SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT 21 TO FACILITY
OPERATING LICENSE NO. DPR-34

OF

PUBLIC SERVICE COMPANY OF COLORADO
FORT ST. VRAIN NUCLEAR GENERATING STATION
DOCKET NO. 50-267

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1.0 INTRODUCTION

Fort St. Vrain, a 330-MWe high temperature gas cooled reactor (HTGR), was designed by the General Atomic Company and is being operated by the Public Service Company of Colorado (PSCo) near Platteville, Colorado. On October 28, 1977, the Nuclear Regulatory Commission (NRC) authorized operation of the reactor up to 70 percent of rated thermal power. All of the power ascension tests have been completed up to 70% of thermal power.

The first refueling at Fort St. Vrain has been completed and the insertion of eight test fuel elements and PGX graphite corrosion surveillance specimens has been finished in accordance with requirements delineated in Amendment 20.

This amendment deals with various design modifications that Public Service Company of Colorado will perform prior to start up and with several changes to the Technical Specifications it has requested; these include:

(1) Modifications to the fire protection system for the three room complex, the Auxiliary Electric Room, the 480 Volt Switchgear Room and the congested cable areas; this constitutes Stage III fire protection implementation;

(2) Conversion of the Interim Alternate Cooling method to the final Alternate Cooling Method;

(3) Installation of two modified control rod drive and orificing assemblies which contain instrumentation to further study and evaluate the power, flux, and temperature fluctuations that were experienced; Technical Specifications changes are not required by this installation.

(4) Testing of the reactor building louver system on a quarterly basis;

(5) Elimination of manual isolation of the high pressure helium supply fro the helium circulator buffer supply header; and

(6) Addition of two firewater booster pumps to the firewater system to provide augmented capacity to operate a circulator water turbine and to supply feedwater for safe shutdown cooling.

Technical Specification changes dealing with the above are identified in later chapters of this report along with a discussion of the corresponding safety significance of each change.

The Fort St. Vrain reactor is described in the Final Safety Analysis Report (FSAR) submitted for our review in November 1969. The FSAR, as amended, formed a basis for our January 20, 1972 safety evaluation report and a first supplement, which was issued on June 12, 1973. The operating license, DPR-34 was issued on December 21, 1973. The operating license has been amended twenty one times, including the amendment supported by this safety evaluation. The FSAR and other early documentation continues to support our safety reviews, as augmented by the additional information and the operational reports referenced herein.

The reactor achieved criticality on January 31, 1974, and low power physics testing was initiated. These low power tests, identified as the "A Series" tests, along with the "B Series," or power ascension, tests were reported in accordance with Section 7 of the Technical Specifications.

Also, in accordance with the Technical Specifications, Public Service of Colorado provides "Reportable Event" reports and "Unusual Event" reports on safety items relating to abnormal, unusual or unanticipated events that occur during the course of plant operations. In addition to the reports received from the licensee, our safety reviews have benefitted from information on plant status and operations provided by the Office of Inspection and Enforcement, and by visits to the plant site by technical specialists to review plant records and the "as-built" condition of the plant. Our safety review has also included consideration of comparable light water reactor experience and policies, information developed on gas cooled reactor safety under the sponsorship of the Office of Nuclear Regulatory Research, and information developed during the review of the General Atomic Standard Safety Analysis Report, GASSAR.

2.0 STAGE III FIRE PROTECTION

Background

In April 1975, a routine NRC inspection of Fort St. Vrain revealed that some fire stops in the electrical cable system had not been installed, and that the routing of some cables deviated from the installation criteria set forth in the FSAR. An audit, initiated by PSCo in the Spring of 1975, included a physical audit of the electrical cable system to establish routing of cables and determined that there were a significant number of deviations in the installed electrical system from FSAR requirements.

The initial discovery of deficiencies in electrical cable routine compared to the criteria in the FSAR came shortly after the occurrence of an electrical cable fire at the Tennessee Valley Authority Browns Ferry Nuclear Plant near

Athens, Alabama. As information on this event became available, it became evident to the PSCo and the NRC staff that the scope of consideration should be broadened beyond the topic of electrical cable segregation and separation deficiencies, and the corrective action required to bring these deficient items into compliance with the FSAR criteria.

The scope of activities was therefore broadened to include consideration of fire prevention, detection, and suppression, and alternate methods for accomplishing orderly plant shutdown and cooldown in case of loss of normal and preferred alternative shutdown and cooldown cable systems for any reason, e.g., a fire. These points had been raised in NRC Office of Inspection and Enforcement Bulletins 75-04 and 75-04A following the fire at Browns Ferry. Discussions with PSCo also included consideration of alternative methods for cooling the reactor which would be capable of operation independent of the occurrence of disruptive faults or events in the congested cable areas; this eventually led to the design of such provisions, and a commitment by PSCo to install the equipment to perform this function.

The items associated with alternative methods of shutdown and cooldown using installed equipment, the installation of equipment for cooldown independent of events in the congested cable areas, and some aspects of the provisions for fire protection, which have been committed to by PSCo, are plant upgrading actions rather than actions taken to correct deviations from the specific provisions of the FSAR or license. The upgrading actions were considered at the same time as the electrical cable corrective actions because of the interrelationships of equipment, equipment locations, and safety requirements.

The discussions held between PSCo and NRC led to an overall approach for corrective and upgrading actions. The PSCo proposal entails the following elements:

(1) A number of specific electrical modifications to correct deficiencies;

(2) Improved fire protection, detection and suppression, including the application to certain cables of a fire retardant material (Flamemastic);

(3) An alternate cooling method to ensure that, even in the event of a major electrical fault or fire in the congested cable area, means would be available to depressurize the reactor and provide continued cooling to the reactor; and

(4) Procedures to cover degraded cooling conditions.

