consists of crystals of boric acid.

Depending on the reactor region they serve, these balls consist of 20 to 40% boron in the form of B_4C in a graphite matrix. They were manufactured to have a specified maximum B_2O_3 content of 0.15%. This oxide will hydrolize in the presence of moisture to form crystals of boric acid (H_3BO_3) or anhydrous boric acid (HBO_2), depending on temperature.

During the period of January-March 1975, when abnormally high moisture was present in the reactor coolant, conditions were favorable for moisture condensation in the reserve shutdown hoppers, since they are located in a relatively cool region in the upper head of the reactor vessel and they are vented to the reactor coolant. During this period it is clear that such condensation did occur and that a fraction of the B_2O_3 in the balls leached out and redeposited as crystalline boric acid when the system was dried out.

From the neutronic standpoint, the reserve shutdown system is unaffected. B_4C in the balls was unaffected since temperatures were far too low to cause oxidation, and the carbide form is not soluble in water to a measurable extent. Since the balls, with their loading of B_4C are neutronically "black" to thermal neutrons, the negative worth of the reserve shutdown system is the same as before B_2O_3 leaching occurred.

If the reserve shutdown system were discharged into the core, and the balls were subsequently removed, some of the crystalline material might be left behind. It is possible that a negative reactivity effect might be detected during subsequent reactor startup, by a small deviation from predicted control rod position. However, the magnitude of the reactivity effect would be highly unlikely to exceed $0.003\Delta\rho$, which is less than one-third of the definition of an anomalous reactivity occurrence as set forth in the Technical Specifications. Reactivity increase, owing to depletion of residual boron by neutron absorption, or relocation of the residual material under high moisture conditions, would be gradual and easily accommodated by normal control rod adjustment. We conclude, therefore that residual boron left in the core following discharge of the reserve shutdown system will not have a detrimental effect on subsequent operation of the plant.

Although boric acid solution is mildly corrosive to carbon steel, no evidence of corrosion was found on reserve shutdown hoppers. In the dry state, crystalline boric acid is not corrosive. Therefore a corrosion potential in the hoppers does not now exist. B_2O_3 does not react with any of the materials used in the reactor system at the operating temperatures involved. Thus there would be no deleterious effects even if some boric acid were left in the core, as postulated above, and were subsequently relocated as B_2O_3 to the various high temperature parts of the system. If hydrolysis and solution of relocated boric acid were to occur as the result of some future water ingress event, corrosion of ferritic materials would be negligible if low oxygen concentrations were maintained in the helium coolant. If not, some minor pitting corrosion could occur. While inspection of selected components would be in order following such a condition, loss of serviceability would not be anticipated owing to the very small quantity of boric acid involved.

The only potential concern over the presence of the boric acid deposit, would be the possibility of adhesion of the balls in their hoppers to the extent that discharge capability would be affected. Laboratory tests, in which crystals were grown under conditions representative of those under which the deposits were formed in the hoppers, indicate that the bond strength of the deposits is negligible, and that the potential for adhesion should be very small. In addition, one of the hoppers was removed from the reactor and discharged in a test fixture. This hopper discharged normally, indicating that the boric acid crystals had no effect on operability. Based on these tests, we conclude that the reserve shutdown system is capable of carrying out its design function.

In the period of time during and following the ingress event of February 25, 1976, records show that temperature of the coolant and of the PCRV liner cooling water system were maintained well above the coolant dewpoint temperatures. Thus condensation on the surfaces of components in the system, including the reserve shutdown hoppers, was prevented. However, should reactor conditions at some future time be such that moisture condensation can again occur in the reserve shutdown system hoppers, the reactor should not be operated until it is reconfirmed that functional capability of the system has not been affected. The NRC will review any future abnormal occurrence reports concerning high primary coolant moisture levels to determine whether condensation has occurred necessitating corrective action.

We conclude that the measures taken to date will reduce the chances of recurrence of water ingress via the helium circulators, that the materials in the reactor system have not been damaged, that the operability of control rod drive assemblies and the reserve shutdown system have been verified, that moisture levels in the primary system have been restored to a normal operating range and that from these standpoints the reactor is ready for resumption of operation.

Potential Effects of Future Operations on Fixed Burnable Poisons

Although we have concluded that the moisture ingress event of January 1975 has had negligible effect on core materials, we have reexamined the potential effects of moisture levels permitted during variou- operating conditions by the Fort St. Vrain Technical Specifications on the B_4- fixed burnable poison material in the core. In this connection PSCo has responded to a number of specific questions and has subsequently proposed modifications to the Technical Specifications to improve reactivity surveillance.

Our concern was that under operating conditions permitted by the Technical Specifications, the fraction of B_4C fixed burnable poison that might be oxidized to B_2O_3 in the presence of H_2O might be greater than predicted. Of itself such oxidation is of no concern, since the B_2O_3 would remain in place at operating core temperatures. However, if an appreciable fraction of B_4C were oxidized before it was depleted by neutron absorption, and the reactor were shut down and then restarted under high moisture conditions, some of the B_2O_3 might be relocated by steam distillation with an attending positive reactivity effect.

Our concern with B_4C oxidation and hydrolysis centered on (1) what appeared to be inconsistencies between the bases for FSV Technical Specifications LCO 4.2.10 and LCO 4.2.11 and the intent and wording of the Technical Specifications and (2) questions related to the GA experimental and theoretical foundation for the B_4C oxidation rate equations.

(1) Under LCO 4.2.10 the reactor can be operated indefinitely with average core outlet temperatures $\geq$ 1200°F and with chemical impurity concentrations, total oxidants, (i.e., $H_2O + CO + CO_2$) $\leq$ 10 vppm, but the analysis of boron oxidation, presented in PSC

of Colorado Quarterly Progress Report, GA-10560[1] was performed on the basis of 1 vppm H_2O, not 10 vppm.

(2) The coolant impurity levels allowed under LCO 4.2.11, at average core outlet temperatures <1200°F, were based on the criterion that not more than 10% of the beginning of life (BOL) B^{10} loading can be present as oxide over a refueling cycle. Using the equations and analyses presented in the GA Fuels Group Report[2] and the PSC progress report, it appeared that this criterion might not be met.

(3) The original GA analysis, leading to the conclusion that the oxidation rate of B_4C is water-transport-limited, did not take into account a water ingress of the magnitude subsequently experienced, as evidenced by the statement in the cited progress report that "the initial water content of the active core is expected to be 30 lbs. or less".

(4) The experiment on which the B_4C oxidation analysis was based consisted of a series of conservative measurements on one piece of lumped, burnable poison rod (0.67" long), whose composition relative to FSV-type lumped burnable poison (LBP) was unknown. Conditions under which the reaction rates were measured were not representative of reactor coolant pressures and flow rates. The B_4C oxidation rates determined from this test required the use of several modifying expressions before application to HTGR conditions. Therefore, these modifying expressions required the use of assumptions regarding rate-limiting oxidation mechanisms.

These concerns were addressed by PSC,[4] in response to our questions.[3] In the response, revised curves for B_4C oxidation were presented. The revised curves were based on an altered combination of diffusive and permeation flow, a measured (lower) value of diffusion in H-327 graphite, and lower values of permeation flow (due to a lower ΔP at reduced power levels). For 1 vppm, using the revised curves, the B_4C oxidation rate was calculated to be 5.4%/yr at 700°C (for transport-controlled oxidation) and 5.5%/yr at 600°C (for reaction rate controlled oxidation). At 1200°F (654°C) and 10 vppm H_2O (permitted under LCO 4.2.11), the calculated B_4C oxidation rate was 42%/yr and 60%/yr in the reaction-rate and transport controlled regimes, respectively; operation at these levels would be limited to 90 days per fuel cycle by LCO 4.2.11. As noted in the PSC response to our questions, the actual fraction of the total oxidants that will be present in the form of H_2O is difficult to specify precisely because the distribution of oxidants is dependent on the graphite chemical reactivity, which in turn is affected by the catalytic effect of impurities and fission products in the graphite, and the previous oxidation history. According to the best current calculations by GA, less than 10% of the total oxidants will be present in the form of H_2O, but operating experience in FSV is needed to confirm this.

In summary, we conclude that the predictions of B_4C oxidation rates under the conditions outlined are reasonable. However, they are based in part on a combination of theoretical and empirical information, for which confirmatory experience is needed to assure complete confidence.

In this regard, PSCo has indicated that they will perform post-irradiation-examinations (PIE) of burnable poison rods to determine the extent of B_4C oxidation and depletion and to verify the GA analysis and out-of-pile experimental results. This PIE will be performed at the earliest refueling outage when a fuel block from the upper one-third of the core and a block from the lower one-third become available, since it has been predicted(5) that B_4C oxidation can only occur in the upper third where temperatures are lowest. PSCo has initiated a Technical Specification change at the request of NRC to add a requirement to report the PIE examination to NRC.

The oxidation of boron carbide by itself will have no effect on the reactivity of the Fort St. Vrain core. Conversion of the oxide to boric acid and subsequent volatilization of this material out of the fuel-graphite matrix could increase core reactivity if the amount of material involved were large enough. To assure that PSCo's reactivity surveillance program is adequate to detect any anomalous reactivity changes we have reevaluated pertinent sections of the Technical Specifications which are part of the operating license.

Technical Specification LCO 4.1.2 requires that sufficient rod worth be available to ensure that cold shutdown can be achieved by a margin of $0.01\Delta\kappa$ with all rod pairs, but the most reactive, inserted. This margin is adequate. Results of calculations contained in the FSAR indicate that expected shutdown margin (assuming that the highest worth rod pair fails to insert) will be in excess of $0.036\Delta\kappa$ at all times during operation of the first cycle core. Assuming an uncertainty in the shutdown margin calculation of $0.016\Delta\kappa$, reduction in shutdown margin of at least 0.01Δκ may occur before the shutdown margin can fall below 0.01Δκ.

As originally stated, Specifications LCO 4.1.8 and LCO SR 5.1.4 required that the reactor be shut down if expected reactivity deviates from observed reactivity by a difference of 0.012Δκ. In addition, it was not clear whether renormalization of base reactivity could be done at some point in the fuel cycle, thus obscuring the observed difference.

To correct these discrepancies, PSCo proposed changes to LCO 4.1.8 and SR 5.1.4 by letter[6] dated September 11, 1975. As proposed, a difference of 0.01Δκ between predicted and observed reactivities, based on normalization to a base steady core condition, will require shutdown until a satisfactory explanation for the anomaly is found and permission is received from PSCo's Nuclear Facility Safety Committee (NFSC) to resume operations. In addition, any renormalization of this base, as approved by the NFSC, will be reported immediately to NRC. These changes provide acceptable means for detection of anomalous reactivity changes, including changes that could potentially occur owing to loss of burnable poison from the core. In addition, NRC will be provided with data for review as may be deemed necessary.

By letter dated March 23, 1976[7] PSCo proposed a number of Technical Specification changes related to assuring operability of the Dewpoint Monitoring System, correcting of inconsistencies between various specifications pertaining to moisture levels permitted under various operating conditions and providing equivalents in terms of dewpoint temperatures as actually measured by instruments in lieu of parts per million by volume (ppmv) as originally used in the specifications. From the standpoint of limiting coolant moisture levels during various operating conditions, these proposed changes are consistent with the previous Technical Specifications and are acceptable. These changes, as well as supplemental requirements for the Analytical Moisture Monitoring System, are discussed further in the following section.

We conclude that reactivity surveillance requirements with the Technical Specifications changed as proposed are adequate to assure that anomalous reactivity increases occasioned by loss of burnable poison from the core will be detected and corrective action taken before shutdown margin specifications are violated. PSCo takes daily readings of reactor coolant moisture levels during plant operation. This record will provide a data base for future evaluation should further operating experience indicate a need.

Moisture Monitors

Two moisture monitoring systems are provided for the Fort St. Vrain reactor. One of these is an analytical system which is capable of continuously drawing primary coolant via a helium sample pump. This system provides an alarm function only. The other system, a Dewpoint Monitoring System (DPMM), is part of the Plant Protection System and is designed to provide automatic corrective action in the event of water ingress during reactor operation - particularly in the event of a steam generator tube rupture.

The DPMM system consists of eight dewpoint moisture monitors which derive their sample flow from the pressure differential imposed across the helium circulators when they are in operation. Six of these are low level monitors, three of which sample each of the two reactor loops. Should a steam generator leak occur, the low level monitors are designed to trip (2 out of 3 logic) thus identifying the loop in which the leak is occurring and locking in protective circuitry which will cause steam generator dump and isolation in that loop following a high level DPMM signal.

The two high level monitors sample both loops simultaneously and operate on 1 out of 2 logic. If either of these monitors trip, the reactor will trip and dump and isolation of the steam generators in the loop identified by the low level monitors will automatically occur. Should the high level monitors fail to trip, they are backed up by a high coolant pressure trip which will initiate reactor trip protective action and dump and isolate a preselected steam generator. Functional performance of this DPMM system was reviewed in detail previously for a variety of conditions and found acceptable in terms of identifying water ingress and protecting the core from undergoing an excessive degree of damage owing to graphite oxidation. It was also concluded that the combined moisture and pressure detection systems, considering various failure modes, are capable of preventing pressure buildup to the point where the reactor vessel pressure relief valves would open, thus minimizing the potential for depressurization should the valves fail to reseat.

As indicated previously, when the reactor was taken critical in January 1975, neither the analytical system nor the DPMM system gave an indication of excessively high moisture levels, even though both systems were in operation at one time or another. This raised questions regarding the capability of both systems, and initiated corrective modifications on the part of PSCo. In the case of both systems it was found that sample line temperatures were so low that condensation prevented a truly representative sample from reaching the instruments. Sample line temperatures for the analytical instruments limited readings to about 2000 ppmv. Trip settings for the DPMMs were set at about 5000 ppmv, but condensation in the sample lines limited the sample to about 4500 ppmv. Our evaluation of the corrective measures is outlined below.

Dewpoint Monitoring System Modifications

In connection with determining what modifications might be necessary to correct deficiencies in the DPMM system, PSCo conducted a number of moisture injection tests, made temperature measurements to determine sample line temperature limitations and determined what Technical Specification changes were necessary to overcome the sample line condensation problem. The moisture injection tests disclosed an additional deficiency with respect to response times at low power levels which necessitated further modifications. These tests and the modifications are summarized in GA-A-13677[8] which was submitted on December 23, 1975 and in GA-A-13823[9] which was submitted in February 1976.

The DPMM sensing heads are located in penetrations in the walls of the prestressed concrete reactor vessel (PCRV). These penetrations are cooled by the liner cooling water system which runs at an average temperature of about 100°F. Since the sample lines to these heads will run at an ambient temperature near that of the cooling water system, an inherent limitation is imposed when penetration heat loads are small during low power operation. At higher power levels, the ambient temperatures will be higher owing to hotter gas recirculating in the sample supply lines. In addition, it was determined that sample lines, in some instances, were being chilled to even lower temperatures because of their close proximity to gaseous nitrogen lines (fed from liquid nitrogen Dewars) which cool the mirrors in the sensors. Corrective measures have included the installation of improved insulation and relocation of some of the sample lines. Contact pyrometer measurements of these sample lines have verified that sample line temperatures will be above 88°F for all low level instruments and above 96°F for both high level instruments. Technical Specifications which have been proposed[7][23] will require that the low level instruments be set at a dewpoint trip setting of $\leq$ 27°F whenever the DPMM system is required to be in operation. Thus, these instruments will have ample temperature margin below sample line temperatures to assure that a saturated gas sample will reach the instruments to cause mirror fogging at the trip setting. A temporary exception to the original Technical Specifications permitted the high level instruments to be set at a dewpoint corresponding to 5000 ppmv during low power operation (which was their setting during the moisture problem of January 1975). To assure an ample margin below sample line temperatures, this exception will be removed and adherence to the Technical Specification dewpoint setting of $\leq$ 67°F, will be required whenever the DPMM system must be operable. This gives sufficient temperature margin to assure that a saturated gas sample can reach the instruments to cause mirror fogging and trip.

As noted previously, testing of the DPMM system indicated that response times would not be adequate at low power levels. Modifications, as described in GA-A13677[8] have been made to correct this deficiency, and tests of the modified system, described in GA-A13823[9], have been conducted.

Functionally, each of the DPMM monitors is provided with a sample source via a recirculating supply line which derives its driving pressure from the helium circulators. The monitors themselves tap this supply line, with much smaller diameter sample tubing, upstream of a bypass valve in the supply line and return the sample, after traveling through the sensing head, downstream of this bypass valve. Thus the bypass valve is in parallel with the sensing head. Based on development testing by General Atomic and data from Hot Functional Testing at the Fort St. Vrain plant, it was believed that the bypass valve in the supply line could be put at a fixed setting such that an adequate sample rate through the sensing head would be provided at all power levels. Tests performed since the water ingress problem have demonstrated that these bypass valves will have to be adjusted to achieve adequate sensing head response times at intermediate to low reactor power levels when the circulators are operated at correspondingly lower speeds, thus reducing the pressure differential across the sampling system.

Based on tests of the DPMM system, representative of circulator pressure differentials over the full reactor power range, PSCo has determined that if the bypass valve were modulated to give a constant flow rate of about 60 SCC/sec through the instruments, at pressure differentials representative of reactor power levels between about 8% and 100% of full power, and a fixed bypass valve setting of 1/8 turn open below about 8%, response characteristics would be acceptable. Accordingly, the system has been modified by installation of an automatic control system.

This modification entails the installation of a reversible control motor, for each DPMM bypass valve, mounted on the outer surfaces of the penetration closures. Since the penetration closures are tertiary barriers against helium leakage, the shafts for these motors are connected through low leakage seal assemblies to linkages which reach the bypass valves inside. These motors transmit modulating control motion to the bypass valves using a control signal derived from existing flow meters in the sample lines to the DPMM instruments. The nominal control setting will be 62.5 SCC/sec, and high and low flow alarms will be set at 50 and 75 SCC/sec respectively. Low-low flow alarms are provided to alert the operators to an abnormally low sample flow rate, such as might result from clogging of filters provided in the sample lines to protect the DPMM mirrors from erosion. Below circulator speeds equivalent to about 8% of reactor power, it is anticipated that the bypass valve will remain at a fixed setting of 1/8 turn open (by limit switch) with sample flow rates decreasing from 62.5 SCC/sec to about 35 SCC/sec at ΔPs corresponding to 5% reactor power. Flow tests conducted on each of the eight DPMMs have verified that the newly installed control systems will maintain sample flows within the prescribed ranges under steady-state and load change conditions.

Response times for the DPMM system predicted in the FSAR and previously accepted as adequate by the staff, in terms of minimizing graphite damage owing to oxidation and assuring protective action before PCRV relief valve settings are reached, are 8.6 seconds at full power and 39.5 seconds at 25% power. Response times for lower power levels were not addressed at the time, but have been extended in the current review.

