DCS MS-016

MAR 24 1982

Docket No. 50-267

Mr. Don Warembourg
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Public Service Company of Colorado
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Dear Mr. Warembourg:

As you know, our review of NUREG-0737 items as they apply to your facility has progressed quite well, with a majority of the items either resolved or near resolution. Enclosure 1 presents the itemized resolution of NUREG-0737 and lists what remains to be completed. Enclosure 2 presents the same information but in tabular, summary form.

If you are ready to close out any item as stipulated in Enclosure 1, please let us know so that we may schedule the appropriate review for resolution.

Sincerely,

Original signed by
Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure: As stated

cc: w/enclosures
See next page

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ENCLOSURE 1

RESOLUTION OF NUREG-0737 REQUIREMENTS AS THEY APPLY TO FORT ST. VRAIN

I.A.1.1 SHIFT TECHNICAL ADVISOR

The STA program has been implemented by PSC. In reviewing PSC's point-by-point comparison of the INPO plans, it was determined that most of the items were in close agreement, and the exceptions taken by PSC were due mainly to their STAs being on one-hour call rather than on shift. PSC has a training program for STAs that includes familiarization with major equipment of plant systems. Implementing procedures are in place governing STA presence in the plant during normal conditions and also includes contingencies. Implementation of technical specifications and, in the early stages of an incident wherein management may not be readily available, interpretation of a technical specification by a STA is acceptable. Use of accident simulation codes by STAs in analyzing plant transients and postulated accidents is recommended. The PSC proposal for keeping the STA position (long term) and upgrading SROs, but using college level expertise as nonshift assistance, is acceptable. The Technical Specifications have been revised to include the STA duties, responsibilities, training with specific training in plant design and response, and analysis of the plant for transients and accidents. Unless further requirements are developed in the future, this item is closed.

I.A.1.3 SHIFT MANNING

PSC stated that all shift manning requirements would be met with the exception that operators would be allowed to work no more than 16 consecutive days without two consecutive days off, rather than the 14 days required by the Commission. As per I&E, R IV, SER and NRR review, the 16 consecutive day cycle for operators is acceptable. PSC plans to meet the minimum staffing requirements by July 1982. Unless further requirements are developed in the future, this item is closed.

I.A.2.1 UPGRADING OF RO AND SRO TRAINING AND QUALIFICATIONS

PSC stated that both the training and qualification programs have been upgraded and that the requirement that SRO applicants must have been a licensed operator for one year is in effect. Both license applications for training instructors and specifics of programs were submitted. The issue for simulator training will be reviewed separately. It is recommended that, because of the unique safety characteristics of the HTGR which allow more time for corrective actions to be taken in an accident and thus allow college trained STAs to be on call rather than on shift, the requirements for college level equivalent training for shift personnel be waived. Unless further requirements are developed in the future, this item is closed.

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

PSC stated that they are in compliance with the short-term requirement that instructors for training centers must demonstrate SRO qualifications and also be enrolled in appropriate requalification programs. They also submitted a training program for both short and long term requirements. Unless further requirements are developed in the future, this item is closed.

I.A.3.1 SIMULATOR EXAMS

Simulators have been shown to be useful in LWR training for operator responses along with the tracking of an event. FSV does not require quick responses for the health and safety of the public but for protection of plant equipment. Since specific requirements for a FSV simulator are not available, and a simulator for a one-of-a-kind plant would be difficult and expensive to develop, the issue of a simulator exam must be resolved at a later date by cognizant management at the Commission. In the interim PSC will provide training in accident analyses and behavior during transients as well as more hands-on experience and use of accident simulation codes.

I.B.1.2 SAFETY ENGINEERING GROUP

Not applicable to Fort St. Vrain

I.C.1 ACCIDENT PROCEDURES

Even before the TMI-2 accident, a significant amount of multiple failure transient analyses was completed by PSC and FSV procedures were written for plant cooling using highly degraded plant cooling systems. The FSV procedures that meet requirements are the "Emergency Procedures" coupled with the "Safe Shutdown and Cooling with Highly Degraded Plant Conditions Procedures". The emergency procedures deal with 18 different specific emergencies; the safe shutdown and cooling procedures provide the operators with an outline of 16 different ways to use plant systems to power the helium circulators and supply water to the steam generators, and 3 different ways to supply cooling water to the PCRV liner cooling system.

PSC issued a set of Emergency Procedures in November 1981. These procedures are being reviewed by ORNL and NRC to determine their completeness and comprehensiveness to the plant operators. Upon favorable completion of the review, this item will be closed.