Because of procurement and installation schedules projected to extend beyond the time when the plant was expected to be ready to resume rise-to-power test operation, PSCo proposed that the actions outlined be implemented in three stages to allow the plant to resume rise-to-power test operation at an earlier time than otherwise possible. Three stages of reactor operation have also been proposed corresponding to the implementation schedule. A summary of the program proposed by PSCo follows:

<u>Stage 1:</u> To resume critical operation and operate through a 22.5-day operational test sequence including a two-day run at about, but not exceeding, 40 percent of rated reactor power:

a. Complete all modifications to the electrical installation to correct the deficiencies identified and provide assurance that redundant equipment will remain operable in the event of

a major electrical fault or fire in the cable system. Action to be taken includes:

(1) Re-routing of a number of electrical cables (including all essential cables) in accordance with the FSAR criteria;

(2) Rearranging of load assignments to essential electrical busses;

(3) Evaluation and appropriate derating of cables having fire retardant applied or asbestos cloth installed;

(4) Correction of any overcurrent protection deficiencies;

(5) Modification of the Class IE power source for the plant protection system equipment supplied by 120v a-c Bus 3;

(6) Correction, as needed, of cable tray loading to assure that tray fill does not exceed FSAR criteria requirements; and

(7) Review electrical cables to ensure that conductor sizes and insulation ratings meet the objectives of the FSAR electrical design criteria.

b. Provide fire protection as follows:

(1) Around the clock roving watch will augment the plant staff. Specific instructions will be issued covering his duties.

(2) Auxiliary Electrical Equipment and 480 Volt Switchgear Rooms and cable areas adjacent to the "G" and "J" walls will be equipped with temporary self-contained fire detection units with local audible alarms;

(3) The number of portable air packs will be increased from six to 16;

(4) Specific procedures will be issued for extinguishing an electrical fire using the hose water system;

(5) Surveillance requirements for air breathing equipment will be issued; and

(6) Personnel will receive training in these measures.

NOTE: As a result of our review (see Amendment 14), we required that additional fire water pumping capability, namely that two diesel-driven pumps be used in connection with the IACM, will be available during Stage 1 operation. Also as a result of our review, we required that the procedures discussed under item a of Stage 2 below be available for Stage 1 operation.

Stage 2: To operate above 40 percent and up to 100 percent of rated reactor power, but no longer than the first refueling, the following corrective action would be implemented:

a. Implement the procedures developed as a result of a study of core cooling under degraded plant conditions. This includes appropriate operator training.

b. Provide an interim version of the ACM (to be known as the IACM) for Stage 2. The IACM will be operable prior to proceeding beyond Stage 1, and until the permanent ACM is implemented. A major difference between the ACM and the IACM is that, in the IACM, firewater will be provided by one of two temporarily

installed diesel engine-driven pumps. The second pump will be used to supply an alternate source of water for PCRV cooling or for the purpose of fire fighting. These pumps will take suction from the Circulating Water Cooling Tower Basin and pump the water through short fire hose connections into the permanently installed fire water system and then into the PCRV Liner Cooling Water System via existing piping cross connects. The return water is then directed back to the Circulating Water Cooling Tower for cooling and recirculating through permanently installed plant piping. Provisions for both the ACM and IACM include depressurization, release of reserve shutdown material, detailed procedures, training and surveillance.

c. Fire protection and detection improvements to be provided for Stage 2 are listed below:

   (1) Continue to employ temporary fire detectors and firewatch described under Stage 1;

   (2) Continue in effect the procedures and instructions for use of water systems on electrical fires;

   (3) Install a manually actuated Halon fire suppression system in the Control Room, Auxiliary Electrical Equipment Room, and 480 Volt Switchgear Room. The first two rooms will be activated simultaneously.

Install three-zone Halon sampling lines in each room. Provide external sampling station adjacent to each manual Halon control valve. Provide portable three-channel Halon detector.

Provide hand held Halon detectors for personnel entering fire spaces;

(4) Install ventilation dampers in Control Room, Auxiliary Electrical Equipment Room and 480 Volt Switchgear Room. Provide remote actuation damper controls adjacent to manual Halon control stations;

(5) Install continuous air breathing hose equipment system in Control Room, and 480 Volt Switchgear Room;

(6) Expand in-plant communication capability to include an alternate communication system to assure communications during a fire emergency;

(7) Apply a fire retardant coating to cables in the Auxiliary Equipment Room, the 480 Volt Switchgear Room, Reactor Building "J" and "G" walls, Turbine Building side of the "G" wall and underground cableways;

(8) Issue procedures for manual Halon activation including ventilation system operation;

(9) Institute personnel training for Stage 2 fire protection measures; and

(10) Issue surveillance requirements for Halon system.

The Stage 1 review of the Fort St. Vrain fire protection program was evaluated in Amendment 14 dated June 18, 1976. Stage 2 review was evaluated in Amendment 18 dated October 28, 1977. This amendment deals with the Stage 3 fire protection provisions along with the conversion of the IACM to the ACM.