In order for the DPMM system to properly carry out its safety function it is necessary that both the low level and the high level moisture monitors reach their trip points before a High Reactor Pressure Trip occurs. This pressure trip is programmed to occur at a coolant pressure 7-1/2% above the operating pressure for any given reactor power level. Calculations based on the design basis water inleakage rate from a steam generator tube rupture indicate that a high pressure trip would be reached in about 113 seconds at power levels below 25%. This defines the time envelope within which the DPMM system must complete its action. A response time of 100 seconds at 5% power will give adequate margin to assure proper DPMM safety action before a high pressure trip is reached, and we have identified this as a requirement for the system.

Below 5% power, the response characteristics of the DPMM system may become uncertain because of reduced helium circulator ΔPs, and in this range the analytical moisture monitors will be utilized as a supplementary means of determining coolant moisture levels. In this low power range, graphite temperatures and coolant temperatures will be sufficiently low that graphite oxidation rates would be insignificant and pressure increases would not reach relief valve settings in the event of water ingress. We will require that the DPMM system be kept in operation below 5% power, but without a specific response time requirement. In the event of a design basis steam generator leak the DPMM system will function but with a relatively long response time. In addition, a high pressure trip could occur at power levels below 5%. This would cause reactor scram and dump and isolation of steam generators in a preselected loop. Test data, reported in GA-A13677[8], indicate that the DPMM system should respond within 113 seconds at power levels above 2% so long as filters in the sample lines remain clean. Accordingly, during initial plant operations there is reasonable expectation that the system will identify a leaking loop and provide automatic corrective action, thus avoiding the complications that might arise from the possibility of "wrong-loop-dump" if sole reliance were placed on the high pressure trip.

With clean filters the predicted response times of the modified system, based on data reduced from moisture injection tests, are summarized in Table 4 and Figure 3 of GA-A13823[9] for both high level and low level DPMMs. These data indicate that for ΔPs corresponding to equilibrium core conditions the overall system will be capable of correctly identifying the loop in which a steam generator tube rupture may have occurred and initiating a reactor trip and dump and isolation of the steam generator in about eight seconds at full power, about 17 seconds at 25% power, and about 33 seconds at 5% power. For beginning of life core conditions, the ΔPs will be smaller for a given power level and the response times slightly longer.

The modified DPMM system incorporates alarms which will assure that deviations from acceptable sample flow rates, in terms of required response times, will be automatically brought to the attention of operators in the plant control room. For initial plant operation, these alarms will be set in accord with sample flow rates representative of those used in testing the DPMM system and thus will be effective over the required range of reactor power levels.

PSCo has proposed changes(7)(23) to several of the Technical Specifications pertaining to the DPMM system. The proposed change to SR 5.4.1 adds periodic surveillance requirements for the modified DPMM system. These are adequate. The proposed change to LCO 4.2.11 substitutes allowable moisture levels as a function of coolant temperature, expressed in dewpoint temperature rather than ppmv as done previously. This change provides consistency within the Technical Specifications. Proposed changes to LCO 4.4.1 substitute equivalent dewpoint settings for the high and low DPMM instruments in place of settings expressed in ppmv. In addition sample flow rate requirements for the DPMM system to be operable have been proposed as follows: Below 2% power a minimum sample flow rate of 1 SCC/sec; between 2% and 5% power a minimum sample flow rate of 5 SCC/sec; at power levels between 5% and 20% a minimum sample flow rate of 15 SCC/sec; between 20% and 35% power a minimum sample flow rate of 30 SCC/sec and between 35% and 50% power a minimum sample flow rate of 50 SCC/sec. The sample flow rates are consistent with assuring that the required DPMM response times are met. We conclude that the proposed Technical Specification changes are acceptable.

Alarms must be set at the minimum sample flow rate identified above for each of these power level steps to assure that operators will be alerted to sample flow rates that go below these minima. Sample flow rates and any associated requirements have not yet been established for power levels above 50%. However, initial operations will be limited to 40% power. Before authorization is issued for plant operations above 40%, the foregoing sample flow rate requirements will be reexamined in the light of moisture injection tests results obtained during power ascension and the Technical Specifications will be modified if necessary and extended to cover operations up to full power.

From our review of the details of the DPMM system, including the control system, the physical installation, the locations of components, provisions for flow alarms and circuit arrangements, together with the proposed Technical Specification changes, we conclude that the modifications made to the DPMM system are acceptable.

The existing flow sensing devices, to be used to derive control signals for bypass valve modulation, are located in the closed instrument penetrations and thus will be exposed to ambient temperatures which will vary considerably with reactor power. Temperatures are not expected to exceed 185°F. PSCo has taken into account the possibility of a 20% flow sensor decalibration at somewhat higher temperatures in assessing response times. We concur that this degree of decalibration should still leave acceptable response times. However, since there is some uncertainty in the penetration ambient temperature calculations, PSCo has agreed, by letter dated January 28, 1976[10] to install a thermocouple in one of the penetrations containing a single DPMM instrument and another in a penetration containing two instruments, to verify the calculations. The thermocouples must be mounted on the flow sensors. Information from these thermocouples will provide data for further evaluation and any needed corrective action should temperatures be projected to exceed the 185°F estimate as the reactor approaches full power. At the request of

NRC, PSCo has initiated a Technical Specification change[23] to limit flow sensor temperatures to a maximum of 185°F unless Bases satisfactory to NRC are provided to justify operation at a higher temperature.

By letter dated December 9, 1975[11], as amended by letter dated March 3, 1976[12], PSCo requested permission to temporarily bypass inputs to the plant protection system from the DPMM system in order to conduct moisture injection tests at 5%, 25%, and 100% of reactor power to measure response times. The existing test system will be utilized to inject small amounts of moisture-laden helium for these tests. During the periods while the system is bypassed, two observers in direct communication with the reactor operator will continuously observe the moisture monitors to assure that if their response should indicate moisture ingress via a steam generator leak or some other cause, the operator can take immediate corrective action manually. In addition, the Analytical System Moisture Monitors will be used to monitor coolant moisture levels during the tests. Based on review of the test procedure to be used, the conditions under which testing will be done and Technical Specification changes[23], we concur in PSCo's conclusion that these tests can be carried out safely.

It is anticipated that data reduced from these tests will confirm the general response time characteristics of the DPMM system as predicted from tests upon which the modifications are based, and that response times will fall well within the limits discussed above, particularly at intermediate to low power levels. Precise correspondence with predicted performance is not necessary. However, if the data indicate that the shape of the response time curve as a function of circulator ΔP

(corresponding to power levels between 5 and 100% of reactor power) deviates substantially from the prediction, further evaluation should be done to assure that response times remain acceptable at all power levels. If necessary, because extrapolation is uncertain, additional moisture injection test data should be acquired. We will review the results of testing at 5% and 25% power in connection with verifying and extending the Technical Specification limits discussed above.

Analytical Moisture Monitoring System

The Analytical Moisture Monitoring System is not part of the plant protection system but may be used under some plant conditions as a means of continuously determining reactor coolant moisture levels when the DPMM system is in its trip mode. Periodic sampling of primary coolant for laboratory analysis may also be employed for this purpose.

The Analytical System contains two dewpoint analytical instruments. The sample line for this system comes through the PCRV wall and travels several hundred feet through areas of the plant where ambient temperatures are less than those encountered in the PCRV wall. Since the moisture ingress event of January 1975, the sample line has been trace heated to assure that the limiting temperature for the sample line will occur in the PCRV wall, thus assuring that a coolant sample can be delivered to the instruments under essentially the same conditions as for the DPMM system. With this change, assurance is provided that the analytical system can cover the same moisture range as that covered by the DPMM system at the new proposed dewpoint trip settings. Previously, this range would have been limited, owing to condensation in the sample line at whatever ambient temperature the line may be exposed to.

By letter received on December 5, 1975[13], PSCo proposed two new Technical Specifications to cover conditions under which the Analytical System Moisture Instrumentation would be required to be in operation (LCO 4.4.5) and Surveillance Requirements for this system (SR 5.4.12) Under this change, the Analytical System will be required to be in operation at power levels of 5% or less. This system performs an alarm function only and operator corrective action would be required if moisture levels became excessive. In essence, the Analytical System provides a backup to the automatic action of the DPMM system over the power range where its response characteristics may be uncertain. We conclude that the proposed changes to the Technical Specifications are acceptable.

CONCLUSION

Based on our review and the considerations discussed above we have concluded that: (1) damage to materials in the Fort St. Vrain reactor has not occurred owing to moisture ingress; (2) adequate measures have been taken to minimize the potential for recurrence of moisture ingress; (3) operability of the control drive assemblies and the reserve shutdown system has been verified; (4) modifications to the moisture monitoring systems are acceptable; (5) suitable tests will be performed during rise-to-power testing to verify response characteristics of the DPMM system; and (6) operation of the reactor under the conditions of the proposed revisions to the Technical Specifications is acceptable.

Based on our review we have also concluded that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

References

1. Public Service Company of Colorado Quarterly Progress Report for the Period Ending March 31, 1971, GA-10560, April 30, 1971.

2. The Status of the Fort St. Vrain Reactor Core Prior to Power Operation, by the General Atomic Fuel Group Staff, March 11, 1975.

3. Letter, R. A. Clark of NRC to R. F. Walker of PSCo, May 9, 1975.

4. Letter, R. F. Walker of PSCo to R. A. Clark of NRC, June 23, 1975.

5. General Atomic Standard Safety Analysis Report, p. 4.2-47.

6. Letter, C. K. Millen of PSCo to R. S. Boyd of NRC, September 11, 1975.

7. Letter, C. K. Millen of PSCo to R. S. Boyd of NRC, March 23, 1976.

8. Test and Evaluation of the Fort St. Vrain Dew Point Moisture Monitor System, GA-A13677, October 10, 1075.

9. Fort St. Vrain Dew Point Moisture Monitor System Post Modification Test Results (RT-355C), GA-13823, February 17, 1976.

10. Letter, R. F. Walker of PSCo to R. P. Denise of NRC, January 28, 1976.

11. Letter, R. F. Walker of PSCo to Robert Clark of NRC, December 9, 1975.

12. Letter, R. F. Walker of PSCo to Robert Clark of NRC, March 3, 1976.

13. Letter, C. K. Millen of PSCo to R. S. Boyd of NRC, undated letter received December 5, 1975.

14. Fort St. Vrain Abnormal Occurrence Report No. 50-267/75/3, January 31, 1975.

15. Fort St. Vrain Abnormal Occurrence Report No. 50-267/75/4, March 12, 1975.

16. Fort St. Vrain Abnormal Occurrence Report No.50-267/75/7A, April 4, 1975.

17. Fort St. Vrain Abnormal Occurrence Report No. 50-267/75/3A, April 7, 1975.

18. Fort St. Vrain Unusual Event Report No. 50-267/75/4A, August 6, 1975.

19. Fort St. Vrain Unusual Event Report No. 50-267/75/18A, December 15, 1975.

20. Fort St. Vrain Unusual Event Report No. 50-267/75/93, March 25, 1976.

21. Status Report on Reserve Shutdown System, GA-A13742, November 25, 1975.

22. Letter, F. E. Swart of PSCo to W. Gilbert of NRC, May 13, 1976.

23. Letter, C. K. Millen of PSCo to R. S. Boyd of NRC, June 14, 1976.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-267

PUBLIC SERVICE COMPANY OF COLORADO

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. DPR-34 issued to Public Service Company of Colorado which revised Technical Specifications for operation of the Fort St. Vrain Nuclear Generating Station, located in Weld County, Colorado. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to (1) add requirements for operation of analytical system moisture monitors between reactor shutdown and 5 percent power; also calibration frequency for these monitors is stated; (2) revise allowable primary system impurity levels and method of specifying moisture impurity from parts per million to dew point temperature; (3) add a definition of operable dew point moisture monitor; (4) add functional checks and tests for dew point moisture monitors; (5) revise the core reactivity status surveillance and limiting conditions for operation; (6) isolate the helium storage system from the helium circulator buffer helium system when the reactor is in operation; (7) allow bypass of plant protective system moisture monitors for testing during the startup testing program; and (8) add reporting requirements.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated September 11, 1975; December 1, 1975; March 23, 1976; and June 14, 1976; (2) Amendment No. 13 to License No. DPR-34, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Greeley Public Library, City Complex Building, Greeley, Colorado 80631.

A copy of items (2) and (3) may be obtained upon request addressed to the United States Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland, this 18th day of June 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Robert A. Clark

Robert A. Clark, Chief
Special Reactors Branch
Division of Project Management

Public Service Company of Colorado

16805 WCR 19 1/2, Platteville, Colorado 80651

December 4, 1984
Fort St. Vrain
Unit #1
P-84515

Mr. Robert Martin, Regional Administrator
Reactor Project Branch 1
Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

ATTN: Mr. E. H. Johnson

REFERENCE: Facility Operating License No. DPR-34

Docket No. 50-267

Dear Mr. Collins:

Enclosed please find a copy of Licensee Event Report No. 50-267/84-012, Preliminary, submitted per the requirements of 10 CFR 50.73(a)(2)(v).

Sincerely,

J. W. Gahm

J. W. Gahm
Manager, Nuclear Production

Enclosure

cc: Director, MIPC

JWG/djm

~~8412260031~~

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort St. Vrain, Unit No. 1	0 5 0 0 0 2 6 7	1 OF 0 7

TITLE (4)
During SR 5.1.2c-X, Only ½ Of Total RSD Material Was Discharged From CRD#21

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 1	0 5	8 4	8 4	— 0 1 2 —	0 0	1 2	0 5	8 4	N/A	0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): N
POWER LEVEL (10): 0 0 0

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	XX 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Jim Eggebroten, Technical Services Engineering Supervisor	AREA CODE 3 0 3 — 7 8 5 - 2 2 2 4

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
XX YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 3	0 5	8 5

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 0830 hours on November 5, 1984, with the reactor shutdown for control rod drive (CRD) inspection and maintenance, the reserve shutdown hopper of control rod drive and orifice assembly (CRDOA) #21 was functionally tested in the hot service facility per SR 5.1.2c-X, "Reserve Shutdown Assembly Functional Test". During performance of the test, it was discovered that about 40 pounds of reserve shutdown material (40 weight percent boron) had been discharged from the hopper assembly. The reserve shutdown hopper is designed to release approximately 80 pounds of material containing neutron absorbing boron carbide into the core upon rupture of the hopper rupture disc.

The event was reported to the Nuclear Regulatory Commission at 1225 hours on November 5, 1984, per the requirements of 10 CFR 50.72(b)(2) "four hour report".

The failure of the CRDOA #21 hopper assembly to discharge an acceptable amount of reserve shutdown material during performance of SR 5.1.2c-X is being reported pursuant to the requirements of 10 CFR 50.73(a)(2)(v).

The reactor remained in a cold shutdown condition throughout this event.

An investigation is presently underway to determine why some of the reserve shutdown material was retained inside the CRDOA #21 hopper assembly.

NRC Form 366
(9-83)

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort St. Vrain, Unit No. 1	0 5 0 0 0 2 6 7	8 4	0 1 2	0 0	0 2 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

EVENT DESCRIPTION:

The purpose of the reserve shutdown system is to provide a means of admitting sufficient negative reactivity into the core to ensure an adequate core shutdown margin from any reactor operating condition completely independent of the control rod system.

The reserve shutdown system is composed of a storage hopper located between the control rod drive mechanism and the thermal shield at the lower end of each refueling penetration. Each hopper contains nominally spherical neutron absorber material composed of boron and graphite. This absorber material is held inside the hopper by a rupture disc.

A steel guide tube extends from the underside of the hopper to the top control reflector block of the associated core region. The guide tube engages the top reflector block, forming a clear passageway for the reserve shutdown material to fall from the hopper, through the guide tube, and into the core (see Figures 1 and 2).

Rupture of the hopper rupture disc and subsequent release of the absorber material into the core is initiated by pressurizing the hopper with helium. Each hopper is connected to a separate high pressure helium bottle (2200 psi nominal) by a pressurizing line that allows helium flow from the bottle into the hopper immediately above the rupture disc (Figure 3). These bottles have an alarm system associated with them that will actuate when the bottle pressure drops below approximately 1640 psig, at which time the bottles are replaced. Section 3.8.3.2 of the FSAR analyzes reserve shutdown system performance with a minimum helium bottle pressure of 1500 psig. In this case, if the rupture discs fail to burst at the design differential pressure of 165 ± 50 psi, the hopper pressure could build to a maximum of 1015 psia. Since the reactor pressure is 700 psia, a minimum differential pressure of 315 psi can be imposed across the disc, assuring its rupture.

SR 5.1.2c is performed to determine the reliability of the differential burst pressure of the disc, and detect any tendancy of the poison material to bridge or deteriorate in the hoppers over extended periods of time. The surveillance consists of placing the CRDOA inside the hot service facility over a pre-weighed container, so that the reserve shutdown material will fall into the container when the rupture disc bursts. A helium line and pressure guage are connected to the CRDOA hopper assembly, and the hopper is pressurized until the rupture disc bursts. The container is then weighed to determine the amount of reserve shutdown material released during the test. Eighty ±eight pounds of reserve shutdown material must be released in order to satisfy SR 5.1.2c-X acceptance criteria.

Upon discovering that only forty pounds of reserve shutdown material had been released during the test, maintenance personnel performed a visual inspection of the hopper internals using a borescope. The material that failed to discharge from the hopper was removed, and samples were collected for internal analysis. Samples were also sent to Los Alamos National Laboratories for Nuclear Regulatory Commission independent analysis.

NRC FORM 366A (9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort St. Vrain, Unit No. 1	0 5 0 0 0 2 6 7	8 4	— 0 1 2	— 0 0	0 3 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

ANALYSIS:

The reserve shutdown system is designed to provide sufficient negative reactivity control to achieve hot shutdown conditions from any operating condition without movement of the control rods. This condition can be met with two of the thirty-seven reserve shutdown hoppers inoperable per LCO 4.1.6, providing for a total negative reactivity insertion of at least .088ΔK in the equilibrium core.

The capability of pressurizing the reserve shutdown hoppers is demonstrated once each quarter, during normal plant operation. The "low bottle pressure" alarm circuitry is functionally tested once per quarter, and calibrated annually to insure that any loss of the minimum required rupture gas pressure is readily detected (see SR 5.1.2).

An off-line functional test of a reserve shutdown assembly has been performed following each of the three refueling cycles to date, as required per the Fort St. Vrain Technical Specifications. During each of these tests, the rupture disc burst pressure was below 300 psid as required per Section 3.8.3.5 of the FSAR, and acceptable amounts of absorber material were released from the hoppers.

FSAR Section 3.8.3.4 analyzes the reserve shutdown neutron absorber material and concludes that bridging and deterioration are not anticipated under the temperature, radiation, and helium environment in which the material is stored inside the hoppers during operation.

Two reserve shutdown hoppers have been functionally tested as a result of control rod drive problems recently encountered (see LER #84-008). The two reserve shutdown hoppers tested were on CRDOA #26 and CRDOA #21. During testing of CRDOA #26, all of the reserve shutdown material (20 weight percent boron) was released from the hopper as designed, however, the hopper assembly of CRDOA #21 (40 weight percent boron material) did not function properly as outlined in this report.

The potential safety consequences of this event are currently being investigated and will be analyzed further once the cause and extent of the problem are known.