I.C.5 FEEDBACK OF OPERATING EXPERIENCE

PSC has procedures for evaluating both external information and internally generated changes, operating problems and procedures. As per SER written by R IV and ORNL and NRR review, PSC is in compliance. Unless further requirements are developed in the future, this item is closed.

I.C.6 PROCEDURES FOR VERIFYING CORRECT PERFORMANCE

I&E, R IV, will continue their dialog with PSC. Systems necessary for safe shutdown will need independent verification. In FSV, some systems needed for safe shutdown are also used during normal operation; their operability can be demonstrated by proper normal operation. PSC will provide necessary input to R IV for review.

I.D.2 PLANT SAFETY-PARAMETER DISPLAY CONSOLE

The objective of the SPDS is to provide the operators with safety-related information not readily accessible on the main control panels. The design of a satisfactory SPDS would be dependent on reactor type, therefore FSV is at a disadvantage in that the entire HTGR SPDS development burden would fall on the one plant, while PWR and BWR owners could pool their resources. ORNL reviewed the requirements for a SPDS for FSV and made several recommendations; PSC is reviewing these recommendations and will continue their dialog for proper resolution.

II.B.1 COOLANT SYSTEM VENTS

Not applicable to Fort St. Vrain

II.B.2. AND II.B.3 PLANT SHIELDING AND POSTACCIDENT SAMPLING

ORNL will review the source term calculations and compare the FSAR values with those resulting from the GA fuel model. The two source term calculations are only for comparison purposes to determine the amount of conservatism that exists.

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Procedures and training in place at FSV are satisfactory with respect to prevention and mitigation of core damage. Training of all operational personnel from plant manager to licensed operator should continue to concentrate on accident prevention. The emergency procedures and corresponding operator training for Loss of Forced Circulation should be augmented with technical information on reactor coolant depressurization including alternate means of achieving depressurization. PSC will review the items recommended by ORNL for severe accident mitigation and control, and possibly include them in a training manual and for management decisions along with a decision tree to evaluate the associated risks.

II.D.1 AND II.D.3 PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES, AND DIRECT INDICATION OF VALVE POSITION

These two requirements are not truly applicable to FSV because of the unique HTGR overpressure protection requirements, and because of the very different implications of a stuck open relief valve. For LWRs the design transient for safety valves is the loss of heat sink from full power, after which the reactor core continues to transfer heat into the coolant at a high rate until power is reduced to decay heat levels. The analogous transient at FSV would be the loss of forced circulation accident, which is initiated by a trip of all four circulators and loss of feedwater to the steam generators. When this happens, the helium pressure does not rise above normal for two hours or more because essentially all of the energy released in the fuel goes into heating up the massive reactor core. The design transient for overpressure protection of the FSV PCRV is the unmitigated ingress of water into the PCRV from a broken steam generator pipe. The water flashes to steam, which increases PCRV pressure and causes safety valve actuation. The water that does not flash to steam will collect in the bottom of the PCRV where it cannot reach the safety valves which are connected to the top of the 75 ft tall PCRV interior cavity. For this transient to cause safety valve actuation, very conservative assumptions must be made, including failure of safety systems and lack of operator action.

Operation of the PCRV safety valves in not realistically expected at any time in the life of the FSV plant. The design has utilized this fact by incorporation of upstream rupture discs that must rupture before the safety valves are exposed to reactor helium. The rupture discs prevent minor operational problems associated with small coolant leaks through imperfectly seating safety or relief valves.

The consequences of a stuck open safety valve at FSV are not analogous to those that could be expected at an LWR. There is no ambiguity about the condition (i.e. void content) of the helium at any pressure, and the shutdown FSV core can be adequately cooled at any pressure down to and including atmospheric pressure.

PSC intends to rely upon the qualification testing program performed by EPRI and will abide by the recommendations that apply to FSV. Unless further requirements are developed in the future, this item is closed.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

The intent of this action item was to analyze the auxiliary feedwater system for PWRs such that the steam generator would perform as a heat sink for the reactor core power. The auxiliary feedwater system for FSV consists of two essentially independent systems: the emergency feedwater system and the emergency condensate system. These systems share a common source of water, the condensate storage tanks. The plant firewater system can also be used as a last resort.

The emergency feedwater system takes feedwater from the feedpump outlet and essentially diverts the flow from its normal path through the top two feedheaters. The emergency condensate system can feed the steam generators, reheaters and water turbines with feedwater from the condensate storage tanks. The head for this flow is supplied by the condensate feedpumps and/or the auxiliary boiler feedpump (Figure 10.2-2 of the FSAR).