Stage 3: Prior to reactor start-up following the first refueling outage the following corrective action would be implemented:

(1) Install a permanent central fire detection and annunciation system, the fire detection system is to automatically activate Halon in the Auxiliary Electrical Equipment Room and 480 Volt Switchgear Room;

(2) Install fixed water spray systems in the Auxiliary Electrical Equipment Room, 480 Volt Switchgear Room, and in cable areas adjacent to "J" and "G" walls;

(3) Install drip shields over electrical cabinets in the above areas;

(4) Modify the fire protection water system as required to support additional spray water requirements;

(5) Provide an automatic closing feature for the Auxiliary Electrical Equipment Room and 480 Volt Switchgear Room dampers;

(6) Modify the common room feature of the Control Room and Auxiliary Electrical Equipment Room ventilation system;

(7) Close the Control Room floor openings;

(8) Issue operating procedures for the fixed water spray extinguishing systems and automatic Halon operation feature;

(9) Institute personnel training for use of fixed spray systems;

(10) Issue surveillance requirements for fire detection system and fire protection water system; and

(11) Establish a plant fire brigade and a brigade training program.

Conclusions

These modifications have been reviewed as part of the Fort St. Vrain Stage 3 fire protection program and have been found to follow the applicable guidelines of Appendix A to BTP 9.5-1.

We have also reviewed the plant locations in which the ACM along with its associated cable routing is located using the acceptance criteria that no single fire should be able to simultaneously result in failure of the ACM and the primary systems for cooling down the plant. From our review, we have concluded that the cable routing through an independent and separate duct bank between the new diesel generator power supply, and the equipment necessary to cool down the plant, satisfies the above requirement and is, therefore, acceptable. This review terminates the three stages of the fire protection program implemented at Fort St. Vrain as outlined above and authorizes the continuation of the rise to power program except as modified by other issues, viz., fluctuations, accident analyses, etc., discussed in previous amendments.

It should be noted that one additional fire protection item is still pending, viz., revision of the existing plant fire protection Technical Specifications to apply to other safety-related plant areas consistent with the requirements of the Standard Technical Specifications dealing with fire protection. This item will be addressed in a later amendment and be implemented before operation of the Fort St. Vrain reactor at 100 percent power.

3.0 ALTERNATE COOLING METHOD

Background

As a result of discussions with the NRC staff concerning the cable separation and segregation problem and considerations regarding additional fire detection prevention and suppression measures, Public Service Company of Colorado has agreed to provide an alternate means of providing continued reactor cooling in the event of the occurrence of disruptive faults or events, such as a major fire, in the congested cable areas. This alternative cooling method (ACM) will ensure that conditions and public health and safety consequences, analyzed and presented in Design Basis Accident number 1 in the FSAR, are not exceeded in the case of such disruptive faults or events in congested cable areas.

The ACM is designed to accomplish the following functions:

(1) To maintain the reactor subcritical by means of local, manual actuation of the Reserve Shutdown System;

(2) To provide for continued cooling of the PCRV and therefore the core by manually initiating resumption of PCRV liner cooling;

(3) To allow manual, local depressurization of the PCRV through the purification system; and

(4) To establish manual local operation of the Reactor Building Exhaust system and radiation monitoring of the exhaust effluent to the atmosphere.

The following is a listing of all equipment associated with the ACM:

Diesel-driven generator (2500Kw)
Plant lighting auto-transfer switches
Electrical equipment transfer switches

4160 V to 480 V transformer
Stack effluent radiation monitor
Firewater pump
Service water pump
PCRV liner cooling water pumps
Reactor plant exhaust fan
Helium purification cooling water pump
Motor operated valves
Firewater pump house vent fans and louvers
Service water tower fan
Service water return pump
Circulating water makeup pump
Diesel oil transfer pump
Plant lighting
Reserve shutdown system

Electric power for the above equipment, about 750 hp, is supplied by the separate dedicated 2500 KW diesel engine driven generator unit in the event offsite and normal onsite emergency power supplies are lost.

The Fort St. Vrain primary coolant system contains four helium circulators, any one of which is capable of supplying adequate core cooling following a reactor shutdown. For shutdown cooling these circulators can be driven by steam from a flash tank or from the auxiliary boiler. They can also be driven by water turbine drives which can be supplied with water from any of three water systems containing a total of eleven separate pumps.

Section 14.4 of the FSAR addressed the short-term (30 minutes or less) loss-of-forced-circulation (LOFC) event. During the subsequent transient following loss of forced circulation, the average primary coolant temperatures within the core would rise to about 1600°F. Forced circulation cooling of the reactor core must be re-established within 5 hours or restart is precluded due to potential damage to the steam generator inlet ducts which pass through the core support floor. An average helium temperature of 2100°F would develop under such conditions. (Ref. FSAR, Appendix D, Section D.2.5)

A walk-through of the manual operations necessary to put the ACM into service to restore liner cooling, if it had been lost, indicates that the system can be made operational in approximately 1 to 2 hours; well within the twenty-five to thirty hour interruption analyzed as the margin available for restoring the liner cooling system to service.

The liner cooling system for the PCRV is designed to remove a total of approximately 24 million BTU per hour with both independent cooling loops operating. Section 14.10 of the FSAR established that heat removal during the LOFC accident is adequate utilizing only one loop of the system.

To maintain the PCRV concrete temperatures within acceptable limits, to assure the basic integrity of the primary containment, and to provide for reactor core cooling, the PCRV cooling system must be maintained functional. This is accomplished by using normal power supplies or if necessary the ACM power supply to run necessary equipment.

Since the reactor core must remain in a shutdown configuration throughout the LOFC event, the reserve shutdown material housed in the control rod drives would be manually released into the core by operator action.