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

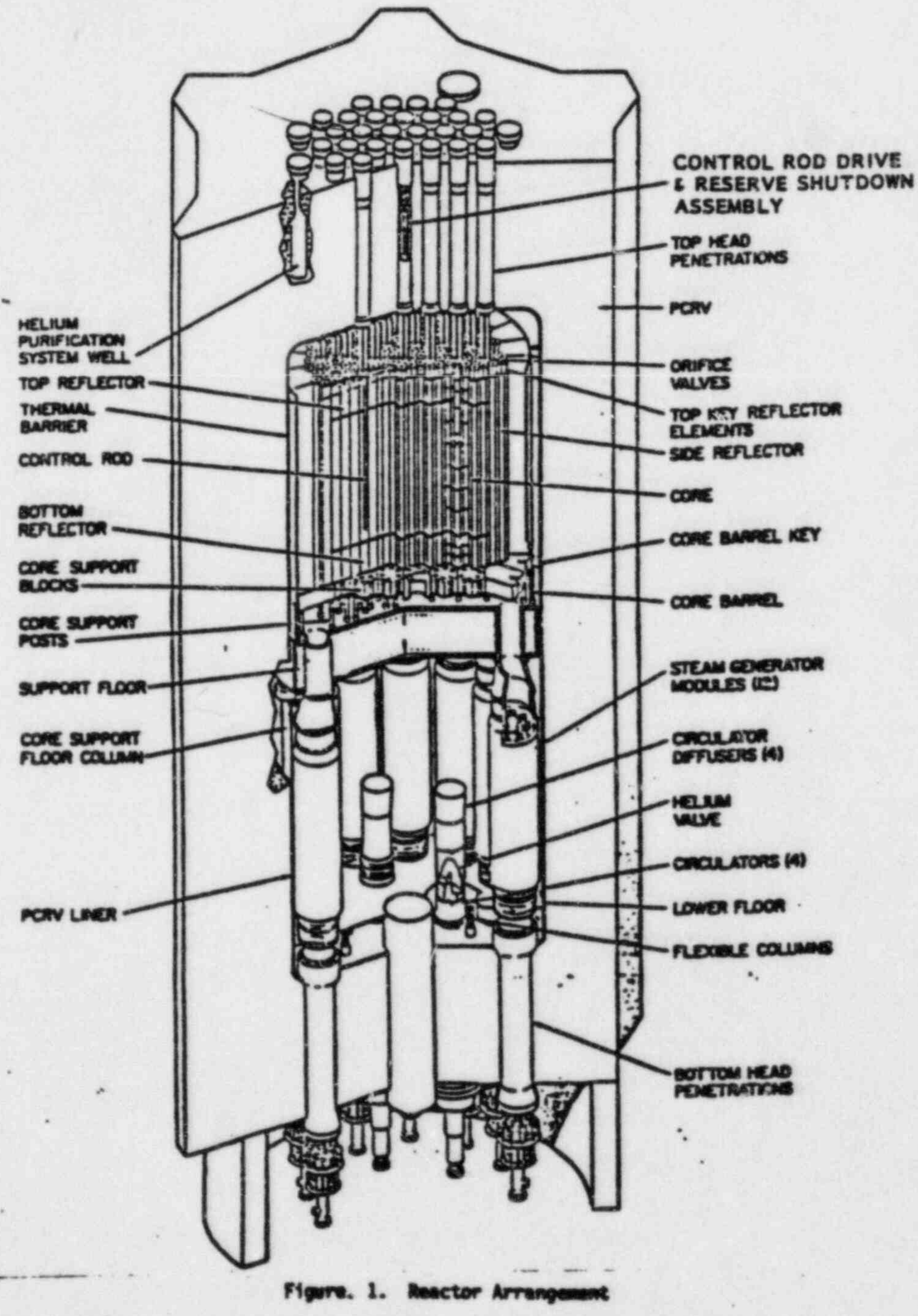

Figure. 1. Reactor Arrangement

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
	0 5 0 0 0 2 6 7	8 4	— 0 1 2 —	0 0	0 5 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

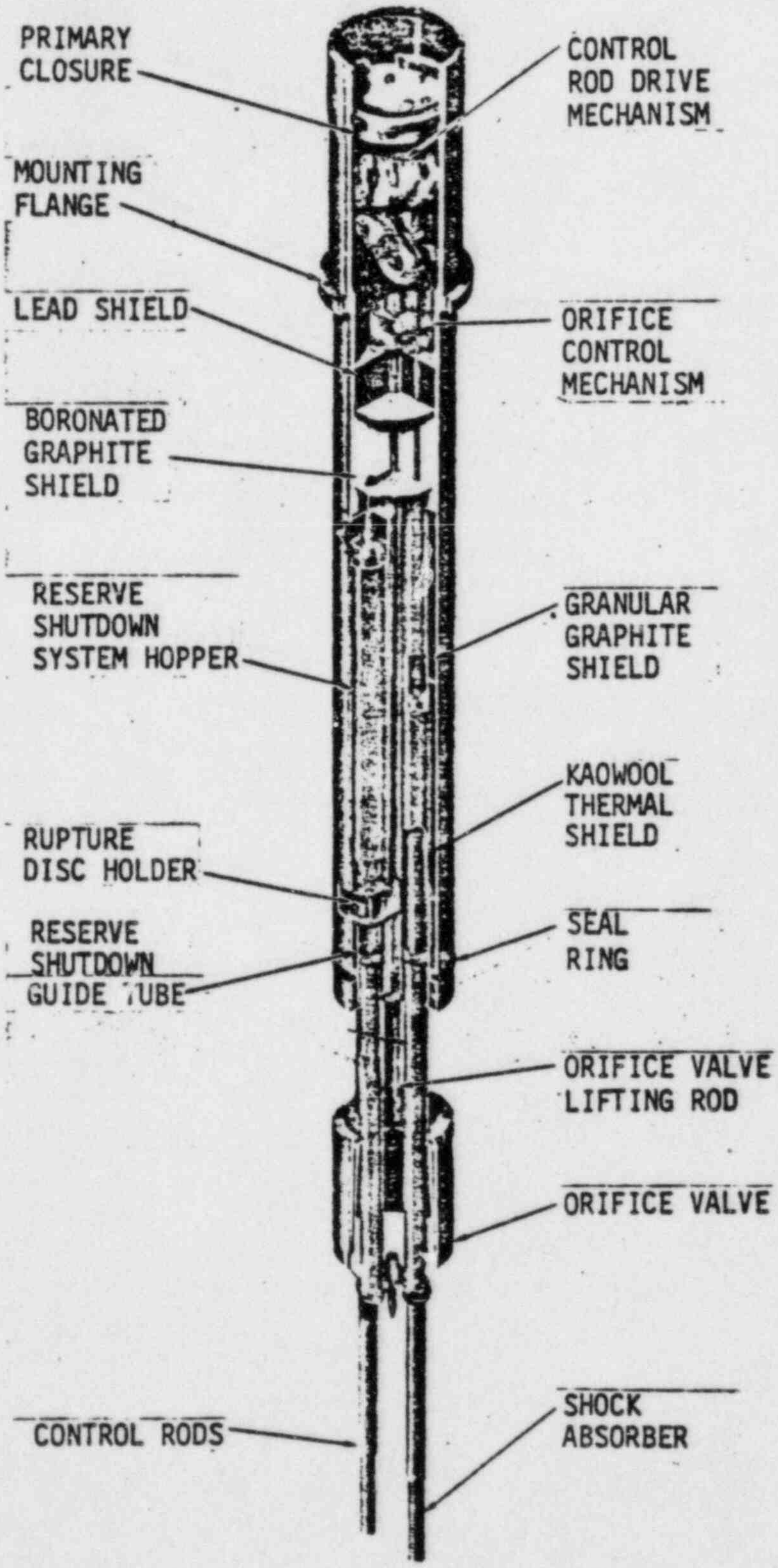

FIG. 2 CONTROL AND ORIFICING ASSEMBLY

NRC FORM 366A
(9-83)

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
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EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
	0 5 0 0 0 2 6 7	8 4	— 0 1 2	— 0 0	0 6 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

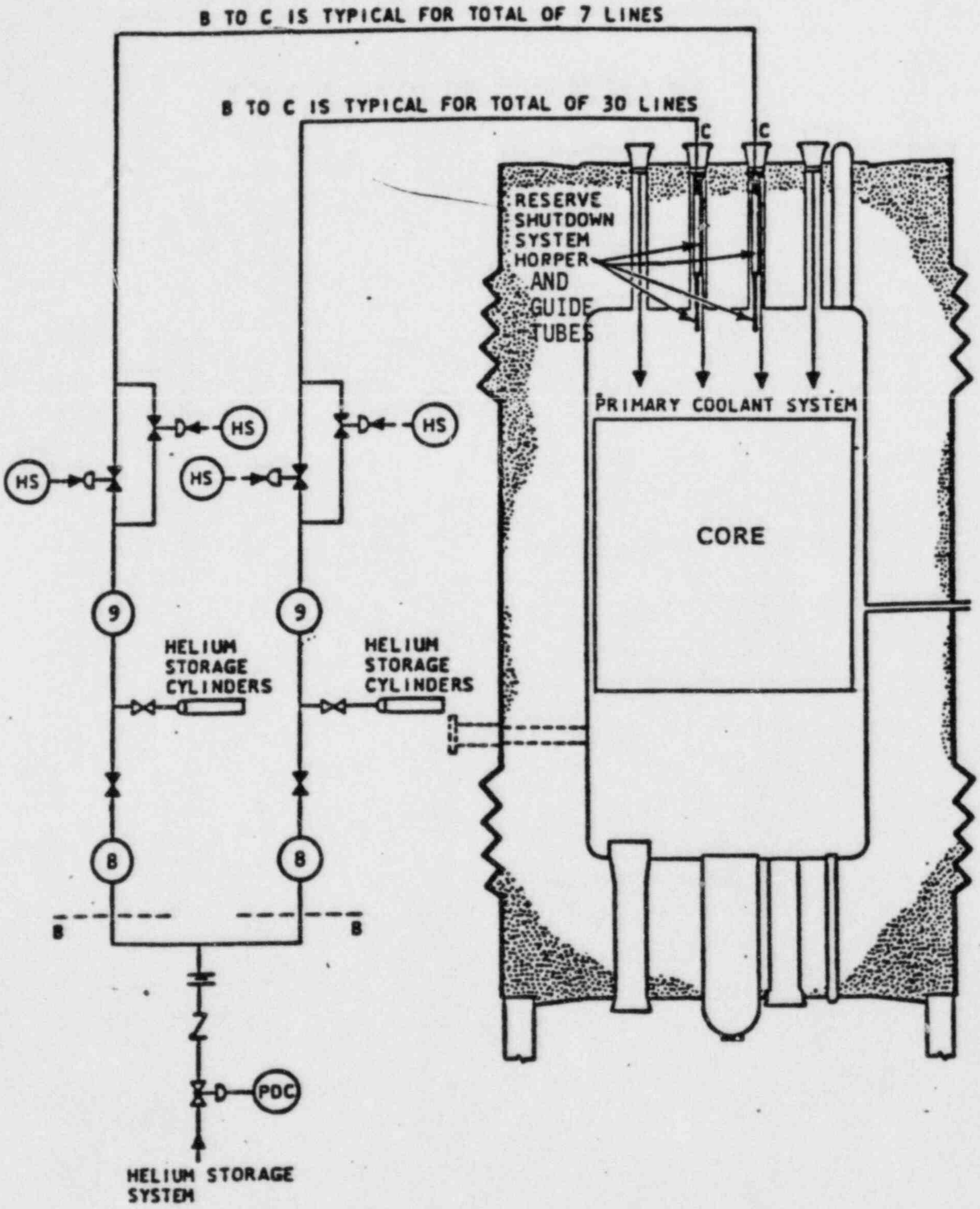

Fig. 3. Reserve shutdown system flow diagram.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort St. Vrain, Unit No. 1	0 5 0 0 0 2 6 7	8 4	0 1 2	0 0	0 7 OF 0 7

TEXT (If more space is required, use additional NRC Form 366A's) (17)

CORRECTIVE ACTION:

As mentioned previously, the cause and extent of this anomaly are presently under investigation, along with the development of an appropriate plan for corrective action prior to returning the plant to operation.

A supplemental report will be submitted March 5, 1985.

Jim Hill
Jim Hill
Techical Services Technician

Jim Eggebroten
Jim Eggebroten
Technical Services Engineering Supervisor

Licensing Review By:

Jim Gramling
Jim Gramling
Nuclear Licensing-Operations Supervisor

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AEC RESEARCH AND
DEVELOPMENT REPORT

GA-9875
UC-2 General,
Miscellaneous, and
Progress Reports

PUBLIC SERVICE COMPANY OF COLORADO
330-MW(E) HIGH-TEMPERATURE GAS-COOLED REACTOR
RESEARCH AND DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
DECEMBER 31, 1969

Prepared under
Contract AT(04-3)-633
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

January 30, 1970

QUARTERLY REPORT SERIES

GA-6830-July, 1965, through September, 1965
GA-6950-October, 1965, through December, 1965
GA-7086-January, 1966, through March, 1966
GA-7314-April, 1966, through June, 1966
GA-7453-July, 1966, through September, 1966
GA-7634-October, 1966, through December, 1966
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GA-8725-April, 1968, through June, 1968
GA-8879-July, 1968, through September, 1968
GA-9130-October, 1968, through December, 1968
GA-9261-January, 1969, through March, 1969
GA-9440-April, 1969, through June, 1969
GA-9720-July, 1969, through September, 1969

INTRODUCTION

The objective of the research and development program work reported here is to develop and verify the information required to design, construct, operate, and maintain the Public Service Company of Colorado power plant as provided in U.S. Atomic Energy Commission Contract AT(04-3)-633.

Part I of this report includes the work described in Appendix B of the contract; this work consists largely of component development and testing, nuclear analysis, and fuel development and testing. Part II covers the work described in Appendix K of the contract on the fuel transfer machine, the series-steam-turbine-driven circulator, the control rod drives, the steam generator, and coated particles.

CONTENT

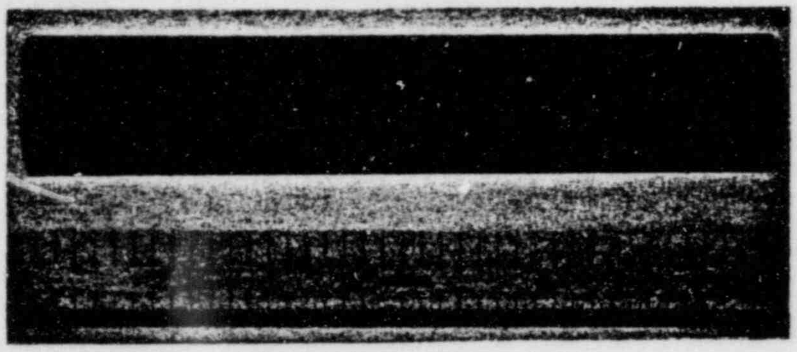


Fig. 3.38--Fracture mode due to hoop stress only in 0.75-in. OD, 0.5-in. ID graphite tube (H-327) (axis parallel to log extrusion). Estimated failure stress in excess of 1200 psi (M-31717-3)

In conclusion, the fuel hole arrangement in a fuel block is a thick shell configuration; thus, when subjected to internal loading, defects on the face of the bore would be the most significant. Defects that breach the web between any two holes, but that are equal to or below the acceptable maximum diameter, do not precipitate failure under static loads. The fracture within a fuel hole remains local to the area of applied load when excessive hoop stresses are created at the inner fibers (i.e., at the ultimate stress, fracture of a coolant hole web occurs without total breakup of the material).

Control Materials

Experiments on the behavior of boronated graphite materials at conditions calculated to occur during the hypothetical loss-of-forced-circulation (LOFC) accident have continued (see earlier quarterly reports GA-9130 and GA-9440 for results of earlier work). The tests are being performed to study the transport of boron from control rod materials and to measure the degree of compaction or slumping of boronated compacts subjected to a compressive load.

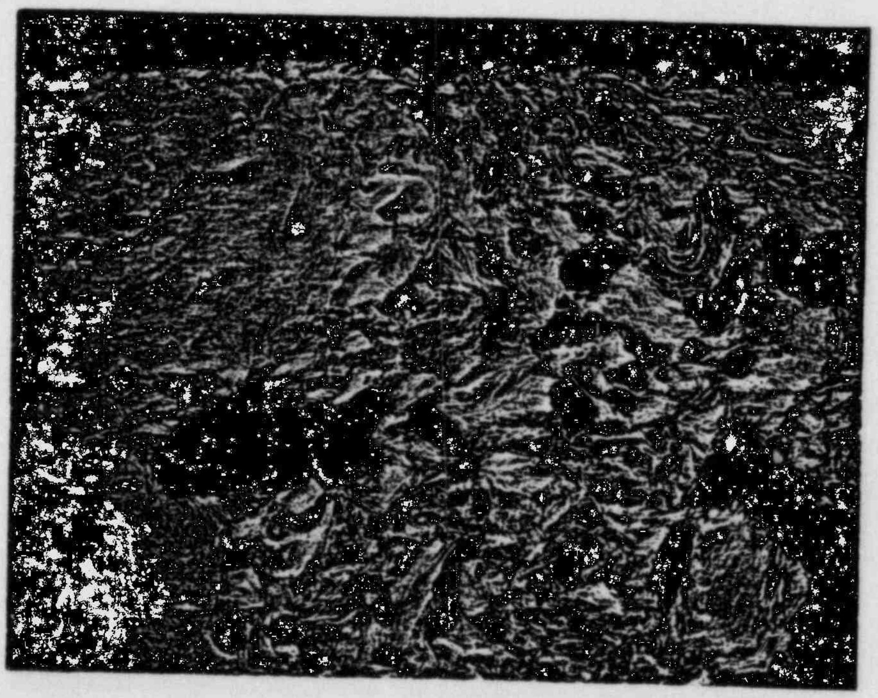
The conditions chosen for these experiments are those calculated to occur in a small region at the center of the core during the LOFC accident. Temperatures up to 2980°C are predicted; and, due to melting of metal components and subsequent slumping of control rods, the central boronated compacts could be subjected to a compressive load of 16 psi.

Test samples were prepared from production boronated graphite compacts and from compacts prepared at Gulf General Atomic. In addition, tests were performed on boronated graphite spheres prepared at Gulf General Atomic (i.e., reserve shutdown material). Measurements of weight loss, boron loss, and structural integrity (i.e., slumping) were performed at 2950°C, at compressive loads of 16 psi, and for exposure times up to 120 hr.

The compact samples were right circular cylinders (1.6 cm diameter by 3 cm long) with the exception of sample 4705-14, which was 5 cm long. Each

M-31717-2 (50X)

Fig. 3.39--Structure through outer fiber section of tube (Fig. 3.39) showing relatively strong matrix devoid of fracture. Wall thickness reduced at defect position by 25%



M-31717-1 (75X)

Fig. 3.40--Section through wall of tube (Fig. 3.39) showing mode of fracture by linkup of discontinuities normal to principal stress direction

sample rested on the bottom of a surrounding graphite crucible and was held at the top by a weighted graphite ram. There was a radial clearance of about 0.5 cm between each sample and the inside wall of the graphite crucible.