By virtue of the single phase coolant, the large heat capacity of the reactor core materials, and the high-temperature capabilities of the fuel, there is a significant amount of time before core damage would result from losing the primary heat sink. This large margin of time allows for manual operation of valves to divert feedwater into either the steam generators or the reheaters. Firewater cooling can be made available by manually connecting spoolpieces.

Because the auxiliary feedwater system is not needed immediately after a loss-of-feedwater accident, the major components are used routinely during power operation or startup, and there are three independent ways of introducing water into the steam generators, the FSV design adequately addresses the intent of this action item.

Unless further requirements are developed in the future, this item is closed.

II.E.1.2 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

This item is not applicable to Fort St. Vrain because the FSV reactor has a single-phase coolant under all operating conditions, a large heat capacity of the reactor core materials, and the high temperature capabilities of the fuel. There is a significant amount of time before core damage would result from losing the primary heat sink. Therefore automatic initiation of auxiliary feedwater flow is not necessary and manual initiation is sufficient to cool the reactor before core damage might occur.

II.E.3.1 PRESSURIZER HEATER POWER

This item is not applicable to Fort St. Vrain.

II.E.4.1 HYDROGEN PENETRATION

This item is not applicable to Fort St. Vrain.

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

This requirement applies to a conventional LWR containment building as opposed to the FSV reactor which has as its primary containment barrier the PCRV inner cavity liner and primary closures and has as its secondary containment the PCRV itself and the secondary closures. The FSV reactor building is designed as a vented tertiary containment or "confinement" building. Even though the FSV containment design is different from that of conventional LWRs, the intent of the regulations, that of assuring automatic isolation of all nonessential lines, must be, and has been met. The concern for the venting of activity from the containment could logically be extended to the possibility of venting activity from the "confinement" reactor building. This problem is addressed in detail in Appendix C of the FSAR under Design Criteria 48.

Unless further requirements are developed in the future, this item is closed.

II.F.1 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

This item consists of six parts dealing with instrumentation necessary to detect certain failed conditions. The containment water level and hydrogen concentration monitors are not applicable to FSV. The containment pressure monitor is for determining if a coolant line has failed; the FSV coolant helium pressure is monitored continuously and a loss of helium is known immediately and a reactor trip is initiated by the plant protective system. FSV has provisions for continuous sampling of plant effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities.

II.F.1.1 NOBLE GAS EFFLUENT MONITOR

The effluent gases at FSV are monitored before release by instrumentation having a continuous recording and control room display. The upper range limit of 10^5 microcuries/cubic centimeter specified in the action item cannot be met with this existing instrumentation. PSC has submitted (P-79312) an analysis of the radioactive gaseous effluent for the design basis accident, having a calculated noble gas effluent activity well within the range of the stack gas monitor, and significantly below the 10^5 μCi/cc limit asspecified for water reactors. Thus the intent of this action item is met in qualitative sense (the noble gas effluent activity is monitored continuously), but the upper limit specified in the action item may be appropriate for water reactors only.

II.F.1.3 CONTAINMENT HIGH-RANGE RADIATION MONITOR

The requirement of monitoring the radiation level of the containment (reactor building for FSV) is met in a qualitative sense at FSV, but the upper limit of radiation specified at 10^8 rad/h cannot meet with the existing installed instrumentation. The power density and fuel configuration are different for water reactors and FSV. The power density is lower and the fuel is encapsulated with a multilayered ceramic coating having a high temperature capability. This coating would delay the release of highly active fission products after reactor scram. Also, the PCRV has a minimum thickness of nine feet. Consequently, the post-accident radiation levels in the reactor building would probably be lower than those of a water reactor. An appropriate radiation upper limit for the FSV reactor building environment monitoring should be lower than that specified for water reactors.

ORNL will determine the upper limits for monitoring of noble gas effluent activity and reactor building radiation level appropriate for FSV. These upper limit values for instrumentation should be based on the physical properties of the reactor and not on the fact that high level radiation monitors are commercially available.

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

The installed instrumentaiton at FSV is sufficient for detection of inadequate core cooling and, combined with appropriate emergency procedures, meets the intent of this item as it applies to FSV. Unless further requirements are developed in the future, this item is closed.

II.G.1 PRESSURIZER POWER SUPPLIES

This item is not applicable to Fort St. Vrain.

II.K.2 ORDERS ON B&W PLANTS

This item is not applicable to Fort St. Vrain.

II.K.3 FINAL RECOMMENDATIONS, B&O TASK FORCE

Except for the following subitems, this item is not, in whole, applicaple to Fort St. Vrain.