For ACM operation, in addition to maintaining the PCRV liner cooling system in service, other plant systems will also be returned to service, as follows:

(1) Fire Protection System - to provide water for fire fighting;

(2) Service Water System - to provide cooling water, as required;

(3) Helium Purification, Liquid Nitrogen, Helium Storage System, and Purification Cooling Water System to permit depressurization of the PCRV;

(4) Circulating Water Makeup System - to provide makeup water to the Service Water and Firewater Systems;

(5) Reserve Shutdown System - to provide backup to the control rods, to assure the reactor remains subcritical;

(6) Reactor Plant Ventilation System - to permit disposal of excess PCRV helium inventory through normal discharge of the Reactor Building exhaust system and to provide for post-accident monitoring of gaseous discharges from the plant;

(7) Auxiliary Boiler Fuel System - to transfer diesel fuel from storage to the ACM diesel - generators; and

(8) Plant Lighting.

Evaluation

The initial safety evaluation of both the ACM and the Interim ACM (IACM) was presented in Amendment #14 dated June 18, 1976. This was followed by a Safety Evaluation of the IACM in Amendment #18 dated October 28, 1977.

We have reviewed the design and implementation of the IACM/ACM conversion cable routing. The requirement to be satisfied was that no single event would simultaneously result in failure of the ACM and the primary systems for cooling down the plant. We have concluded from our review that the cable routing through an independent and separate duct bank between the new diesel generator power supply and the equipment necessary to cool down the plant satisfies the above requirement and is therefore acceptable.

In addition, we have reviewed the design, installation and operational requirements of the Alternate Cooling Method and conclude that the ACM can provide all necessary functions to assure safe plant shutdown and emergency cooling under the degraded conditions caused by a large electrical fault or fire in the electrical system. Therefore, the operation of the plant under the conditions of the proposed revisions to the Technical Specifications as modified by the staff pursuant to agreement reached with PSCo, is acceptable.

4.0 INSTRUMENTED CONTROL RODS

Background

Cyclic temperature fluctuations were first noted at the Fort St. Vrain reactor on October 31, 1977 at 58% power during the initial rise in power above the previously authorized 40% level. The fluctuations were observed in outlet helium temperatures, external thermal neutron flux, steam temperatures, and PCRV movement. Temperature fluctuations have remained within design and Technical Specification limits, are non-divergent and reproducible. The fluctuations are core-wide and generally out-of-phase from one refueling region to another with a range in period from 5 to 20 minutes. The average core thermal power and average helium temperatures remain relatively constant during the fluctuations.

A total of 30 core fluctuation events, some occurring spontaneously and some induced for test purposes, have been identified at Fort St. Vrain. Following the initial fluctuation event, plant parameters were monitored throughout plant operations to enable immediate detection of fluctuations. In all cases it was demonstrated that reducing reactor power was a reliable means of stopping the fluctuations. As part of a continuing comprehensive diagnostic program to investigate and characterize the fluctuations. in-core instrumentation will be installed by modifying two control rod drive assemblies, as described in PSCo letter dated November 17, 1978.

The temporary modifications to the two control and orificing assemblies consist of removing one control rod from each assembly and adding an instrument package which occupies the space vacated by the removed control rod. Although the instrument package does occupy the space vacated by the removed rod, the package is not connected to the corresponding control rod cable; instead, the instrument package is isolated from and independent of normal operation of the remaining control rod. The in-core instrumentation location is shown in Figure 1 and consists of the following:

(1) Three thermocouples to monitor helium temperature one on each of three axial locations;

(2) A self-powered neutron detector (SPND) with a compensating cable to monitor the local/region flux level at the core midplane;

(3) A fission couple to monitor local/region flux level near the bottom of the core;

(4) A fission chamber to monitor region flux level in the bottom block of the upper reflector; and

(5) Two microphones to detect changes in turbulence (flow).

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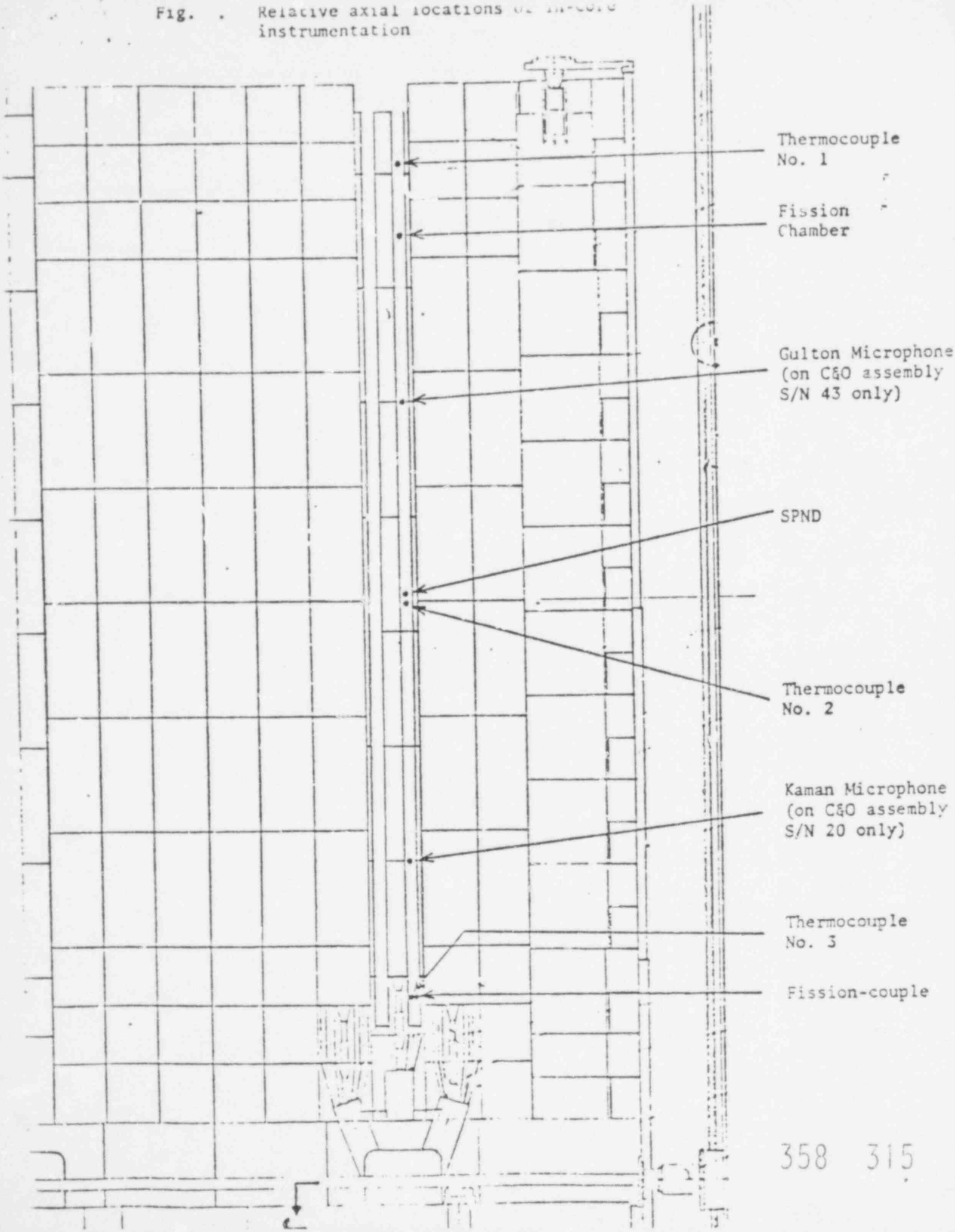
Fig. . Relative axial locations of in-core instrumentation