Two types of spheres were tested in one experiment. Four spheres, two of each type were stacked vertically, the two types being separated by a graphite wafer. The sphere column was subjected to a 16-psi compressive load (based on the area of the equatorial plane of the spheres).

The experiments were conducted in an induction furnace, the temperatures being measured with an optical pyrometer that has been calibrated to 2300°C with a standard lamp. Temperature calibration runs utilizing the thermal arrest of melting ZrC were performed. Definite indications of melting occurred at an apparent temperature of 2845°C, and resolidification was observed at 2870°C. The published melting points for ZrC range from 2800° to 2911°C, with several investigators agreeing on 2850°C. It is concluded that temperatures measured during these tests are accurate to within ±50°C.

The experimental results of all of the tests are shown in Table 3.34, including data on slumping behavior, weight loss, and boron loss. In addition, estimates of the diffusion coefficient for boron loss from the compact samples are included as well as the apparent boron/carbon ratio in the effusing species.

All of the test samples slumped relatively little (see Col. 8, Table 3.34) as a result of the combined conditions of compressive load and high temperature (with the exception of sample 4705-48 which compressed about 30%). It is of interest that in all of the tests, sample compression was essentially complete after the first 5 or 6 hr of the run. This was true even for the 120-hr test in which the sample (No. 4705-14) slumped 8% during the first 5 hr and an additional 4% during the next 45 hr. The sample then showed no further sample compression throughout the remainder of the test period.

Boron losses are shown in Col. 7 of Table 3.34. These results are also shown in Fig. 3.41, where log boron loss is plotted versus log time. The data roughly fit a square-root-of-time relationship, which indicates a diffusion-controlled mechanism.

At the conclusion of the 120-hr test, all parts of the furnace internal components were assayed to obtain a boron material balance and to ascertain where the major portion of the effusing boron was collected. The data given in Table 3.35 indicate that significant boron redistribution can occur in a graphite system where an efficient sink is provided (i.e., graphite at lower temperatures).

Boron transport in laboratory-scale tests such as these is expected to be much higher than in the reactor core. This is because the increase in the boron content in the core graphite surrounding the control material is not duplicated in the laboratory.* Moreover, the diffusion path through the

*Boron in the core graphite would cause a back-pressure effect. This effect would not occur in the laboratory because cold regions in the furnace would act as sinks for the boron and would tend to limit the increase in boron content of the graphite surrounding the sample region.

Table 3.34

BEHAVIOR OF BORONATED GRAPHITE AT 2950°C AT A COMPRESSIVE LOAD OF 15 PSI

Sample No.[a]	Density (g/cm^3)	Initial Boron Content (%)	Anneal Time (hr)	Total Weight Loss (%)	Final Boron Content (%)	Total Boron Loss (%)	Sample Compression (%)	B/C[b]	D_{eff}[c] (10^{-7} cm^2/sec)
Production Compacts:									
4705-46	1.7	28.6	1.5	7.1	27.0	12.1	6.7	0.9	3.3
4705-38	1.7	28.6	4	8.6	23.1	26.7	3.9	8.0	6.0
4705-40	1.7	28.6	10	34.4	13.8	68.8	5.4	1.6	16.7
4705-29	1.7	28.6	12.5	29.6	17.3	57.6	9.4	1.5	10.8
4705-34	1.7	28.6	27	25.5	21.0	45.6	4.0	1.2	2.6
4705-42	1.7	28.6	38	35.6	10.2	80.2	6.8	1.2	5.9
4705-14	1.7	28.6	120	38.2	0.16	99.7	12	3.5	---
GGA Compacts:									
4705-48	1.76	30.7	10	34.1	20.0	57.5	29.5	1.0	11
4705-58	1.59	20.9	10	9.0	17.[illegible]	25.1	2.5	1.3	2.2
4705-62	1.68	20.3	10	8.8	16.9	23.8	1.3	1.5	2.0
4705-64	1.78	21.5	10	9.5	15.0	36.9	1.9	6.0	4.8
GGA 1/2-in.-diam. spheres:[d]									
4705-56L	1.49	20.7	10	--	12.9	44.5	3.7	---	3.7
4705-56H	2.14	18.5	10	--	12.2	33.6	2.1	---	2.1

[a]All compact samples were right circular cylinders (1.6 cm diam. by 3 cm long) with the exception of 4705-14, which was 5 cm long.

[b]Atom ratio (boron/carbon) in transported species.

[c]D_{eff} calculated from $F = A/L \sqrt{D_{eff} t}$ (see text).

[d]Two of each type of sphere were used in the same experiment (see text).

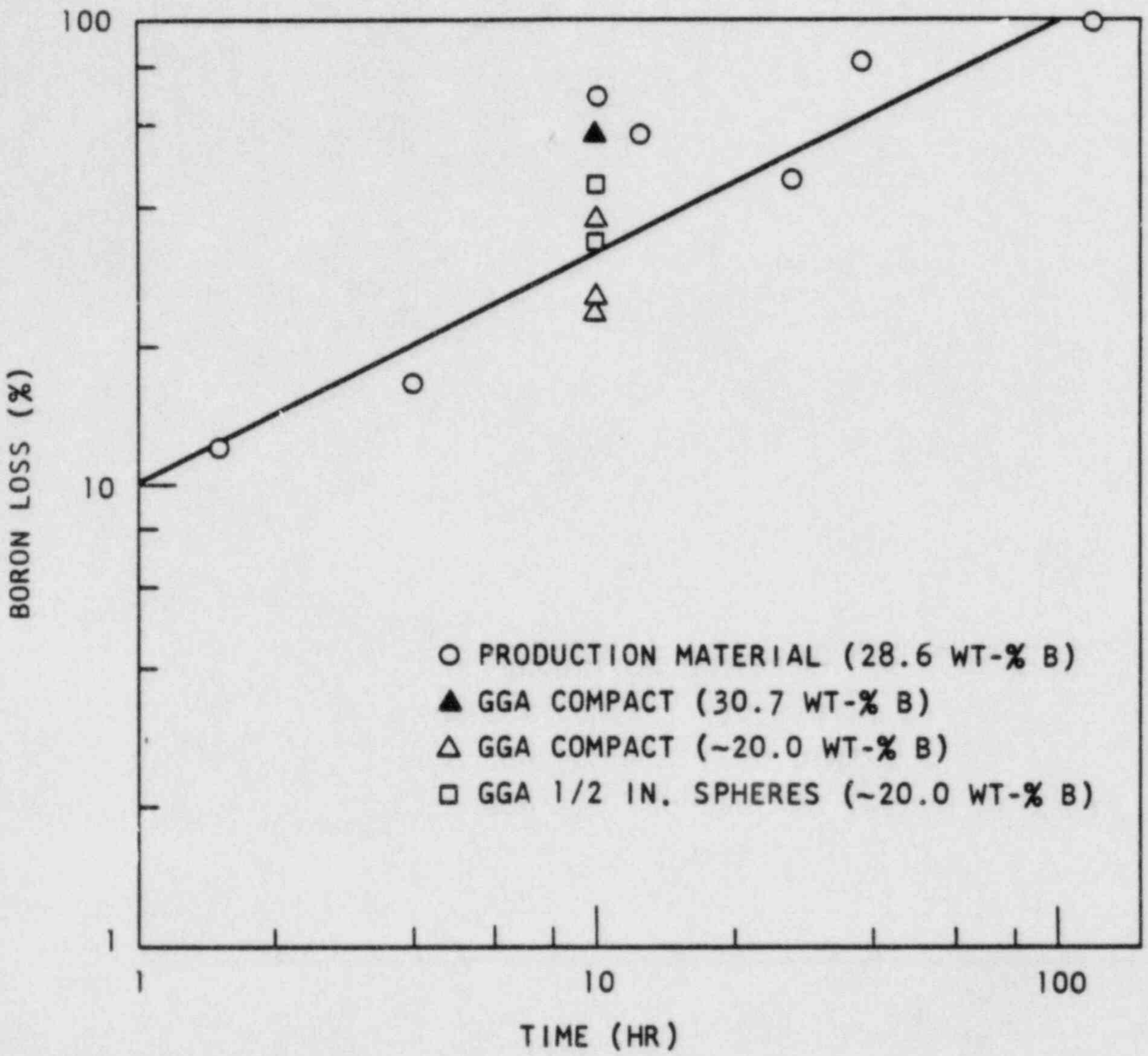


Fig. 3.41--Boron loss from boronated graphite at 2950°C

graphite is much greater in the core than in the experiment. Accordingly, an important objective of these experiments is to use the boron loss data to determine diffusion coefficients from which boron distribution in the core can be calculated.

Effective diffusion coefficients for boron losses from the test specimens were calculated using the relationships:

$$F = \sqrt{Dt}\ 2.256/L \text{ for cylindrical compacts, and}$$

$$F = \sqrt{Dt}\ 3.385/L \text{ for spherical specimens,}$$

where F = fraction boron lost,

D = effective diffusion coefficient, cm^2/sec,

t = time, sec, and

L = radius of sphere or cylinder, cm.

These relationships, which are not exact because geometry factors are not precisely taken into account, yield diffusion coefficients that should be of sufficient accuracy for reactor calculations.

The diffusion coefficients thus calculated are listed in Col. 10 of Table 3.34. They range from 2×10^{-7} to 1.7×10^{-6} cm^2/sec. The diffusion coefficient for boron in graphite at 2950°C used in safety analyses is 3×10^{-6}. This value is greater than the above experimental values, indicating that the boron redistribution calculations for the LOFC accident based on diffusion through graphite are conservative.

Table 3.35

DISTRIBUTION OF BORON IN FURNACE FOLLOWING 120-HR ANNEAL OF SAMPLE 4705-14 IT 2950°C

Specimen	Specimen Weight (g)	Boron Content (%)
Compact sample	0.0225	0.38
Graphite ram, above sample	0.0585	0.99
Compact crucible, around sample	0.194	3.28
Crucible holder	0.266	4.51
Graphite susceptor:		
Top cap	0.156	2.65
Middle	0.388	6.56
Bottom cap	0.087	1.47
Lampblack, 3 cm around susceptor	3.363	56.93
Remaining lampblack	1.372	23.22

In all of these high-temperature anneals, total sample weight losses invariably exceeded the amount of boron transported. Moreover, it was observed that nonboronated graphite control samples, heated under similar conditions, lost relatively little weight. This suggests that the vapor above boron carbide in graphite at high temperatures is composed of species containing both boron and carbon. Column 9 of Table 3.34 gives the apparent atom ratio (B/C) for the transported species. It is seen that the B/C ratio ranges from 0.9 to 3.5. (The B/C values of 8.0 and 6.0 obtained for samples 4705-38 and 4705-64, respectively, are believed to be anomalous.) This phenomenon is consistent with experimental data of other workers where the species B, BC, B_2C, and BC_2 have been observed to exist in the vapor phase at high temperatures.

Tests planned for the future include measurements of boron diffusion coefficients in H-327 graphite and boron vapor phase transport in a mocked-up control rod system.

Irradiation Testing

Irradiation experiments are being conducted to proof test TRISO and BISO coated particles, blended beds, and fuel rods under Fort St. Vrain design irradiation exposures. These experiments are intended to confirm the selection of the reference fuel designs and to demonstrate the performance of these materials under service conditions. To date a total of 96 TRISO samples and 129 BISO samples have demonstrated successful irradiation performance, as shown in Table 3.36.

During this report period, two full-exposure tests (P20 and P22) of TRISO coated particles completed irradiation, and postirradiation examination of one (P20) was completed. Five proof tests of TRISO particles, blended beds, and fuel rods to half (F-25, F-27, and F-28) and full exposure (F-26 and F-29) began irradiation. One test of a full-size blended bed (F-30) is being designed.

Capsule P20. Capsule P20 is the highest exposure test of TRISO coated particles ever conducted with significant burnup. The particles were irradiated to fast fluences up to 8.7×10^{21} n/cm^2 and to burnups up to 27% FIMA, both of which exceed the reactor maximums of 8×10^{21} n/cm^2 and 20% FIMA, respectively.

The capsule contained twelve different batches of TRISO particles. Each was tested at the maximum temperature (1300°C programmed down to 1100°C) and three were also irradiated at the median temperature (approximately 900°C), making a total of 15 tests.

The particles were designed to compare the irradiation performance of TRISO-I and TRISO-II coating designs with LTI and HTI outer PyC coatings. Various SiC layer thicknesses were included. Two of the samples had outer isotropic PyC coatings doped with silicon to determine the effectiveness of this advanced concept for further improving irradiation stability of the PyC layer. All of the particles were coated in laboratory-scale equipment.

PUBLIC SERVICE COMPANY OF COLORADO

P. O. BOX 840 · DENVER, COLORADO 80201

OSCAR R. LEE
VICE PRESIDENT

December 14, 1984
Fort St. Vrain
Unit No. 1
P-84530 12

Regional Administrator
Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Attention: Mr. Eric H. Johnson

DOCKET NO: 50-267

SUBJECT: Technical Specification Upgrade Program

REFERENCES:
1) NRC Letter, H. R. Denton to R. F. Walker, dated 10/16/84 (G-84392)
2) PSC Letter, O. R. Lee E. H. Johnson, dated 11/16/84 (P-84498)
3) NRC/PSC meeting on November 28 - 30, 1984

Dear Mr. Johnson:

Enclosed, for your information, are the Work Specification and Schedule that Public Service Company has developed for the Fort St. Vrain Technical Specification Upgrade Program. As discussed in References 1, 2 and 3, the program objective is to improve the accuracy, completeness, and clarity of the FSV Technical Specifications, and to provide a draft of the upgraded Technical Specifications to the NRC by April 1, 1985.

Attachment 1 is the Work Specification which provides the requirements and guidance for the review of the Final Safety Analysis Report (FSAR), and for the review and upgrading of the Technical Specifications.

Attachment 2 is the project schedule which has been developed to support the April 1, 1985 draft submittal date. The Technical Specifications have been grouped into forty-eight (48) subject categories or work packages, and various priorities have been assigned to each one, based on the degree of difficulty and complexity of the subject matter.

As discussed in the reference meeting, the overall schedule for submitting the upgraded Technical Specifications is as follows:

Provide draft Technical Specifications to NRC	-	April 1, 1985
NRC comments provided to PSC	-	May 1, 1985
Submit for PORC/NFSC approval	-	June 1, 1985
Submit proposed Technical Specifications to NRC	-	July 1, 1985

Public Service Company anticipates that further revisions and elaborations to the attachments will be required as the upgrade program develops.

If you have any questions or comments about the information contained herein, please contact Mr. M. H. Holmes at (303) 571-8409.

Very truly yours,

O. R. Lee, Vice President
Electric Production

ORL/JMG/kss

Attachments

REVIEWED BY: Scott Hofstetter

ATTACHMENT 1

WORK SPECIFICATION FOR
TECHNICAL SPECIFICATION
UPGRADE PROGRAM

PUBLIC SEF ICE COMPANY OF COLOF DO
FORT ST. VRAIN NUCLEAR GENERATING STATION

SPECIFICATION COVER SHEET

FORM 344 - 22 - 4082

SPECIFICATION FOR	PLANT ITEM NO'S.
WORK SPECIFICATION FOR TECHNICAL SPECIFICATION UPGRADE PROGRAM	N/A

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PAGE SCHEDULE

ATTACHMENTS

ISSUE SUMMARY

ISSUE	PREPARED BY	ENGRG. REVIEW	Q.A. REVIEW	APPROVED BY	DATE	BASIS FOR REVISION
A	[signature]	T.A. Gowling	[signature]	M.H. Holmes	12-14-84	Initial Issue

FORT ST. VRAIN
TECHNICAL SPECIFICATION UPGRADE PROGRAM

A. PURPOSE AND SCOPE

1. Purpose

The objective of the Fort St. Vrain Technical Specification Upgrade Program is to improve the accuracy, completeness, and clarity of the Fort St. Vrain Technical Specifications consistent with the licensing basis of the Fort St. Vrain plant as embodied in the Fort St. Vrain Final Safety Analysis Report (FSAR).

The purpose of this Work Specification is to provide requirements and guidance for the review of the FSAR and the review and revision of the Technical Specifications.

2. Scope

The Fort St. Vrain Technical Specification Upgrade Program consists of two parallel review efforts leading to preparation of a more accurate, complete, and clear set of Technical Specifications. The overall program plan is illustrated in Figure 1 and the project flow chart is shown in Figure 2.

The scope of this Work Specification includes the review and revision of the following existing Technical Specifications sections and associated subsections:

Section	1.0	Introduction
Section	2.0	Definitions
Section	3.0	Safety Limits and Limiting Safety System Settings
Section	4.0	Limiting Conditions for Operation
Section	5.0	Surveillance Requirements
Section	6.0	Design Features
Section	7.0	Administrative Controls

The Fort St. Vrain Final Safety Analysis Report (FSAR) shall be reviewed to the extent necessary to verify the bases for the existing Fort St. Vrain Technical Specifications. The FSAR shall also be reviewed to identify any omissions from the existing Technical Specifications.

The Standard Technical Specifications (STS) for Westinghouse light water reactors shall be considered during the performance of this work as indicated in the following sections of this Work Specification. However, conformance with the Standard Technical Specifications is not intended to be a requirement of this program, nor is substantiation or justification of any differences with STS requirements necessary. If instances arise whereby Fort St. Vrain's design features and Technical Specification requirements may represent a possible safety concern relative to STS requirements, those instances will be addressed as separate licensing issues outside the scope of the Fort St. Vrain Technical Specification Upgrade Program. Plant modifications and hardware backfits will not be undertaken to permit the adoption of any STS requirement.

It is outside the scope of the Fort St. Vrain Technical Specification Upgrade Program to utilize or consider any Standard Technical Specification requirement which opens the licensing basis of the Fort St. Vrain plant for further justification or analysis.

Significant research and development efforts or analytical investigations beyond those documented in the FSAR will not be undertaken to determine how or whether a Standard Technical Specifications requirement can be utilized at Fort St. Vrain. Questionable Standard Technical Specifications requiring such efforts and investigations will not be utilized or given further consideration.

B. WORK TO BE PERFORMED

1. Each existing FSV Technical Specification within the scope of this work specification shall be reviewed using the criteria described in Section C of this work specification.

2. The FSV Final Safety Analysis Report shall be reviewed using the criteria described in Section D of this work specification.

3. Upgraded FSV Technical Specifications shall be prepared as necessary according to the criteria described in the following sections and deficiencies identified during the reviews shall be corrected.

C. TECHNICAL SPECIFICATION REVIEW CRITERIA

1. General Criteria

a. The purpose of the technical specifications is to require that the overall facility status is consistent with the assumptions in the safety analysis. These assumptions deal with the following:

(1) Facility Physical Characteristics, i.e., features that are expected to remain constant.

(2) Status of Equipment, i.e., system and component operability.

(3) Operating State of Equipment, i.e., physical equipment parameters which concern system or component actions or the position or running condition of equipment.

(4) Values of Process Parameters i.e., flows, temperatues, pressures, etc..