II.K.3.17 ECCS OUTAGES

PSC will refine their definition of ECCS and will determine what systems or parts thereof constitute the ECCS for FSV and will continue to monitor ECCS outages. PCS will develop a trend analysis system at a later date.

III.A.1.1 EMERGENCY PREPAREDNESS, SHORT TERM

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

III.A.2 EMERGENCY PREPAREDNESS

PSC is in compliance with most of these requirements. The early warning alert system has been inspected during a January 1982 review. Dialog will continue between PSC and NRC for complete resolution.

III.D.1.1 PRIMARY COOLANT OUTSIDE CONTAINMENT

Even though the requirement is "for PWRs and BWRs", the intention is for all power reactors to review the possibilities for serious leaks during postulated accidents. Due to the inherent design and safety features of an HTGR, many of the specific requirements are not applicable. Because of the PCRV containment, normally only the small primary helium sampling lines would contain highly radioactive gases after an accident. Radioactive gas and liquid cleanup systems are designed to filter, monitor and store effluents as required at FSV, and the systems are well monitored. Unless further requirements are developed in the future, this item is closed.

III.D.3.3 INPLANT RADIATION MONITORING

PSC has responded to this item in their letter dated December 30, 19'0 (P-80444). Unless further requirements are developed in the future, this item is closed.

II.D.3.4 CONTROL ROOM HABITABILITY

PSC has responded (P-80438) to this action item and claims that although they disagree with a few specific figures of the guidelines, they meet the intent of all the listed regulatory guides. This response should be evaluated from a human factors viewpoint. Most other aspects have been incorporated by PSC.

TABLE 1. SUMMARY OF NUREG-0737 ACTION ITEM REVIEWS

Item No.	Brief Title	Apply to FSV	Status
I.A.1.1	Shift Tech. Advisor	yes	Closed. STA on one-hour call.
I.A.1.3	Shift Manning	yes	Overtime issue closed; shift constituency being reviewed by Division of Human Factors.
I.A.2.1	Training Upgrades	yes	Closed. Simulator training reviewing separately; college level equiv. training for shift personnel waive
I.A.2.3	Training Programs	yes	Closed. In compliance.
I.A.3.1	Simulator Exams	yes	To be reviewed by Division of Human Factors.
I.B.1.2	Safety Engr. Group	no	Closed.
I.C.1	Accident Procedures	yes	Emergency Procedures under review by ORNL.
I.C.5	Feedback of Experience	yes	Closed. In compliance.
I.C.6	Verify Operations	yes	I&E, R IV will review.
I.D.1	Control Room Design	yes	Closed.
I.D.2	Safety Param. Display	yes	PSC reviewing ORNL recommendations.
II.B.1	Coolant Syst. Vents	no	Closed.
II.B.2	Postaccident Shielding	yes	ORNL is reviewing source term calculations; will compa FSAR with GA fuel model results.
II.B.3	Postaccident Sampling	yes	
II.B.4	Trng. for Core Damage	yes	PSC reviewing ORNL recommendations.
II.D.1	Test Relief Valves	yes	Closed; PSC will follow EPRI recommendations as applicable.
II.D.3	Valve Pos. Indication	no	Closed.
II.E.1.1	Aux. Feedwater Eval.	yes	Closed.

Table 1 Cont'd.

Item No.	Brief Title	Apply to FSV	Status
II.E.1.2	Aux. FW Indicators	no	Closed.
II.E.3.1	Press. Heater Power	no	Closed.
II.E.4.1	Hydrogen Penetration	no	Closed.
II.E.4.2	Containment Isol.	yes	Closed.
II.F.1	Noble Gas Monitor	yes	II.F.1.1 and II.F.1.3. ORNL will determine upper limits for FSV.
II.F.2	Detect Inadequate Cooling	yes	Closed; PSC is in compliance.
II.G.1	Pressurizer Power	no	Closed.
II.K.2	B&W orders	no	Closed.
II.K.3.x	B&O Task Force	no	Closed.
II.K.3.17	ECCS outages	yes	PSC will monitor outa es but will develop trend analysis at a late: date.
II.K.3.18	Auto. Depressurization	no	Closed.
II.K.3.30	Small-break LOCA	no	Closed.
II.K.3.31	10 CFR 50.46	no	Closed.
III.A.2	Emergency Preparedness	yes	Most aspects incorporated.
III.D.1.1	Hot System Integrity	yes	Closed; PSC in compliance.
III.D.3.3	Iodine Instruments	yes	Closed; PSC in compliance.
III.D.3.4	Control Rm. Habitable	yes	PSC response needs Human Factor Engineering review; most aspects incorporated.