These instruments are securely attached at various axial locations to a support rod. The axis of the support rod is maintained at a position near the centerline of the control rod channel by means of several spiderlike assemblies at various axial locations. The basic materials of construction are magnesium oxide, stainless steel, inconel and chromel-alumel and all joints are welded.

In addition to the above noted instrumentation, the two control and orificing assemblies are provided with pressure transmitters located in the mechanism compartment. They are plumbed in a manner to measure the pressure differential across their respective orifice valves.

One of the control and orificing assemblies is provided with a linear variable differential transducer (LVDT). It is also located in the mechanism compartment and is mounted at the upper end of the orifice valve drive mechanism. This device will provide information relative to the vertical movement (and/or growth) of the region's central column over which the control and orificing assembly is mounted.

Other temporary modifications were required in order to pass the temporary in-core instrumentation leads through the primary and secondary closures. These included making two temporary ports in the primary closure piece of the control and orificing assembly, and the fabrication of temporary secondary closure plates containing four ports.

After exiting the temporary secondary closure plate, the temporary in-core instrumentation leads will pass between hold down plates through a notch provided to accommodate their passage.

Following completion of all measurements requiring the use of the instrumented control and orificing assemblies, all temporary instrumentation is to be removed (at the hot service facility) and disposed of. The drives will then be returned to their original operational conditions; i.e., the removed control rod will be reinstalled and all nonfunctional ports required for the temporary instrumentation will be sealed.

Evaluation

We have reviewed the proposed modifications with regard to their effect on those aspects of core neutronic and thermal-hydraulic behavior important to reactor safety.

Public Service of Colorado has proposed to replace one of the control rods in each of two control rod pairs by a string of detectors. The detector strings would not be attached to the control rod drive and the remaining control rods of the pairs would be free to move in accordance with the present withdrawal sequence. Modifications to the control and orificing assemblies will permit the leads from the detectors to be routed out of the core.

Four different pairs of refueling regions have been investigated as candidates for insertion of the detector strings. They will be located initially in regions 35 and 5. The effect of locating the detector strings in each refueling region on core parameters has been calculated for cycles 1 and 2. The effect on shutdown margin, power distribution, and rod withdrawal sequence have been determined. The Technical Specification requirement of a one percent shutdown margin is met throughout the remainder of Cycle 1 and all of Cycle 2 (the minimum calculated shutdown margin is 2.1 percent). Since the modified control rod pairs are not expected to meet the scram time requirements it was assumed that these rods did not scram as well as the most reactive of the remaining rods. The absence of the second rod of the control rod pair during that portion of the rod withdrawal

sequence when these rods are deep in the reactor causes a significant increase (40-50 percent when the rods are fully inserted) in the peaking factor of the affected refueling region. However, the power in these regions is low (region peaking factors less than 0.5) and the increase in peaking factor is easily accommodated by adjusting the orifice valves. The differences are well within the Technical Specification limits (LCO 4.1.3).

The effect of the removal of the control rods on the worth of other rods or banks in the control rod withdrawal sequence was investigated. It was determined that the conditions of the Technical Specifications (LCO 4.1.3) regarding maximum rod worth were met.

When operating at power the detector strings are still in the core. However, they have a negligible effect on core power distribution. The analysis of the effects of the control rod modifications have been performed using the calculations methods that were used in the final design of the core. The core startup following the modifications will be used to obtain rod bank worths, critical positions, etc., to verify the analyses.

We have reviewed the core physics aspects of the proposed control rod modifications and find them to be acceptable. This conclusion is based on the following considerations.

(1) The analysis methods are those previously used (for the FSAR analyses) and accepted.

(2) The results of the calculations show comfortable margin to operating limits.