(5) Condition of Equipment and Structures, i.e., the state of preservation of quality.

(6) Administrative controls (e.g. shift staffing, review and audit) that must be maintained.

b. Prior to establishing the technical specification, the basis shall be defined thereby establishing the rationale for the specification.

c. Technical specifications shall be provided only for items relied upon in the safety analysis, and for other items specifically required by Federal regulations to be in the technical specifications.

d. Technical specifications shall be written in a clear and concise manner with the intent that only one interpretation can be made. The use of vague terms such as "immediately" or "sufficiently" shall be avoided or defined to assure uniform interpretation by all auditors and operators.

e. Technical specifications shall be formulated such that compliance is physically possible based on the plant design, including test and measurement limitations.

f. Allowance for calculational inaccuracies and dynamic effects shall be considered.

g. Technical specifications shall clearly state the facility operating conditions (e.g. power operation, refueling) to which they appply. The operating conditions selected shall be limited to those conditions for which equipment must be operable or for which parametric limits exist due to assumptions of the safety analysis.

h. Values for parameters shall be specified in units directly available to the operating personnel, shall include allowable tolerances on the specified value, and shall include allowance for the effect of any associated instrument error, as appropriate.

i. Technical specifications shall preserve defense in depth (e.g. multiple barriers, redundancy, backup systems) only to the extent that it has been relied upon in the safety analysis.

j. Technical specifications shall preserve the single failure criterion to the extent relied upon in the safety analysis and may permit relaxation from this criterion for justifiable periods, for example as based on probability, reliability, previous analyses, or experience.

k. Adverse impact on plant availability shall be considered in the development of technical specifications consistent with the maintenance of an acceptable level of safety.

l. Technical specifications shall be developed such that on-site personnel exposure is as low as reasonably achievable while ensuring the health and safety of the public.

m. Incorporating requirements by references to the Final Safety Analysis Report, Federal regulations, or industry codes and standards shall be held to a minimum. Where utilized, these references shall be to the subdivision of the document rather than a general reference.

n. The selection of values for technical specifications shall be done by (a) deterministic methods, or (b) probabilistic and reliabilty methods. Probabilistic and reliability methods shall be utilized only when suitable justification is presented, and only on a case-by-case basis.

o. Technical specifications shall be stated in the simplest terms possible to clearly convey their meaning without ambiguity.

p. Technical specifications shall be reviewed and expanded, as necessary, to assure accuracy, completeness, and consistency with existing safety analysis documentation.

q. The technical specifications shall account for and utilize existing plant equipment and safety systems.

2. <u>Criteria for Bases for Technical Specifications</u>

a. Bases for technical specifications shall be summary statements of the reasons for such specifications and shall be provided for safety limits, limiting safety system settings, limiting conditions for operation, and surveillance requirements.

b. The bases shall explicitly correlate the plant design and safety analyses with the technical specification limits and operating conditions, thereby providing a validation of the overall design for the prescribed modes of operation.

c. The bases for technical specifications shall be developed with appropriate consideration of the following general requirements:

(1) The bases shall not contain requirements over and above those in the specification

(2) For each technical specification requirement, there shall be a corresponding and clearly identified basis which is solely related to an identified safety requirement.

(3) Where applicable, the bases shall identify the specific plant process condition which is controlling for the corresponding specification.

(4) The relationship between the values specified in the technical specification and those used in the safety analyses shall be provided in the bases.

(5) Errors, from instrumentation or other sources, assumed in the development of the technical specification limits shall be discussed in the bases to provide a clear relationship between the technical specification and the safety analysis values.

(6) The bases shall explain the rationale for the requirements in remedial action statements and the appropriateness of the condition restoration times relative to an acceptable level of safety.

(7) The sources of information summarized in the bases shall be cited.

(8) The justification contained in bases shall not be considered part of the Technical Specification requirements and may be changed by the licensee without prior NRC approval, providing that the change is evaluated and determined not to involve an unreviewed safety question.

3. <u>Criteria for Definitions</u>

a. The technical specifications shall include a list of definitions of terms which are frequently used within the document and which are not in general every day use. In addition, terms which have technical connotations, or terms which are applicable only to Fort St. Vrain should be included. These terms shall be explicit and clearly defined in simple and direct language with the intent that a uniform, unambiguous interpretation of the technical specifications can be achieved for facility operation and regulatory enforcement.

b. Relevant standard technical specification definitions shall be adopted where the definitions are consistent with existing plant features and the licensing basis of the plant, i.e.; FSAR terminology and analyses.

4. <u>Criteria for Safety Limits</u>

a. Safety limits shall be prescribed for selected process variables related to the integrity of barriers to fission product release. Compliance with safety limits shall provide assurance that the barrier will perform as assumed in the safety analysis.

b. The bases shall identify the barrier to fission product release that is being protected by the limit and show why that limit is adequate.

5. Criteria for Limiting Safety System Settings (LSSS)

a. Limiting safety system settings shall be defined to assure that no safety limit would be violated as a result of a frequent plant process condition and that no infrequent or limiting plant process condition would have consequences which do not meet the acceptance criteria for that condition.

b. Values for limiting safety system settings shall be based on the assumption that the facility is at or within its limiting conditions for operation when one of these process conditions occurs.

An adequate margin shall be provided between the limiting safety system settings and the safety limits so that safety limits would not be exceeded in the event that protective action is initiated if a limiting safety system setting is exceeded.

c. Conditions under which channels, features, and interlocks may be bypassed shall be specified either together with the relevant limiting safety system setting or with a relevant limiting condition for operation.

d. The bases for limiting safety system settings shall identify the safety limit or other safety requirement that is being ensured by the LSSS and shall describe all allowances included in determining the relationship of the LSSS to the safety limit or other safety requirement. The bases shall discuss the conditions under which the bypass of automatic protection associated with an LSSS is permitted.

6. Criteria for Limiting Conditions for Operation (LCO)

a. The limiting conditions for operation shall define the lowest functional capability or performance levels necessary to assure safe operation of the facility as evaluated in the FSAR accident and safety analyses.

b. Limiting conditions for operation shall be provided for the following when they are relied upon in the safety analysis:

(1) Condition, or status, of equipment or systems;

(2) Parameter limit with no associated instrument alarm or protective action setpoint;

(3) Instrument setpoints for monitored parameters with no associated automatic protective action (for this case, the LCO limit shall be the limiting value of the parameter, while the SR value shall be the instrument setpoint).

(4) Instrument setpoints for monitored parameters with associated automatic protective actions.

c. Each LCO shall include an applicability statement that clearly identifies the operating modes to which the LCO applies.

d. Values for limiting conditions for operation shall be consistent with extremes of initial conditions which have been shown to result in acceptable consequences for the various plant conditions as demonstrated by the safety analysis.

e. Included in the limiting conditions for operation shall be an action statement that describes the remedial action to be taken if:

(1) the operable status of equipment or systems is less than the required minimum;

(2) the monitored parameters are not within the specified range; or

(3) the instrument setpoints are less conservative that the specified value.

f. Remedial action statements shall specify the condition restoration time and shall require that, unless restoration is accomplished within that time, the facility be taken to a specified mode of operational safety consistent with the safety protection available from the remaining equipment or systems. The time interval allowed for each action shall be specified.

g. In developing remedial action requirements, consideration shall be given to: the operability of redundant or diverse systems; the probability of an event taking place during the condition restoration time which would be influenced by the limiting condition for operation; the reliability of the redundant or diverse systems; the risk of inducing an undesireable incident while performing the remedial action (for example, the thermal transients induced by a shutdown and cooldown); and the potential cost of complying with the proposed remedial action versus the benefits thus derived.

h. The allowable condition restoration times shall be established based on level of equipment availability required to assure an acceptable level of safety and should consider events that will reduce the level of availability such as surveillance and maintenance.

i. When necessary to preserve acceptable channel or train availability, condition restoration time requirements shall include establishment of cummulative downtime limits.

j. Each LCO shall include a cross-reference to the surveillance requirements that support the LCO, except that surveillance shall not be required if the normal operating status of equipment or systems, for the applicable operational modes, equals or exceeds the lowest functional capability of performance level relied upon in the safety analysis.

k. The bases for limiting conditions for operation shall identify the safety analysis assumption or other safety requirement that establishes the need for the LCO, and shall discuss why the specified lowest functional capability, performance level of equipment, limiting value of a process parameter, or conservative actuation limit for specified automatic protection devices is appropriate. The rationale for deviations from the specified conditions as allowed by remedial action statements shall also be discussed.

7. Criteria for Surveillance Requirements (SR)

a. Surveillance requirements shall delineate testing, calibration, monitoring, and inspection in sufficient scope, depth, and frequency to provide assurance that equipment, systems and process variables are within limiting conditions for operation. Each limiting condition for operation shall be supported by a surveillance requirement except where the normal operating status of equipment or systems, for the applicable operational modes, equals or exceeds the lowest functional capability or performance level relied upon in the safety analysis. Every surveillance requirement shall be cross-referenced to a limiting condition for operation, or to an administrative control.

b. Minimum disturbance of normal plant operation should be assured by relating surveillance requirements to normal operational cycles such as the refueling period, where practical.

c. Customary surveillance scopes, depths and frequency which have been found compatible with an acceptable level of safety shall be employed unless sufficient design, operation, or research information suggests alternate approaches. The Standard Technical Specifications may be used for guidance in this regard.

d. The surveillance program shall demonstrate acceptable availability for equipment for which there is limited experience or reliability data. (A sliding surveillance frequency can be established by choosing an initial surveillance frequency with provision to lengthen or shorten the time between tests based on experience gained with the equipment involved).

e. The surveillance shall be consistent with the requirements of recognized and relevant industry codes and standards.

f. Where it is not obvious that the surveillance supports the LCO, the bases shall describe how the specified surveillance will assure compliance with the LCO. The rationale for the surveillance frequency shall be identified to facilitate consistent modifications to the frequencies where warranted by plant performance.

8. Criteria for Design Features (DF)

   a. Design features of the facility which, if altered or modified, could have significant effects on safety and are not covered by the safety limits, limiting conditions for operation, or surveillance requirements shall be incorporated in the design features section of the technical specifications.

   b. Particular sections or criteria of the FSAR may be referenced as an alternative to providing design details in the technical specifications; however, such references should be limited and specific since referenced criteria or features will become part of the technical specifications and cannot be changed under the provision of Title 10, Code of Federal Regulations, Part 50, "Licensing of Production and Utilization Facilities," Section 50.59, "Changes, Tests and Experiments," without Commission approval. References to the FSAR that provide further information but are not intended to be part of the technical specification, should be located in the bases.

   c. Provisions should also be included to allow for normal degradation of design features where applicable.

9. Criteria for Administrative Controls

   a. Administrative controls shall be included in the technical specification to assure that operation of the facility is conducted in a safe manner. Implicit in this are the requirements for: organization; procedures; record keeping; review; audit; reporting; staffing qualifications and resolution of safety limit violations.

   b. Specific responsibility and authority shall be delineated for those portions of the organization charged with fulfilling these requirements.

   c. The administrative controls shall also require that the facility procedures include those operator actions relied upon in the safety analysis.

   d. Additional guidance can be found in other standards and regulatory guides for:

(1) Administrative controls: American National Standard, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plant," N18.7-1976/ANS-3.2;

(2) Selection and training of personnel: American National Standard, "Selection and Training of Nuclear Power Plant Personnel," N18.1-1971/ANS-3.1; and

(3) Reporting of operating information: Regulatory Guide 1.16, Revision 4, August 1975, "Reporting of Operating Information - Appendix A, Technical Specifications.

D. FSAR REVIEW CRITERIA

1. The entire Fort St. Vrain Final Safety Analysis Report shall be reviewed to identify the underlying assumptions used to determine that operation of the plant does not present an undue risk to the health and safety of the public.

2. The essential safety functions that protect the health and safety of the public are those related to:

   a. Protecting the integrity of fission product boundaries.

   b. Controlling reactivity.

   c. Cooling the fuel.

   d. Limiting the release of radioactive fission products, and

   e. Mitigating the consequences of accidents and natural and manmade phenomena.

3. The underlying assumptions to be identified consist of:

   a. Values of process variables that must be kept within certain bounds.

   b. Operating state of equipment that must be maintained.

   c. Operating status (or operability) of equipment that must be maintained.

   d. Condition (or quality) of equipment and structures that must be maintained.

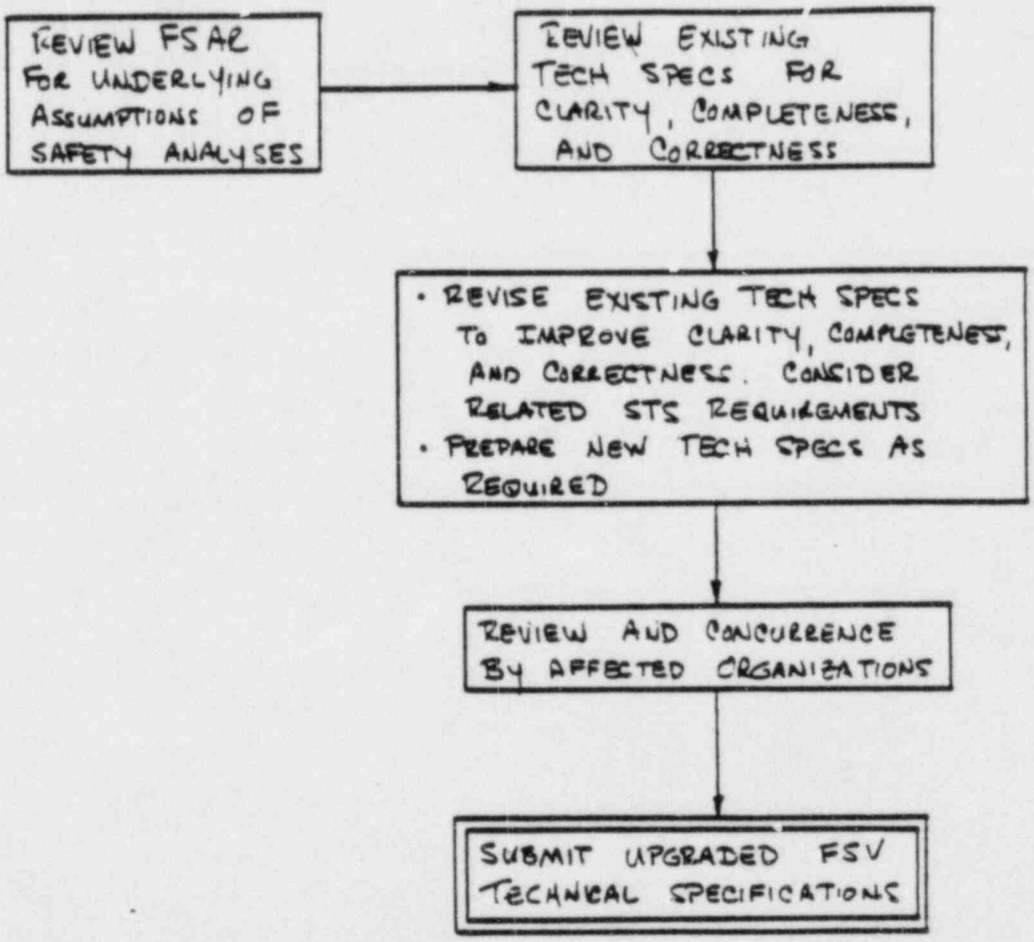
e. Physical characteristics of the plant that must remain fixed, and

f. Administrative controls that must be maintained.

4. Underlying assumptions that are expected to, or could vary with time or circumstances, throughout the life of the plant shall be identified as being subject to technical specification control. A list of these items shall be forwarded to the Program Coordinator.

TECHNICAL SPECIFICATION UPGRADE PROGRAM

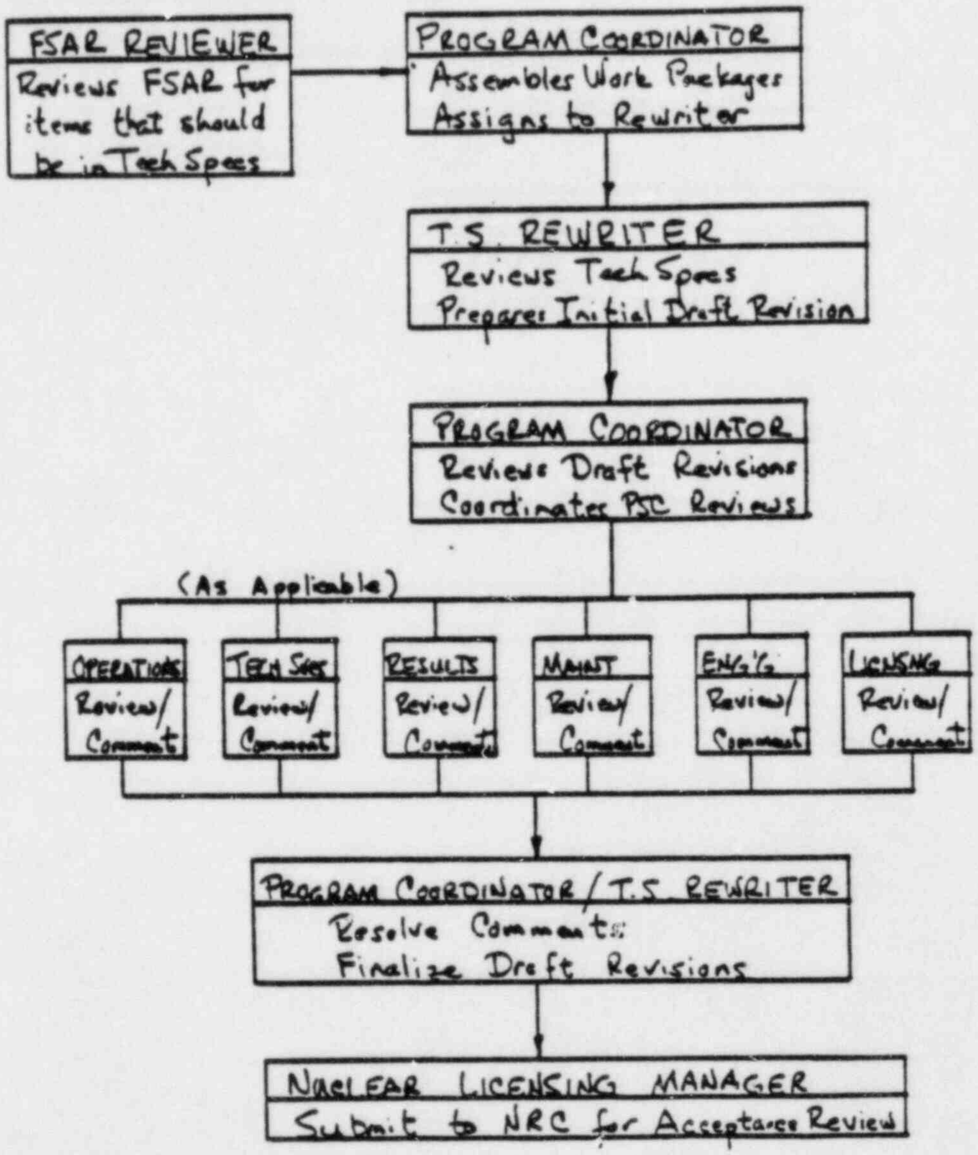
OVERALL PROGRAM PLAN



ATTACHMENT 1

TECHNICAL SPECIFICATION UPGRADE PROGRAM

PROJECT FLOW CHART



ATTACHMENT 2

ATTACHMENT 2

TECHNICAL SPECIFICATION
UPGRADE PROGRAM
PROJECT SCHEDULE

TECHNICAL SPECIFICATION UPGRADE PROGRAM

PROJECT SCHEDULE

The Technical Specification Upgrade Program includes two parallel reviews, one to review the FSAR for items that should be included in the Technical Specifications, and the other to review the Technical Specifications for accuracy, completeness, and clarity.