(3) Confirmation of the rod worths and peaking factors will be performed during the first startup after the modifications.

Mechanical failure of an instrument package is analogous to the failure of a control rod. The bottom of the instrument package has been designed such that in the unlikely event of a failure, the exit to the control rod channel would not be blocked preventing the flow of coolant. Nevertheless, thermal and fuel performance analyses have been performed assuming the instrument assembly drops to the bottom of the channel and completely blocks the coolant flow while the plant is operating at full power. These analyses show that, even under these severe conditions and if the failure of the instrument assembly went unnoticed for 40 hours, the instrument package would not melt and the impact on fuel performance would be negligible. If the instrument package should fail, the failure would result in anomalous readings which would give indication of such a failure. Also, the bottom of the instrument package has been designed to act as a catcher for small parts, should an unlikely occurrence of that nature happen.

The presence of the instrument package in the control rod channel will have an insignificant effect on the normal helium flow through the channel. Calculations indicate that the flows through the channel containing the instrument package, through the corresponding channel with the control rod inserted, are within 10%. These conclusions result from a comparison of the relative flow resistances for these cases. The values of flow resistance for the instrument package and for the control rod were determined by means of full-scale flow tests.

The two principal resistances to flow through the control rod channel are at the location where coolant passes through two 5/8-inch diameter holes in the guide tube inside the plenum elements, and at the exit of the channel

where the coolant leaves the channel through a 1-inch diameter hole. Friction and other losses in the channel are small compared to these for both the regular control rods and the instrument support structure of the ICRD. The largest flow resistance occurs when the control rod is in any withdrawn position. Control rod components are then opposite the two 5/8-inch diameter holes in the guide tube and require the entering flow to turn after entering. For the inserted control rod on instrument support structure, the resistance of the channel is less than 15% of the total resistance so the flow is little affected by the presence or absence of these assemblies.

We conclude, based on the analyses and data presented, that the use of the instrumented assembly of the Instrumented Control Rod Drive (ICRD) assembly has a minor effect on the flow through the control rod channel. We conclude also that the temporary modifications to the closure assemblies to accommodate instrument leads will not affect the performance or integrity of the closures.

Having reviewed the core neutronic and thermal-hydraulic behavior of the instrumented assembly as it affects reactor safety and the health and safety of the public, we find use of the ICRD assembly acceptable. Existing Technical Specifications adequately cover operations with the ICRD assemblies in the reactor.

5.0 TESTING OF REACTOR BUILDING LOUVER SYSTEM

Background

The Fort St. Vrain Technical Specifications, Surveillance Requirement 5.5.2, Reactor Building Pressure Relief Service, indicates that the reactor building louver system is to be exercised annually. Public Service Company of Colorado has determined that this interval is too long to assure operability of these devices and PSCo's experience indicates that the louver system should be exercised quarterly to assure its operability. In a June 2, 1978 letter

PSCo notified the NRC that to increase the probability of successful operation of the reactor building louver system when called upon to operate, PSCo would intend to exercise the system on a quarterly basis regardless of reactor status. The NRC found this unacceptable since testing the reactor louvers regardless of reactor status could permit interference with the building ventilation system during fuel handling, for example, and expose a pathway for activity to the environment without any treatment. We notified PSCo of this and recommend several prerequisites prior to louver testing by letter dated December 7, 1978. These recommendations have been incorporated in a formal request for a change to the Technical Specifications dated April 22, 1979.

Evaluation

The following prerequisites must be adhered to prior and during quarterly louver testing.

(1) Reactor shall be under normal steady state operating conditions;

(2) Primary coolant pressure is within the normal envelope for existing conditions;

(3) Reactor building ventilation system is operating per Technical Specifications;

(4) No radioactive gas waste releases are in progress, nor is fuel handling being performed;

(5) No airborne activity above background as indicated by the building activity monitors;

(6) Area radiation monitors and local alarms are operable per Technical Specifications;

(7) No surveillance testing is being performed on the reactor ventilation system or the radiation monitoring systems;

(8) Only one segment (group of louvers) of the louver system shall be tested at any given time;

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(9) Communication shall exist between personnel performing the tests and the control room operators;

(10) Capability shall exist to manually shut the louver panels;

(11) Testing of the louver system shall not exceed a total duration of six (6) hours in any one quarter; and

(12) Non-compliance with any of the above conditions will require testing to be discontinued and the louver system will be returned to normal.

The louvers around the reactor building consist of 94 sections which are divided into 20 groups. 6 groups contain 4 louver sections and 14 groups contain 5 louver sections. Therefore, one group is tested at a time on approximately 5% of the total.

The functional testing of the louver system takes approximately one hour, so that if problems are encountered, an additional four hours are available to identify and resolve the problems and another hour to retest that portion of the system involved. If the system were to be tested when the primary coolant system is pressurized and the reactor is in operation, there would be a short period of time during which one section of the louver system would be open, that there would be a possibility for a release of activity to the atmosphere from the reactor building without passing through the reactor building ventilation system HEPA filters and iodine absorbers.

In the analysis of the "maximum hypothetical accident" (MHA) primary coolant helium and the associated radioactivity is released into the reactor building over a period of two hours. In order to account for uncertainties

in the operation of the reactor building louvers, we have assumed that 10% of the primary coolant system inventory will be released unfiltered from the reactor building. Using our conservative assumptions, the exclusion area boundary (590 meter) doses, viz., Thyroid 4.6 rem, Whole Body 8.6 rem, Bone 36 mrem, are well below 10 CFR 100 guidelines.

If an assumption is made that only one group of the louvers, out of a total of 20 groups, is open then the unfiltered release from the reactor building is approximately 5% and the analysis for the MHA is the bounding case. Based on the above review, we find quarterly testing of the louver system, as outlined by PSCo, acceptable.

6.0 HIGH PRESSURE HELIUM SYSTEM VALVE CLOSURE

Background

On February 25, 1976, while the reactor was shut down and depressurized, two circulators were being operated in the self-turbining mode. The 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 connecting the 88 psi high pressure bottle header to a 12 psi buffer helium header, producing a 76 psia difference. This upset the helium buffer system by preventing buffer helium recirculation and interfering with drain of bearing water to the high pressure separator.

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.

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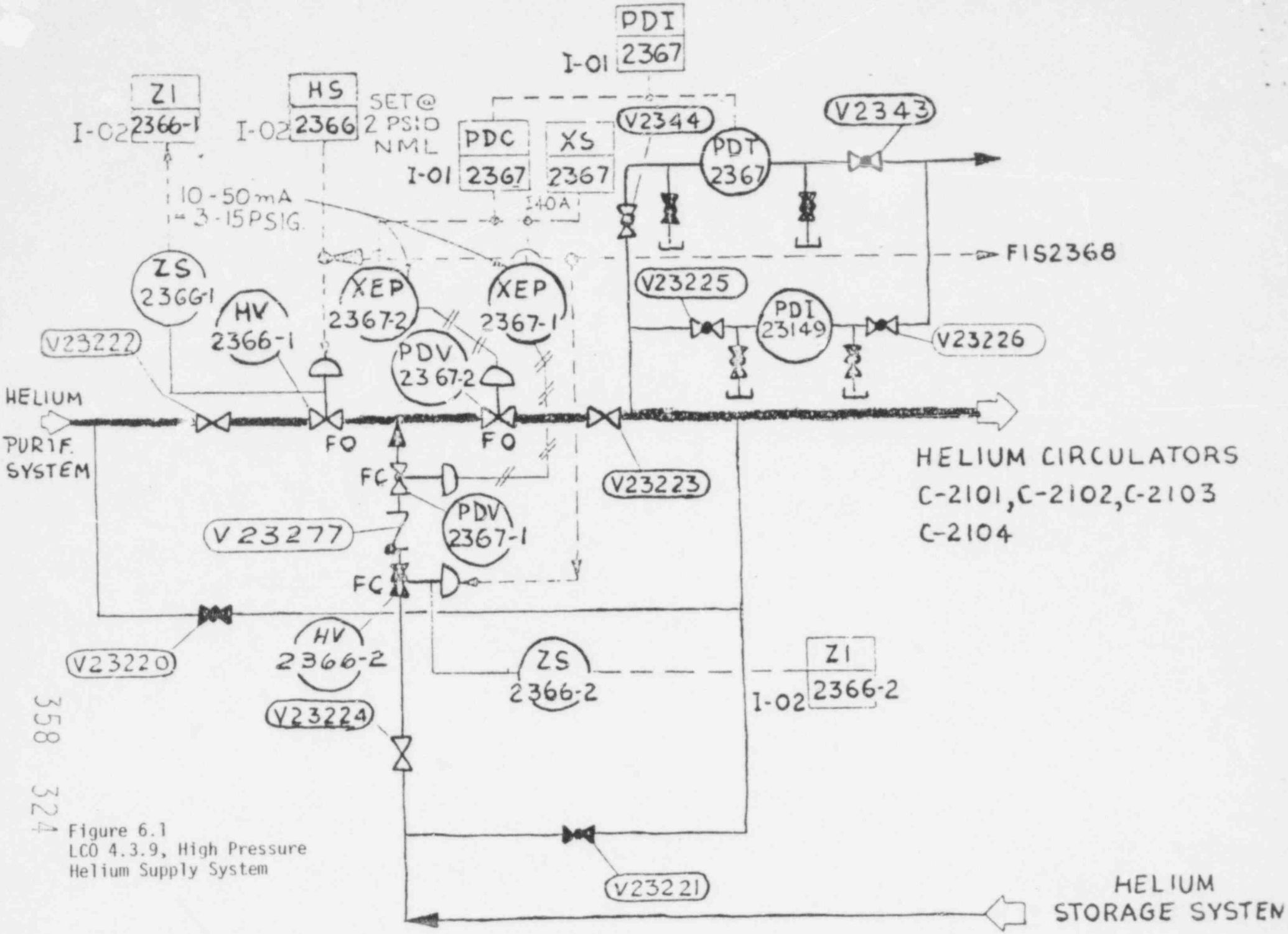

Figure 6.1
LCO 4.3.9, High Pressure
Helium Supply System

The Technical Specifications were also changed to require that valves V-2322[illegible] 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.

As a follow-up to the above mentioned changes, the system design was re-evaluated and modified so that pressure regulator valve PDC-2367-1 fails closed upon loss of electrical power, thereby precluding reoccurrence of the original event. Also in the design, the block valve HV-2366-2 is normally shut. This valve will only be opened when there is a loss or unavailability of purified helium flow from System 23, as was the operating mode when the unusual event occurred.

Evaluation

Technical Specification LCO 4.3.9, as written, insures against the unwanted accident flow only in the case of a double casualty, namely loss of purified helium flow followed by failure of the PDC-2367-1 valve to regulate pressure by accidently being energized to the wide open position. However, with the LCO as written, a sequence of events could happen when the high-pressure storage bottle system is used, which could cause a circulator trip. This is the type of occurrence which the LCO set out to specifically avoid.