The FSAR review is scheduled to begin on December 17, 1984, and is to be completed by February 15, 1985, as follows:

FSAR SECTION	COMPLETION DATE
1, 2 & Appendix G	1/04/85
3 & 4 & Appendix A	1/11/85
5 & 6 & Appendix E	1/18/85
7 & 8	1/25/85
9 & 10 & Appendix H & I	2/01/85
11, 12 & 13	2/08/85
14 & Appendix D	2/15/85

The Technical Specification review has been initiated. The Technical Specifications have been grouped into work packages, based on their subject matter, and priorities have been assigned for their completion. The schedule for work package completion is as follows:

High Priority work packages are expected to be completed from January 15, 1985 through March 1, 1985;

Medium Priority work packages are expected to be completed from February 1, 1985 through March 15, 1985; and

Low Priority work packages are expected to be completed from February 15, 1985 through March 20, 1985.

The work packages, their assigned priority, and the Technical Specification sections that they include are listed below.

Work Package	Description	Section/LCO	SR
High Priority			
2	Definitions	2.0	
3	Reactor Core	3.1, part 3.3	5.1.6
4	Reactor Vessel	3.2, part 3.3	
5	Core Irradiation	4.1.1	
6	Control Rods	4.1.2 4.1.3 4.1.4	5.1.1 5.1.5
7	Reactivity	4.1.5 4.1.6 4.1.8	5.1.2 5.1.3 5.1.4
8	Inlet Orifice Valves	4.1.7 4.1.9	5.1.7 5.4.3 5.4.8
9	Primary Coolant System	4.2.1 4.2.2 4.2.3 4.2.4 4.2.5 4.2.19	5.2.7 5.2.8 5.2.9 5.2.18 5.2.23 5.2.24 5.2.27
17	Steam Generators/ Safe Shutdown Cooling	4.3.1 4.3.2 4.3.4 4.3.5 4.3.6 4.3.7	5.2.24 5.3.2 5.3.4 5.3.3 5.3.5 5.3.6 5.3.7 5.3.10
20	PPS Instrumentation	4.4.1	5.4.1 5.4.3 5.4.4 5.4.5 5.4.6 5.4.7 5.4.8

Work Package	Description	Section/LCO	SR
24	Analytical/PPS Moisture Monitors	4.4.5 4.9.2	5.4.12
26	Auxiliary Electrical	4.6.1	5.6.1 5.6.2
Medium Priority			
10	Firewater Systems	4.2.6 4.10.5 4.10.7 4.10.8	5.2.10 5.10.6 5.10.8 5.10.9
11	PCRV Pressurization	4.2.7 4.2.9 4.5.2 4.7.1	5.2.1 5.2.13 5.2.14 5.2.15 5.2.16 5.2.28 5.4.1 5.3.9
12	Primary/Secondary Activity	4.2.8 4.3.8	5.2.6 5.2.11 5.3.7
13	Loop Impurity Levels	4.2.10 4.2.11	5.2.12 5.2.22 5.2.25 5.4.12
15	PCRV Liner Cooling	4.2.13 4.2.14 4.2.15	5.4.4 5.4.5 5.4.11
18	Steam Water Dump Tank	4.3.3	5.3.1 5.3.7
31	Room Isolation Damper/ Halon Fire Suppression	4.10 1 4.10.2	5.10.1 5.10.2
32	Smoke Detectors	4.10.3	5.4.2 5.10.3
33	Fire Barrier Penetration Seals	4.10.4	5.10.4

Work Package	Description	Section/LCO	SR
35	Tendon Surveillance		5.2.2 5.2.3
36	Concrete Surveillance		5.2.4
37	Liner Specimen Surveillance		5.2.5
38	RCD Surveillance		5.2.26
39	S/G Bimetallic Welds/ Tubeleaks		5.3.11 5.3.12
41	Design Features	6.1 6.2.1 6.2.2 6.2.3 6.3	
42	Administrative Controls	7.1 7.1.1 7.1.2 7.1.3	
43	Safety Limits	7.2	
44	Records	7.3	
45	Procedures	7.4	
46	Reporting Requirements	7.5 7.5.1 7.5.2 7.5.3	
47	Environmental Qual-ification	7.6	
48	Depressurization/Helium Purification	4.2.18	

ATTACHMENT 2

Comparison of Reserve Shutdown

Material Purchase Specifications

Purchase Specification		Manufacturer	Material Purchased
12-D-1	Revision B	Union Carbide Corporation	7/16" and 9/16" (original)
12-D-1	Revision C	Advanced Refractory Technologies	7/16" and 9/16" (1982-1983)
12-D-14	Revision A	Advanced Refractory Technologies	7/16" and 9/16" (1984-1985)

SECTION	12-D-1 Issue B	12-D-1 Issue C	12-D-14 Issue A
2.1.2	Boron Density Tolerance ±.03 in a ball	Average Boron Density Tolerance ±.03 for all samples of a production lot; Boron Density Tolerance ±10% in a ball	Boron Density Tolerance ±.03 in a ball
2.2	B_2O_3 <1% of ball weight	B_2O_3 <1% of ball weight	B_2O_3 <.25% of ball weight for 9/16 B_2O_3 <.15% of ball weight for 7/16
2.2	Concentration of elements other than Boron or Carbon $\leq$ 1 wt%	Concentration of elements other than Boron, Carbon, Oxygen and Iron $\leq$ 1 wt%	Concentration of elements other than Boron, Carbon, Oxygen and Iron $\leq$ 1 wt%
2.3.2	Bulk Density $\geq$ 1.80 gm/cc for all balls	Bulk Density $\geq$ 1.50 gm/cc for 9/16" diameter balls Bulk Density $\geq$ 1.35 gm/cc for 7/16" diameter balls	Bulk Density $\geq$ 1.50 gm/cc for 9/16" diameter balls Bulk Density $\geq$ 1.35 gm/cc for 7/16" diameter balls
2.4.2	The same ball used for density testing shall be used to determine Boron and B_2O_3 concentration	The same ball used for density testing shall be used to determine Boron concentration; this same ball or others from the same sub-group will be used to determine B_2O_3 Fe, and other impurity values; a minimum of 6 balls will be tested for other impurities; 2 additional balls will be retested if any individual ball fails.	The same ball used for density testing shall be used to determine Boron concentration; this same ball or others from the same sub-group will be used to determine B_2O_3, Fe and other impurity values; a minimum of 6 balls will be tested for other impurities; 2 additional balls will be retested if any ball fails.
2.7	Acceptance by the Buyer of each lot shall be subject to tests and analyses by the Buyer for all requirements of this specification	Acceptance by the Buyer of each lot shall be subject to tests and analyses by the Buyer for all requirements of this specification except as provided in paragraph 2.4.2	Acceptance by the Buyer of each lot shall be subject to tests and analyses by the Buyer for all requirements of this specification except as provided in paragraph 2.4.2
2.9.5	- - -	Boronated balls must be kept away from nuclear fuel.	Boronated balls must be kept away from nuclear fuel.

Section	12-D-1 Issue B	12-D-1 Issue C	12-D-14 Issue A
2.9.6	---	All packaging, shipping receiving, storage, and handling requirements shall be per Attachment 7.10, Level B.	All packaging, shipping, receiving, storage, and handling requirements shall be per Attachment 7.10, Level B.
3.1.1	B_4C particles < 50 mesh and > 325 mesh; B_4C chemical composition: Boron-70-76 wt%, B_2O_3, < 3 wt%, Boron + Carbon $\geq$ 94 wt%, Iron $\leq$ 2 wt%, all other impurities $\leq$ 4 wt%	B_4C particles < 50 mesh; B_4C chemical composition: Boron-70-76 wt%, B_2O_3 < 3 wt%, Boron + Carbon $\geq$ 94 wt%, Iron $\leq$ 2 wt%, all other impurities $\leq$ 4 wt%.	B_4C particles < 50 mesh
3.1.3	The binder shall be coal tar pitch or other material approved by Buyer.	The binder shall be phenolic resin or other material approved by Buyer.	The binder shall be phenolic resin or other material approved by Buyer.
3.3	The balls shall be baked in an inert atmosphere at a temperature of 3400 ±100°F for a minimum of two hours.	The balls shall be heated in an inert atmosphere. The target temperature of the bake shall be greater than 1820°C, but less than or equal to 2180°C, with no individual reading greater than 2250°C for more than 15 minutes. Refiring: In the event that balls initially heat treated do not pass the drop tests specified in 2.4.1, the lot in question may undergo one complete refiring per the above initial heat treatment conditions. If, after this refiring, the balls do not pass the drop tests specified in 2.4.1, the lot shall be rejected.	The balls shall be heated in an inert atmosphere at a temperature between 2600°F and 3500°F for a minimum of 60 minutes of which at least 15 minutes will be at 3400±100°F.

ATTACHMENT 3

Purchase Specifications

12-D-1, Revision B and 12-D-14, Revision A

GGA FORM 317 9-67

Notations in this column indicate where changes have been made.

PRODUCT SPECIFICATION FOR BORONATED GRAPHITE BALLS OF THE RESERVE SHUTDOWN SYSTEM

Specification Prepared By: O.M. Stansfield 6-2-69
O. M. Stansfield

Approved By: W.V. Goeddel 6/3/69
W. V. Goeddel - Materials Branch

E.O. Winkler 6/6/69
E. O. Winkler - Fuel Element Design Branch

R.C. Dahlberg
R. C. Dahlberg - NARP

W.E. Bell 6/13/69
W. E. Bell - Chemistry Branch

W.L. Wyman 6/16/69
W. L. Wyman - Fuel Operations Division

W.P. Wallace 6/17/69
W. P. Wallace - Quality Assurance

R.F. Turner 6/19/69
R. F. Turner - Fuel Development

C.F. Fox 6/23/69
C. F. Fox - Project Manager

GGA FORM 317 9-67

Notations in this column indicate where changes have been made.

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Gulf General Atomic
Incorporated

Specification For Boronated Graphite Balls - Reserve Shutdown System			
Proj. No. 90	Spec. No. 12-D-1	Issue B	Date 5-26-69

ISSUE SUMMARY

Issue	Date	Prepared by	Approved by	Purpose of Issue Sections Changed
A	3/28/68	T. Macken		Inquiry
B	5/26/69	O. Stansfield	See Page 1	Inquiry

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Notations in this column indicate where changes have been made.

Equipment No. D-1201	Page 3 of 8

Notations in this column indicate where changes have been made.

GGA FORM 317 9-67

1.0 SCOPE

This specification establishes requirements for boronated graphite balls for use as the reserve shutdown material in the 330 MW(e) High Temperature Gas Cooled Reactor (HTGR) Power Plant to be constructed for the Public Service Company of Colorado at Fort St. Vrain. The reactor site is located approximately four miles northwest of Platteville, Colorado.

2.0 PRODUCT REQUIREMENTS

2.1 Design Features and Boron Content

2.1.1 Description of Components. The boronated graphite balls form part of the reserve shutdown system for the reactor by providing a neutron absorber material which is released into 37 channels (or a portion thereof) in the reactor core. The spheres are normally stored in hoppers located in each of the 37 refueling penetrations in the top head of the reactor vessel. Activation of the reserve shutdown system releases the balls into the core by rupturing a retaining disk in the bottom end of the hopper. High pressure helium gas injected into the hopper causes the disk to rupture and releases the balls from the hopper.

2.1.2 Boron Content and Dimensions. The balls shall be boronated graphite containing natural boron in the form of boron (B_4C). Two types of balls designated Type A and Type B, shall be fabricated. The differences between the two types shall be (1) boron content and (2) diameter. The boron density in a ball and the diameters of the two ball types is shown in the following table with the required tolerances.

Type Ball	Diameter[(a)] (Inches) ± 1/32	Boron Density In a Ball (gm Boron/cc Ball) ± 0.03
A	7/16	0.32
B	9/16	0.66

(a) The ratio of major to minor axes must be less than 1.15. Any flashing or mold mark must be included in the measured diameter and shall not protude more than 1/32 inch above the ball surface.

The boron density in the balls shall be determined by the following equation:

(Boron Concentration in Weight Fraction) (Bulk Density, gm/cc) = (Boron Density, gm Boron/cc Ball).

GGA FORM 317 9-67

Notations in this column indicate where changes have been made.

2.2 Chemical Purity of Finished Product

The concentration of B_2O_3 must be less than 1% of ball weight. The coneentration of elements other than boron and carbon must not exceed 1 wt-%. The iron impurity level must be less than 0.5 wt-%. The balls must be free of contaminants such as dirt, grease, and wax or other foreign material associated with manufacture or storage.

2.3 Physical Properties

2.3.1 Strength. The structural integrity and impact strength of the balls shall be sufficient to satisfy the test conditions specified in paragraph 2.4.1.

2.3.2 Density. The bulk density of the balls shall be equal to or greater than 1.80 gm/cc.

2.4 Qualification (Chemical and Physical Property Testing)

A sampling of balls from each lot processed shall be subjected to chemical and mechanical tests by the seller to confirm that composition and performance requirements have been met. Records and test reports shall be maintained as specified in paragraph 2.8.

2.4.1 Impact Strength. The number of balls selected for impact strength tests shall be determined by lot size. The following table shall be used to determine sample size.

Number of Balls in Lot	Number of Balls in Sample
1201 to 3200	125
3201 to 10,000	200
10,001 to 35,000	315
35,001 to 150,000	500
150,000 to 500,000	800

The test procedure shall be as follows:

Each sample shall be subjected to two consecutive free fall drops of the entire sample in a continuous cascade from a height of 30 feet into a 3.75 inch diameter closed end hole in a block of PGX or similar grade of commercial graphite. After the second drop, all dust, fragments and balls of the sample shall be sieved on a number 3 (U.S.) sieve (0.265 inch opening). The lot shall be rejected if the weight of material passing through the sieve is greater than 0.5% of the original sample weight. Portions of the sample not subsequently used as described in paragraph 2.4.2 shall be sent to the Buyer in accordance with paragraph 2.9.3.

2.4.2 Density and Chemical Analysis. Each sample of balls selected for impact testing in accordance with 2.4.1 will be divided into five equal sub-groups after impact testing and the density of one ball from each sub-group will be determined in accordance with ATSM C-559-65T. The same ball will then be subjected to chemical analysis to determine boron and B_2O_3 concentrations. In the event that there are less than 20 separate lots of balls, the number of balls selected from each sub-group will be increased so that a minimum of 100 balls are analyzed separately. Density, boron, B_2O_3 and Fe concentration of each ball shall be within the limits specified in paragraphs 2.1.2, 2.2, and 2.3.2.

2.4.3 Quality Control. The manufacturer shall submit for the written approval of the Buyer, detailed testing and quality control procedures, including the requirements of paragraph 2.4.1 and 2.4.2, prior to start of production.

2.5 Definition of "Lot"

The term "lot" shall designate the balls produced from a single blend of raw materials, shaped under identical conditions, and baked at the same time in the same furnace.

2.6 Change Approval

Any deviation or change in the product specifications shall require prior written approval of the Buyer.

2.7 Acceptance

Acceptance by the Buyer of each lot shall be subject to tests and analyses by the Buyer for all requirements of this specification.

If a sample from a lot is outside of specification limits for any requirements, two additional samples and subsequent tests or analyses shall be conducted. The failure of either of the two additional samples to meet specification requirements shall be cause for rejection of the entire lot.

2.8 Records and Test Reports

2.8.1 The seller shall maintain records identifying the raw materials used in each batch of material.

2.8.2 The system of labeling of lots of balls and their respective test samples shall provide that the Buyer can readily identify each batch and lot with the corresponding samples tested and analyzed under provisions of this specification. Records of the tests and analysis shall be available for Buyer's inspection for two years after the balls have been accepted.

2.8.3 The Seller shall furnish to Buyer 5 copies of certified test reports listing in tabular form all results of the test specified in section 2.4. The results of all tests and analyses shall be traceable to the lot and sample tested.

2.9 Packing, Marking, and Shipping

2.9.1 The Seller shall provide suitable packaging containers for the balls to permit safe, contamination-free shipment and to insure protection during covered storage at the reactor site. The Seller shall prepare a packaging procedure to be approved by the Buyer not less than 120 days prior to shipment of the material.

2.9.2 Each package of balls shipped shall contain material from one lot only.

2.9.3 The unused portion of test samples discussed in paragraph 2.4 shall be packed in separate containers clearly marked "Sample Material" for each lot and sent to the Buyer no later than the date of shipment of the production lot.

2.9.4 Each shipping container shall be labeled with the specification number, including revision numbers, batch and lot number, purchase order number, gross, tare, and net weights and Seller's name and plant location.

3.0 PROCESS REQUIREMENTS

3.1 Raw Material

3.1.1 Boron Carbide. A certified chemical analysis of the boron carbide raw material shall be made by the Seller using standard ASTM analytical methods. The B_4C shall have the following chemical composition

Boron 70 to 76 wt-%	B_2O_3 < 3 wt-%
Boron + Carbon $\geq$ 94 wt-%	Fe $\leq$ 2 wt-%
	All other impurities $\leq$ 4 wt-%

The B_4C particle size shall be less than 50 mesh (U.S.) and greater than 325 mesh (U.S).

3.1.2 Filler. The filler shall be a graphite flour which has been graphitized at 2700°C for a minimum of 1 hour prior to blending in the mix. The ash content shall be less than 1 wt-%.

3.1.3 Binder. The binder shall be coal tar pitch or other material approved by the Buyer.

3.2 Mixing and Forming

The boronated graphite shall be fabricated in a manner which assures a uniform dispersion of boron within a ball. The vendor shall select, subject to approval of the Buyer, a technique for mixing and forming, which assures satisfactory strength, uniformity of the boron content, and shape of the balls.

3.3 Heat Treatment

The balls shall be baked in an inert atmosphere at a temperature of 3400 ± 100°F for a minimum of two hours.

Gulf General Atomic
Incorporated

Specification for Boronated Graphite Balls - Reserve Shutdown System			
Proj. No. 90	Spec. No. 12-D-1	Issue B	Date 5-26-69

VENDOR'S DATA (NOTE: IDENTIFY ALL DATA WITH GULF GENERAL ATOMIC PURCHASE ORDER NUMBER, SERVICE AND EQUIPMENT NUMBER.)	QUOTATION COPIES		APPROVAL COPIES		FINAL COPIES	
	PRINTS	DIAZO-MASTERS	PRINTS	DIAZO-MASTERS	PRINTS	DIAZO-MASTERS
OUTLINE DIMENSIONS AND FOUNDATION REQUIREMENTS						
CROSS SECTIONAL DRAWINGS WITH COMPLETE PARTS LIST						
PIPING DRAWINGS						
ELECTRICAL DRAWINGS						
INSTRUMENT DRAWINGS INCLUDING MOUNTING DETAILS						
SHOP DETAIL AND ERECTION DRAWINGS						
COMPLETED GULF GENERAL ATOMIC DATA SHEETS						
MATERIAL SPECIFICATIONS			3	1	3	1
MATERIAL CERTIFICATIONS					3	1
REPRESENTATIVE PERFORMANCE DATA						
CERTIFIED PERFORMANCE DATA FROM ACTUAL TESTS					3	1
VENDOR'S DESIGN REPORTS AND CALCULATIONS						
VESSEL STRESS REPORT						
MANUFACTURERS' DATA REPORTS FOR VESSELS						
MANUFACTURERS' TEST REPORTS FOR SAFETY VALVES						
FABRICATION PROCEDURES			3	1	3	1
WELDING PROCEDURES						
NONDESTRUCTIVE EXAMINATION PROCEDURES			3	1	3	1
QUALITY CONTROL PROCEDURES			3	1	3	1
INSTALLATION, OPERATION AND MAINTENANCE INSTRUCTIONS						
RECOMMENDED SPARE PARTS AND SPECIAL TOOLS						
Chemical Analysis procedures & Certification			3	1	3	1
Packaging ~~Marking~~ & Shipping Procedures			3	1	3	1

Equipment No. D-1201

Notations in this column indicate when changes have been made

PUBLIC SERVICE COMPANY OF COLORADO

FORT ST. VRAIN NUCLEAR GENERATING STATION

FORM 344 - 22 - 4082

SPECIFICATION COVER SHEET

SPECIFICATION FOR	PLANT ITEM NO'S.
BORONATED GRAPHITE BALLS - RESERVE SHUTDOWN SYSTEM	

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ISSUE SUMMARY

ISSUE	PREPARED BY	ENGRG. REVIEW	Q.A. REVIEW	APPROVED BY	DATE	BASIS FOR REVISION
A	J. Saltmarsh	[signature]	[signature]	[signature]	1-16-85	CN-1966

1.0 SCOPE

This specification establishes requirements for boronated graphite balls for use as the reserve shutdown material in the 330 MW(e) High Temperature Gas Cooled Reactor (HTGR) Power Plant owned by Public Service Company of Colorado at Fort St. Vrain. The reactor site is located approximately four miles northwest of Platteville, Colorado.

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2.1.2 Boron Content and Dimensions. The balls shall be boronated graphite containing natural boron in the form of boron (B_4C). Two types of balls, designated Type A and Type B, shall be fabricated. The differences between the two types shall be (1) boron content and (2) diameter. The boron density in a ball and the diameters of the two ball types are shown in the following table with the required tolerances.

Type Ball	Diameter[a] (Inches) ± 1/32	Boron Density In a Ball (gm Boron/cc Ball) ± 0.03
A	7/16	0.32
B	9/16	0.66

(a) The ratio of major to minor axes must be less than 1.15. Any flashing or mold mark must be included in the measured diameter and shall not protrude more than 1/32 inch above the ball surface.

The boron density in the balls shall be determined by the following equation:
(Boron Concentration in Weight Fraction) (Bulk Density, gm/cc) = (Boron Density, gm Boron/cc Ball).

2.2 Chemical Purity of Finished Product

The concentration of B_2O_3 must be less than .25% of ball weight for 9/16 and .15% of ball weight for 7/16. The concentration of elements other than boron, carbon, oxygen, and iron must not exceed 1 wt-%. The iron impurity level must be less than 0.5 wt-%. The balls must be free of contaminants such as dirt, grease, and wax or other foreign material associated with manufacture or storage.

2.3 Physical Properties

2.3.1 Strength. The structural integrity and impact strength of the balls shall be sufficient to satisfy the test conditions specified in paragraph 2.4.1.

2.3.2 Density. The bulk density of the balls shall be equal to or greater than 1.5 gm/cc for 9/16 and 1.35 gm/cc for 7/16.

2.4 Qualification (Chemical and Physical Property Testing)

A sampling of balls from each lot processed shall be subjected to chemical and mechanical tests by the seller to confirm that composition and performance requirements have been met. Records and test reports shall be maintained as specified in paragraph 2.8.

2.4.1 Impact Strength. The number of balls selected for impact strength tests shall be determined by lot size. The following table shall be used to determine sample size:

Number of Balls in Lot	Number of Balls in Sample
1,201 to 3,200	125
3,201 to 10,000	200
10,001 to 35,000	315
35,001 to 150,000	500
150,001 to 500,000	800

The test procedure shall be as follows:

Each sample shall be subjected to two consecutive free fall drops of the entire sample in a continuous cascade from a height of 30 feet into a 3.75 inch diameter closed end hole in a block of PGX or similar grade of commercial graphite. After the second drop, all dust, fragments and balls of the sample shall be sieved on a number 3 (U.S.) sieve (0.265 inch opening). The lot shall be rejected if the weight of material passing through the sieve is greater than 0.5% of the original sample weight. Portions of the sample not subsequently used as described in paragraph 2.4.2 shall be sent to the Buyer in accordance with paragraph 2.9.3.

2.4.2 Density and Chemical Analysis. Each sample of balls selected for impact testing in accordance with 2.4.1 will be divided into five equal sub-groups after impact testing and the density of one ball from each sub-group will be determined in accordance with ASTM C-559-65T. The same ball will then be subjected to chemical analysis to determine boron concentration. To determine B_2O_3, Fe, and other impurity values, this same ball or others from the same sub-group will be utilized. In the event that there are less than 20 separate lots of balls, the number of balls selected from each sub-group will be increased so that a minimum of 100 balls are analyzed separately. To determine the concentration of other impurities, a minimum of six balls will be randomly selected throughout production and tested. Density, boron, B_2O_3 and Fe concentration of each ball shall be within the limits specified in paragraphs 2.1.2, 2.2, and 2.3.2. In the event that any individual ball fails to meet the requirements above, two additional balls from the same sub-group will be selected and retested for the requirement in question. If either retested ball fails, the Seller must submit a written request for deviation or scrap the lot.

2.4.3 Quality Control. The manufacturer shall submit for the written approval of the Buyer, detailed testing and quality control procedures, including the requirements of paragraph 2.4.1 and 2.4.2, prior to start of production.

2.5 Definition of "Lot"

The term "lot" shall designate the balls produced from a single blend of raw materials, shaped under identical conditions, and baked at the same time in the same furnace.

2.6 Change Approval

Any deviation of change in the product specifications shall require prior written approval of the Buyer.

2.7 Acceptance

Acceptance by the Buyer of each lot shall be subject to tests and analyses by the Buyer for all requirements of this specification except as provided in paragraph 2.4.2. If a sample from a lot is outside of specification limits for any requirements, two additional samples and subsequent tests or analyses shall be conducted. The failure of either of the two additional samples to meet specification requirements shall be cause for rejection of the entire lot.

2.8 Records and Test Reports

2.8.1 The seller shall maintain records identifying the raw materials used in each batch of material.

2.8.2 The system of labeling the lots of balls and their respective test samples shall provide that the Buyer can readily identify each batch and lot with the corresponding samples tested and analyzed under provisions of this specification. Records of the tests and analyses shall be available for the Buyer's inspection for two years after the balls have been accepted.

2.8.3 The Seller shall furnish to the Buyer 5 copies of certified test reports listing in tabular form all results of the test specified in Section 2.4. The results of all tests and analyses shall be traceable to the lot and sample tested.

2.9 Packing, Marking, and Shipping

2.9.1 The Seller shall provide suitable packaging containers for the balls to permit safe, contamination-free shipment and to insure protection during covered storage at the reactor site. The Seller shall prepare a packaging procedure to be approved by the Buyer not less than 120 days prior to shipment of the material.

2.9.2 Each package of balls shipped shall contain material from one lot only.

2.9.3 The unused portion of test samples discussed in paragraph 2.4 shall be packed in separate containers clearly marked "Sample Material" for each lot and sent to the Buyer no later than the date of shipment of the production lot.

2.9.4 Each shipping container shall be labeled with the specification number, including revision numbers, batch and lot number, purchase order number, gross, tare, and net weights and the Seller's name and plant location.

2.9.5 Boronated balls must be kept away from nuclear fuel.

2.9.6 All packaging, shipping, receiving, storage, and handling requirements shall be per Attachment 7.10, Level B.

3.0 PROCESS REQUIREMENTS

3.1 Raw Material

3.1.1 Boron Carbide. The source of boron shall be B4C powder with particle size less than 50 mesh (U.S.).

3.1.2 Filler. The filler shall be a graphite flour which has been graphitized at 2700°C for a minimum of 1 hour prior to blending in the mix. The ash content shall be less than 1 wt-%.

3.1.3 Binder. The binder shall be phenolic resin or other material approved by the Buyer.

3.2 Mixing and Forming

The boronated graphite shall be fabricated in a manner which assures a uniform dispersion of boron within a ball. The vendor shall select, subject to approval of the Buyer, a technique for mixing and forming, which assures satisfactory strength, uniformity of the boron content, and shape of the balls.

3.3 Heat Treatment

The balls shall be heated in an inert atmosphere at a temperature between 2600°F and 3500°F for a minimum period of 60 minutes of which at least 15 minutes will be at 3400° ±100°F.

4.0 Quality Assurance and Documentation

4.1 The Seller's and his sub vendors' work and material supplied for this order shall be in accordance with the requirements of 10CFR50 Appendix B. The Seller shall not start fabrication until the Seller's Quality Assurance Manual, Quality Assurance Plan, and Quality Assurance Acceptance Procedures have been reviewed and approved by PSC. The above documents shall be submitted to Public Service Co. c/o Quality Assurance Supervisor, 16805 Road 19½, Platteville Co, 80651. After the above documents are reviewed, PSC will develop inspection and witness hold points and submit them to the Seller.

4.2 The Seller's inspection program shall allow for the necessary personnel and procedures to inspect, test, and document his manufacturing process, product inspections and examinations required by applicable codes and specification.

4.3 The Seller shall ensure that the requirements of this Specification and all other related documents are a part of any order purchased from a sub vendor. Copies of these documents are required to be at the locations where any work, fabrication, or processes are being performed.

4.4 The Buyer and Buyer's designated engineering agent shall have free access to the Seller's plant at all times to witness or verify, or to observe any processes, procedures, inspections or tests required by this specification. These representatives shall have the right to any information regarding engineering procurement, scheduling and production. The Seller shall provide whatever personnel, facilities, test equipment tools, or instruments as necessary to facilitate any inspection or survey. The purpose of these inspection surveys is to assure that nonconforming Material/Equipment is not shipped to the job site. The Inspection/Surveys do not relieve the Seller of his obligation to conduct an adequate inspection of his own, nor does it relieve the Seller of his obligations regarding nonconforming Material/Equipment missed by such inspections.

4.5 The Seller shall state, as part of his Quality Assurance Plan, his intended Test/Inspection points and the procedures he will conduct these Test/Inspections to. The Seller shall also state any restrictions or deviations he intends to make in the conductance of the test. Items requiring In-Process Test/Inspection shall be subject to the approval of the Buyer or the Buyer's designated engineering agent. Notification for these Test/Inspections shall be given to the Buyer at least five (5) days prior to the Test/Inspection. All equipment shall be inspected and code stamped by the authorized agency where so required. Supplementary inspection will be conducted by the Buyer. Inspection or audit by any agency of the Buyer in no way relieves the Seller of his responsibilities to provide Equipment/Material 100 percent in compliance with the specification.

4.6 At time of final shipment, all documentation required by the applicable codes and standards, and that are specifically required by this order, shall be available for review at the request of the Buyer or Buyer's designated engineering agent.

4.7 Items such as model number, material specification, etc., which have been approved by the Buyer or the Buyer's designated engineering agent may not be substituted without the prior approval of the Buyer or the Buyer's designated engineering agent.

4.8 Inspectors or testers and evaluators of tests shall be qualified to the requirements of ANS~~I~~ N45.2.6 prior to performing required testing and evaluations.

4.9 Quality Assurance Records shall be arranged in an orderly fashion, indexed, and supplied as part of the final shipment.

STANDARD PACKAGING, SHIPPING, RECEIVING, STORAGE AND HANDLING REQUIREMENTS

GENERAL - The items in this purchase order are assigned physical protection classification levels for packaging, shipping, receiving, storage and handling in accordance with the guidelines of ANSI N45.2.2-1972. The designated levels are noted as follows: ANSI Level 'A, 'ANSI Level 'B', ANSI LEVEL 'C, 'or ANSI Level 'D'.

CLEANING - The item(s) shall be cleaned to remove as much dirt, metal chips, slag, rust, grit, mill scale, oil or grease residue, chemical residue or other contaminents as is customarily removed under normal industry practice.

PACKAGING - The item(s) shall be wrapped or packaged in accordance with good commercial practice.

Packaging shall be designed to provide protection to prevent damage, deterioration or contamination of the material as a result of handling, shipping or storage of the material.

Each item shall be marked in accordance with applicable codes and standards for such items. The marking code used shall be identified in the shipping documents.

Items and packages shall be marked as follows:

a) Individual items or packages of identical items shall be identified with the following minimum information: Manufacturer, part number, PSC PO number and quantity.

b) External packaging shall be marked with the following minimum information: Manufacturer, destination, PSC PO number, handling instruction, weight of container and number of containers in shipment.

c) Identification and marking shall not be deleterious to the material and shall be designed to preclude loss due to handling, storage, shipping or as a result of environmental conditions.

SHIPPING - The item(s) shall be shipped utilizing a method of transportation consistent with the packaging methods employed.

RECEIVING - A receiving inspection, acceptance and control system shall be established to assure that parts or subassemblies to be used in the fabrication or assembly of these item(s) are in accordance with the specified requirements.

STORAGE - Storage procedures shall be established by the seller to provide protective measures which will prevent damage, deterioration or contamination of the item(s) during extended storage.

HANDLING - The item(s) shall be handled in accordance with good handling practice and in a manner so as not to degrade the item in any way.

ATTACHMENT 4

Drop Test Surveillance

SR 5.1.2c Adequacy Question

Overview of Hopper Surveillance SR 5.1.2c

Existing Surveillance Requirement:

An off-line functional test of a reserve shutdown assembly shall be performed in the hot service facility, or other suitable facility, following each of the first five refueling cycles and at two refueling cycles thereafter. These tests will consist of pressurizing the reserve shutdown hopper to the point of rupturing the disc and releasing the absorber material. If a reserve shutdown hopper rupture disk does not rupture at a differential pressure less than 300 psi and release the absorber material, the reactor shall be placed in a shutdown condition until it can be shown that LCO 4.1.6 can be met.

New Surveillance Requirement:

Testing of two reserve shutdown assemblies, one containing 20 wt% boronated material and one containing 40 wt% boronated material, shall be performed on assemblies removed during each refueling outage up to the end of plant life. The reserve shutdown system material from the tested hoppers will be visually examined for evidence of boric acid crystal formation and chemically analyzed for boron carbide and leachable boron content. In addition, these tests will consist of pressurizing the reserve shutdown hopper to the point of rupturing the disc and releasing the absorber material. If a reserve shutdown hopper rupture disk does not rupture at a differential pressure less than 300 psi and release the absorber material, the reactor shall be placed in a shutdown condition until it can be shown that LCO 4.1.6 can be met. Failure of a reserve shutdown system assembly to perform acceptably during functional testing, or evidence of extensive boric acid crystal formation will be reported to the NRC.

It has been determined that the clumping of the balls could be predicted by the formation of boric acid crystals which can be seen in the visual examination or detected in the chemical analysis. Since the existing surveillance was expanded to include a hopper test of both types of balls (20 wt% and 40 wt%) in addition to the visual and chemical examinations at each plant refueling cycle, any failure of the hopper to discharge should be prevented.

It is felt that the surveillance requirements SR 5.1.2c, pressurizing the hopper, and the new surveillance requirements are adequate to provide assurance of the operability of the reserve shutdown system.

ATTACHMENT 5

Report on Blending of

the ART Absorber Material

Manufactured in 1982 - 1983

Blending of the RSS Absorber Material

The RSS absorber material purchased from Advanced Refractory Technologies (ART) during the period 1982-1983 contained deviations from the purchase specifications regarding boron density. The boron content for the 9/16" balls varied from the required nominal specification value of .66 gm/cc lot average to the extent that a review of the boron density values by lot was conducted by G.A. Technologies at PSC's request. The review concluded that the reserve shutdown material boron density variations were inconsequential in that the variations would not result in reduced shutdown margins. It was concluded, however, that the production lots should be mixed to even out the boron density and a PSC Controlled Work Procedure was written to accomplish the task at the plant site. The mixing of the specified lots of 9/16" occurred in June, 1983 in a mixing sequence as specified by sketch 1 (SK-1) of Controlled Work Procedure (CWP)83-74. The 7/16" ART balls had but two lots (P-50R & P-51) that required mixing to even out boron density and this mixing was accomplished in August, 1983 under CWP 83-91.

Provided for your review is a more detailed chronological history and copies of the CWP's used to accomplish the work.

PSC BLENDING - 9/16 INCH DIAMETER MATERIAL

Nonconformance report (NCR) 82-89 dated 11-19-82, identified deviations from specification in the areas of boron content (5 lots), iron content (7 lots), impurity content (3 lots), and sintering temperature (2 lots). Samples from the material identified in the NCR were examined by GA Technologies and found acceptable provided the production lots were mixed (GP-1709 dated 11-24-82). GS-AR-401, dated 12-8-82, was initiated by the procuring engineer to develop a Controlled Work Procedure (CWP) which would intermix 9/16 inch lots to produce a homogeneous mixture. GS-AR-428, dated 3-24-83, was written assigning project responsibility to Site Engineering and requesting the CWP be completed as soon as possible. CWP-83-74, dated 6-3-83, was written and is attached for reference as to the method used for the blending process. Approximately 1000 pounds of 9/16 inch diameter balls were purchased under P.O. N-3554, shipment A. This allows for the formation of 12 80-pound composite lots with approximately 40 pounds of material remaining. This remainder was identified as Lot #P-15 and was scrapped per D.C. #75285. (Lot #P-15 had been previously identified as having the maximum deviation from specification values in NCR-82-89.) Sketch 1 of CWP-83-74 identifies the composition of each of these 12 80-pound composite lots. CWP-83-74 was completed on 6-8-83 with Quality Assurance sign-off on 6-19-83.