We have reviewed the system design and pertinent information in the failure mode of pressure regulatory valve. We conclude that the present design of the instrumentation and controls for the system which provides buffer helium makeup from the helium storage bottle system requires two separate failures which could result in the loss of buffer helium to the

circulators. Therefore, since the design now satisfies the single failure requirements, we find it acceptable. Accordingly the proposed deletion of LCO 4.3.9 is acceptable.

7.0 FIREWATER BOOSTER PUMP

Background

While simulating safe shutdown cooling of the Fort St. Vrain reactor with firewater on 6/9/76 , it was ascertained that flow through the steam generators and helium circulator speeds did not meet the published FSAR values. In this test firewater at 125 psig was supplied to the emergency condensate header using the condensate system while the steam generator discharge pressure was being maintained at 75 psig. With these conditions, the flow through the steam generator was about 540 gpm (270,000 lbs/hr) whereas the FSAR indicates one firewater pump should supply 1,400 gpm to the steam generator and drive one helium circulator to supply about 1.5% to 2% helium flow or approximately 700 rpm for a cold core.

Subsequent testing identified and corrected the water flow problem, resulting in circulator helium flow consistent with FSAR commitments for shutdown cooling on firewater. However, subsequent reanalysis of the firewater cooldown event indicated that temperatures could exceed those predicted in the FSAR for full power operations. It has been established, however, that the present system without the booster pumps is adequate to keep temperatures within FSAR predicted values for operations up to 70% power. Provision of means for cooldown using firewater as a source of feedwater and circulator drive provides backup means for orderly cooldown in the event normally provided sources are lost. Cooldown in this mode is not essential to safety, since cooling can be provided via the liner cooling system, using normal power

sources or the ACM installation previously described, in the remote event that forced convection cooling cannot be provided.

Firewater for safe shutdown cooling is routed to the steam generators and helium circulator pelton wheel drives through the emergency condensate or emergency feedwater headers. Two boost pumps, including associated valves and piping, have been installed to boost the firewater pressure differential at the pelton wheel nozzle to a minimum of 175 psid, even though operation of only one pump is required. These pumps are not needed for operations up to 70% power, but are being provided to assure adequate cooling for operations at full power.

Firewater supply to both pumps is available from two separate headers. The manual supply and routing is from the emergency condensate header and back again through normally open valves. The only action required to obtain the boost pressure is to start either pump once firewater has been admitted to the emergency condensate header. The second source of supply to both pumps is directly from the firewater header through two normally closed valves. The discharge from both pumps can also be routed to the emergency feedwater header.

This use of the emergency feedwater header required the installation of a flexible metal hose spool and the operation of two normally closed valves. The purpose of this removable spool is to prevent leakage from the high pressure feedwater header to the lower pressure rated pumps and piping components. Normal plant operation with the emergency condensate or feedwater headers remains unchanged.

Both pumps are provided with suction and discharge pressure gauges for testing purposes. Pump operability can be verified by throttling emergency condensate to firewater supply pressure and measuring at the pump discharge and at the nozzle of the operating pelton wheel. The electrical power for the firewater booster pumps is obtained from essential Bus 1 and Bus 3.

Evaluation

We have reviewed the design and implementation of the electrical, instrumentation and control systems for the two fire water booster pumps which are to be added to the plant design. The purpose of this design change is to provide adequate circulator speed and water supply to the steam generators for safe shutdown cooling utilizing firewater and one helium circulator pelton wheel drive. The requirement to be satisfied is that no single event will result in failure of both firewater booster pumps. We have reviewed the firewater booster pumps information contained in PSC's letters to NRC (P-78205), Fort St. Vrain FCN-4089 dated 3/27/78, and FCN-3651 dated 2/20/78, in the areas of physical and electrical separation, equipment qualification, IEEE279-1971, power supply assignments and operability assurance. From our review, we have determined the following:

(1) The cable routing for each pump motor and its associated controls will be physically separated in accordance with the cable routing criteria for redundant safety related systems (reviewed and found acceptable by NRC in 1975). The cables in the three room control complex and any congested areas will be coated with Flamemastic. In all other areas adequate physical separation is maintained.

(2) The motor and control power for each pump motor is assigned to a separate and independent Class 1E power bus. As each bus is supplied power from a diesel generator, the bus loading values

were reviewed and it was found that the continuous rating of the diesel generator would not be exceeded while shutting down the plant using the firewater mode.

(3) The environmental test reports for the additional equipment were audited and found to be in accordance with previous requirements found acceptable by NRC. The seismic reports were also audited and found acceptable on the same basis except for hand switches HS-21535 and 21536 and the local pressure indicators PI-21535-1 and -2 and 21536-1 and -2. The information on the hand switches is in accordance with previous requirements found acceptable by NRC. However, this information is not included in PSCo's system of records. The local pressure indicators were not seismically qualified and therefore cannot be depended on to provide assurance that the pumps are operating properly when they are started. Further review identified that the helium circulator speed indicators, which are Class 1E and are located in the control room, will indicate a minimum speed increase of at least 200 RPM when the pump is started and operating properly. These indicators are required to determine pump operability. We consider this adequate.

We conclude that the booster pump installation is acceptable and that the proposed Technical Specification changes associated with its surveillance and use are adequate.

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8.0 CONCLUSIONS

Based on our review of the documentation referenced in this safety evaluation report, an evaluation of plant operations thus far, evaluations of the plant through site visits by NRC technical specialists, and favorable reports by the NRC Office of Inspection and Enforcement on completed work, we conclude that: (1) because the changes do not involve a significant increase in the probability or consequence 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.