PUBLIC SERVICE COMPANY OF COLORADO

CONTROLLED WORK PROCEDURE

CWP NO. 83-74 ISSUE 1
CN NO. H. Fisher Memo

WORK BY PSC MAINT

DESCRIPTION Mix THE Boron Balls IN THE Reciving Warehouse

X SAFETY RELATED ☐ NON-SAFETY RELATED

PREPARED BY [signature] (SIGNATURE) 6-2-83 (DATE) REVIEWED BY QA/QC Paul M. Burck (SIGNATURE) 6-3-83 (DATE)

LIST REQ'D. TCR'S ✓ NA — OPERATION IMPACT REVIEWED ✓ YES ☐ NO

LIST REQ'D. SCR'S ✓ NA — TRAINING REQUIRED ✓ YES ☐ NO

FT TEST REQUIRED ☐ YES ✓ NO — CCT TEST REQUIRED ☐ YES X NO

TYPE OF H.P. REQ'D. N.A.

W.J. Franek (SIGNATURE) 6/3/83 (DATE) J Gramling (NEG - SIGNATURE) 6/3/83 (DATE) RAHeller (TECH. SERV. - SIGNATURE) 6-3-83 (DATE) [signature] (HP - SIGNATURE) 6-3-83 (DATE)

LIST OPER. PROC. REQ. REVISION X NA

OJ Clayton (SIGNATURE) 6-3-83 (DATE)

REQ'D. REPORTS: 50.59 ☐ YES X NO TECH. SPEC. ☐ YES X NO — OJ Clayton (SIGNATURE) 6-3-83 (DATE)

NPRDS UPDATE REQ'D. ☐ YES X NO — OJ Clayton (SIGNATURE) 6-3-83 (DATE)

SAFETY EVALUATION CONCURRENCE ✓ YES ☐ NO — RAHeller (SIGNATURE) 6-3-83 (DATE)

APPROVAL — TO BE COMPLETED AS NOTED

PORC REVIEW: SAF. SIG. ☐ YES X NO — UNREVIEWED SAF. QUES. ☐ YES X NO

APPROVED X YES ☐ NO Ward Craine — PORC 519 (MTG NO.) JUN 3 - 1983 (DATE)

NFSC CONCUR: SAF. SIG. ☐ YES ☐ NO — UNREVIEWED SAF. QUES. ☐ YES ☐ NO

APPROVED ☐ YES ☐ NO ______ (MTG NO.) ______ (DATE)

APPROVED ✓ YES ☐ NO SUPT. / SIGN. W. J. Franek 6/3/83

SCHEDULING: PLANT [initials] 6-4-83 (INITIAL AND DATE) OUTSIDE CONTRACTOR N/A (INITIAL AND DATE)

WORK CONTROL

(1) SYSTEM TAGGED AND RELEASED FOR WORK — NO TAGS REQUIRED — [signature] (SHIFT SUPV. - SIGNATURE) 6/6/83 (DATE)

(1A) CONSTRUCTION COMPLETE — N/A (DATE)

(2) WORK SURRENDERED FOR TEST — N/A (WORK COORDINATOR - SIGNATURE) (DATE)

(3) TAGS REHUNG FOR REWORK ☐ NA — N/A (SHIFT SUPV. - SIGNATURE) (DATE)

(4) WORK SURRENDERED FOR RETEST ☐ NA — N/A (WORK COORDINATOR - SIGNATURE) (DATE)

(5) WORK / TEST / INSP. COMPLETE — Dopke (WORK COORDINATOR - SIGNATURE) 6-8-83 (DATE)

(6) CCT COMPLETE ☐ NA Q. A. DEPT. REVIEW [initials] 6/14/83 — N/A (SHIFT SUPV. - SIGNATURE) (DATE)

(7) FT COMPLETE ☐ NA — N/A (SHIFT SUPV. - SIGNATURE) (DATE)

(8) TCR'S / SCR'S COMPLETE ☐ NA — N/A (SHIFT SUPV. - SIGNATURE) (DATE)

(9) OPER. PROC(S) REVISED AND ISSUED ✓ NA — (PLANT CLERICAL - SIGNATURE) (DATE)

(10) TAGS REMOVED AND EQUIP. RETURNED TO SERVICE — [signature] (SHIFT SUPV. - SIGNATURE) 6-8-83 (DATE)

FORM (A) 372-30-3378

DC# 75385

CONTROLLED WORK PROCEDURE
SUMMARY

CWP 83-74
CN# M. Fisher Memo
SYS# N/A

(Sheet 1 of 2)

DESCRIPTION OF WORK AUTHORIZED:

Mix The Boron Balls in the Receiving Warehouse

DEFINED WORK BOUNDARIES:

As per the Clearance Issued At The Discretion of Operations Dept

HOUSEKEEPING ZONE DESIGNATION (S):

IV

BASIS AND/OR REASONS FOR WORK

To Allow Boron Balls to be Mixed

MARKED-UP OR REVISED P & I DIAGRAMS OR ELECTRICAL SCHEMATICS:

N/A

DESCRIPTION OF INSPECTION AND/OR TESTING REQUIRED:

See PIT Record

Special Equipment #/ Calibration Dates ______ ______

CONTROLLED WORK PROCEDURE
SUMMARY

CAP 8.3-74
CI# M. FISHER MEMO
SYS# N/A

(Sheet 2 of 2)

COLD CHECKOUT AND/OR FUNCTIONAL TEST REQUIREMENTS:

N/A

"MATERIALS USED" DOCUMENTATION, i.e., Direct Charge Requisitions, Purchase Orders:

N/A

INSTALLATION DRAWINGS:

N/A

ABNORMAL/SPECIAL CONDITIONS: None Specified

EQUIPMENT NO (S) N/A

LINE NO (S) N/A

WORK LOCATION: Fuel Storage Building

PROCESS/INSPECTION/TESTING RECORD

WORK PERFORMED BY: Maintenance

JOB TITLE: Fuel Handling Criteria Boron Balls.

PREPARED BY: EWJ 6-2-83

REVIEWED BY: [illegible] 6-3-83

CN NO. R. Fisher memo

CWP NO. 83-74

CK.LST.NO. 83-74

PAGE 1 OF 3

SUGGESTED SEQUENCE *	ACTIVITY DESCRIPTION	RESPONSIBLE ORG.	INSP. BY TYPE **	COMP. BY DATE	QA/QC INSP. BY DATE
1	Obtain several cardboard barrels. These should have tight fitting lids. A barrel with ridges inside, not smooth sides, would be best if it can be rigged up or found.	PSC		Dopke 6-8-83	
2	Insure that the barrels are clean; blow with air to clean if necessary.	PSC		Dopke 6-8-83	
3	Weigh out each amount of balls from the production lots, as defined on SKETCH 1, to form a composite lot of 80 pounds. Put each weight from the production lots in separate containers not the mixing barrel. Record the exact weights.	PSC	SNCC WI	Dopke 6-8-83	ADG 6/8/83
4	Pour all the containers contents into the mixing barrel all at the same time to aid in mixing. If it is not possible to add the contents at the same time add a little of each container creating layers. Continue adding until all the balls are in the mixing barrel. The mixing barrel should not be more than ½ - 2/3 full.	PSC		Dopke 6-8-83	

* It is [illegible] necessary to list or perform activities between hold points in an exact order.

** Hold point inspection must be performed prior to proceeding to following sequence steps.

WI = Witness

PROCESS/INSPECTION/TESTING RECORD

WORK PERFORMED BY: Maintenance

JOB TITLE: Fuel Handling Criteria Koron Balls

PREPARED BY: KWJ 6-2-83

REVIEWED BY: [illegible] 6/3/83

CN NO. M. Fisher Memo

CWP NO. 83-74

CK.LST.NO. 83-74

PAGE 2 OF 3

SUGGESTED SEQUENCE *	ACTIVITY DESCRIPTION	RESPONSIBLE ORG.	INSP. BY TYPE **	COMP. BY DATE	QA/QC INSP. BY DATE
5	Stir the mixture with a smooth wooden stick such as a broom handle. Do not damage the balls. If stirring appears to cause damage skip this step.	PSC		Depke 6-8-83	
6	Secure the lid on the barrel.	PSC		Depke 6-8-83	
7	Roll the barrel to create mixing action. The barrel should be rolled for 10-15 minutes. Rolling can be done either manually or if an electric roller can be found it can be used.	PSC	QA/QC WP	Depke 6-8-83	ADK 6/8/83
9	If it is visibly apparent that mixing is not occurring or if the balls are not thoroughly mixed contact Engineering.	PSC	N/A	Depke 6-8-83	
1	When mixing is completed put the balls in a thick plastic bag. Mark the bag with the composite lot number and insert a copy of the weight record of each production lot	PSC	QA/QC WP	Depke 6-8-83	ADK 6/8/83

* It is not necessary to list or perform activities between hold points in an exact order.

** Hold point inspection must be performed prior to proceeding to following sequence steps.

[illegible]

PROCESS/INSPECTION/TESTING RECORD

WORK PERFORMED BY: Maintenance

JOB TITLE: Fuel Handling Criteria Boron Balls.

PREPARED BY: KWS 6-2-83

REVIEWED BY: [illegible] 6/3/83

CN NO. M. Fisher memo

CWP NO. 83-74

CK. LST. NO. 83-74

PAGE 2 OF 3

SUGGESTED SEQUENCE ⁂	ACTIVITY DESCRIPTION	RESPONSIBLE ORG.	INSP. BY TYPE ⁂⁂	COMP. BY DATE	QA/QC INSP. BY DATE
9	used. Seal the bag which can then be stored in barrels stacked on top of one another. Two 80 pound composite lots per barrel for storage.	PSC	—	Depke 6-8-83	—
	Note: The production lot weights used to make a composite lot should be kept as close as possible to the amounts indicated on SK-1 but the weights can change as necessary. A list is attached which correlates between production lot number (Powder number) and drum number.	PSC	—	Depke 6-8-83	—
10	Attach the recorded weights of each composite lots to the CWP for our records.	PSC	[illegible] WP	Depke 6-8-83	[illegible] ⁎ 6/8/83
	Note: Lot P15 is not used and should be disposed of by trashing it.		—	—	—
	Note: When all 12 lots are mixed transfer them to the warehouse for permanent storage (via the receiving warehouse if necessary.		—	—	—

⁂ It is . necessary to list or perform activities between hold points in an exact order.

⁂⁂ Hold point inspection must be performed prior to proceeding to following sequence steps.

WP = Witness [illegible]

⁎ LEFT OVER B[illegible]S (21.5 LBS TOTAL) WERE DI[illegible] POSED OF

Composite Lots

Lot 1	Lot 2	Lot 3	Lot 4
30# P2 ✓	18# P2 ✓	30# P7 ✓	10.2# P7 ✓
20# P6 ✓	20# P6 ✓	30# P10 ✓	19.8# P10 ✓
20# P9 ✓	12# P7 ✓	20# P20R ✓	20# P11 ✓
10# P11 ✓	30# P9 ✓		30# P20R ✓

Lot 5	Lot 6	Lot 7	Lot 8
20# P8 ✓	30# P8 ✓	20# P12 ✓	30# P13 ✓
30# P11 ✓	30# P12 ✓	11# P13 ✓	10# P14 ✓
30# P21 ✓	20# P21 ✓	9# P14 ✓	30# P19R ✓
		7# P16 ✓	10# P23 ✓
		23# P19R ✓	
		6.5# P20R ✓	
		3.5# P21 ✓	

Lot 9	Lot 10	Lot 11	Lot 12
10# P13 ✓	7# P13 ✓	20# P16 ✓	3# P7 ✓
20# P14 ✓	25# P14 ✓	30# P18R ✓	7# P8 ✓
20# P17 ✓	10# P16 ✓	30# P22 ✓	4# P9 ✓
30# P23 ✓	23# P17 ✓		6# P12 ✓
	15# P23 ✓		20# P16 ✓
			13# P18R ✓
			27# P22 ✓

Production Lots Used.

P2	P11	P18R
P6	P12	P19R
P7	P13	P20R
P8	P14	P21
P9	P16	P22
P10	P17	P23

Production Lot P15 was not used.

Record actual weights used to form the composite lots one copy should be kept with the composite lot and the original filed with the CWA

RECEIVED NOV 2 9 1982

GA Technologies Inc.
P.O. BOX 81608
SAN DIEGO, CALIFORNIA 92138
(619) 455-3000

November 24, 1982
GP-1709
FSV-12-D-1

Mr. H. L. Brey, Manager
Nuclear Engineering Division
Public Service Company of Colorado
5909 East 38th Avenue
Denver, CO 80207

Subject: Transmittal of a Justification for Accepting the 40% Boron Reserve Shutdown Material

Reference: PSC P.O. N-3980

Dear Mr. Brey:

As requested by Jack Levin, GA has reviewed the deviations from specification for the reserve shutdown material manufactured by Eagle Pitcher Co. for PSC. Enclosed are two GA internal memoranda describing the results of this review. Memoranda CNE:VM:125:82 addresses excess boron density. Memoranda RDB:005:CM:82 addresses the excess impurity levels.

These reviews conclude that the reserve shutdown material is acceptable. The deviations from the specified boron density are inconsequencial. They will not result in reduced shutdown margins. It is concluded, however, that these production lots should be mixed to even out the boron density in any one hopper.

Should you have any questions, please contact Gary Hein at (619) 455-2645.

Very truly yours,

William A. Graul

William A. Graul, Manager
Fort St. Vrain Project

Enclosures

Holms to follow

PSC BLENDING 7/16 INCH DIAMETER MATERIAL

Nonconformance Report (NCR) 83-144, dated 7- 6-83, identified Lot P-50R as being slightly low and Lot P-51 as being slightly high for boron content. Lot P-50R consists of 10 pounds. Lot P-51 consists of 24 pounds of balls. The disposition of NCR 83-144 requires that lots P-50R and P-51 be mixed prior to use and their containers appropriately marked. Controlled Work Procedure (CWP) 83-91 was prepared on 7-25-83 to mix equal amounts of lots P-50R and P-51 and is attached for reference. As equal amounts were required for mixing, this resulted in an excess of 14 pounds from Lot P-51. The remaining 14 pounds from Lot P-51 was scrapped per D.C. #27745. CWP 83-91 was completed on 8-22-83 with final Quality Assurance sign-off on 8-31-83.

83-91

R. FISHER MEMO

PSC MAINT.

Mix the Boron Balls

Paul M. Burk 7-26-83

7-25-83

W. J. Granek 8/8/83 J. Gramling 7/27/83 RAHeller 8-10-83

O. J. Clayton 8-9-83

O. J. Clayton 8-9-83

8-9-83

SAFETY EVALUATION CONCURRENCE YES NO

RAHeller 8-16-83

PORC REVIEW

UNREVIEWED SAF. QUES. YES NO

W. J. Granek

PORC 529 AUG 11 1983

UNREVIEWED SAF. QUES. YES NO

William J. Granek 8/8/83

8-15-83

N/A

H. O'Hagan 8-18-83

SLB 8-22-83

N/A

N/A

N/A

SLB 8-22-83

H. O'Hagan 8-22-83

H. O'Hagan 8-22-83

H. O'Hagan 8-22-83

H. O'Hagan 8-22-83

Q. A. SECT. REVIEW

CONTROLLED WORK PROCEDURE
SUMMARY

CWP 83-91
CN# M. Fisher Memo
SYS# N/A

(Sheet 1 of 2)

DESCRIPTION OF WORK AUTHORIZED:

Mix The Boron Balls in The Receiving Warehouse

DEFINED WORK BOUNDARIES:

As per the Clearance Issued At The Discretion of Operations Dept

HOUSEKEEPING ZONE DESIGNATION (S):

IV

BASIS AND/OR REASONS FOR WORK

To Allow Boron Balls to Be Mixed

MARKED-UP OR REVISED P & I DIAGRAMS OR ELECTRICAL SCHEMATICS:

N/A

DESCRIPTION OF INSPECTION AND/OR TESTING REQUIRED:

See PIT Record

Special Equipment #/ Calibration Dates

CONTROLLED WORK PROCEDURE
SUMMARY

CAP 83-91
CN# M. Fisher Memo
SYS# N/A

(Sheet 2 of 2)

COLD CHECKOUT AND/OR FUNCTIONAL TEST REQUIREMENTS:

N/A

"MATERIALS USED" DOCUMENTATION, i.e., Direct Charge Requisitions, Purchase Orders:

N/A

INSTALLATION DRAWINGS:

N/A

ABNORMAL/SPECIAL CONDITIONS: None Specified

EQUIPMENT NO (S) N/A

LINE NO (S) N/A

WORK LOCATION: Receiving Warehouse

PROCESS/INSPECTION/TESTING RECORD

WORK PERFORMED BY: Maintenance

JOB TITLE: Fuel Handling Criteria Boron Balls

Work to be performed in receiving warehouse

PREPARED BY: KWJ 7-25-83

REVIEWED BY: [illegible] 7/26/83

CH NO. [illegible]

CMP NO. [illegible]

CK.LST.NO. [illegible]

PAGE 1 OF [illegible]

SUGGESTED SEQUENCE #	ACTIVITY DESCRIPTION	RESPONSIBLE ORG.	INSP. BY / TYPE XX	COMP. BY / DATE	[illegible]
1	Obtain cardboard barrels with tight fitting lids. A barrel with ridges inside is preferable.	Psc		S 8-19-83	
2	Insure that the barrels are clean; blow with air to clean if necessary.	Psc		S 8-19-83	
3	Weigh out equal amounts of P50R and P51. The barrel should not be more than ½ - ⅔ full. Record the exact weights	Psc	[illegible] WP	S 8-19-83	[illegible]
4	Pour both lots, P51 and P50R at the same time into the mixing barrel. If this is not possible pour in a little of each lot creating layers.	Psc		S 8-19-83	
5	Stir the mixture with a smooth wooden stick. Do not damage the balls	Psc		S 8-19-83	

\# It is not necessary to list or perform activities between hold points in an exact order.

PROCESS/INSPECTION/TESTING RECORD

PREPARED BY: [illegible] 7-25-83
REVIEWED BY: [illegible] 7/26/83

CH NO. [illegible]
CMP NO. E3-[illegible]
CK.LST.NO. [illegible]
PAGE 2 OF [illegible]

WORK PERFORMED BY: Maintenance

JOB TITLE: Fuel Handling Criteria Boron Balls

Work to be performed in the recieving warehouse

SUGGESTED SEQUENCE #	ACTIVITY DESCRIPTION	RESPONSIBLE ORG.	INSP. BY TYPE XX	COMP. BY DATE	QA/QC INSP. DATE
6	Secure the lid on the barrel	P&C		S 8-19-83	
7	Roll the barrel to create mixing action. The barrel should be rolled for 10 or 15 minutes	P&C	QA/QC WP	S 8-19-83	[illegible]
8	If it is visible apparent that mixing is not occurring, or if the balls are not throughly mixed, contact Engineering.	P&C		S 8-19-83	
9	When mixing is complete put the balls in a thick plastic bag. Mark the bag as a mixture of lots P50R and P51, the Co. number and a copy of the recorded weight. Seal the bag which can then be stored in barrels stacked one on top of another.	P&C	QA/QC WP	S 8-19-83	[illegible]
10	Attach the weight records to the CWP. Transfer to the warehouse for permanent storage via the recieving warehouse.	P&C	QA/QC WP	S 8-19-83	[illegible]

It is not necessary to list or perform activities between hold points in an exact order.
[illegible] must be performed prior to proceeding to following sequence steps.