Public Service Company of Colorado

16805 WCR 19 1/2, Platteville, Colorado 80651

February 9, 1984
Fort St. Vrain
Unit #1
P-84046

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Generic Letters 83-37 and 83-36

ATTACHMENTS: G-82084, P-83352, G-83166, G-83020, G-80049, G-81257, P-80438 (excerpt only), G-83333

Dear Mr. Eisenhut:

We have reviewed the request for amendments to plant Technical Specifications pursuant to the completion of plant modifications which may have been required for specific NUREG-0737 items. PSC's position on each specific item is detailed in Attachment 1 to this letter. It is important to note that the NUREG-0737 items were developed with Lignt Water Reactor technology and accident modes in mind, and as such, several distinctions and exceptions for the FSV HTGR design have been taken by PSC in meeting the intent of the individual action items. We have included all referenced correspondence as attachments to this submittal.

Based on our review of the subject, we have concluded that no revisions to the Technical Specifications are required, because either the requirements are already present in the existing Technical Specifications or the item is not applicable to Fort St. Vrain.



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NOTE: ORIG TO REGION IV
DELETE REG. FILE

If you have questions with regards to this submittal, please contact Chuck Fuller at (303)785-2223.

Very truly yours,

Don W. Warembourg

Don W. Warembourg
Manager, Nuclear Production

Attachments

DWW/dkh

Attachment 1

- Item II.B.1 (Reactor-Coolant-System Vents)

  This item was determined to be not applicable to the Fort St. Vrain HTGR in correspondence from Robert A. Clark, Chief Operating Reactors Branch #3, to Don Warembourg, dated March 24, 1982 (G-82084, attached), and, as such, no amendment to the Technical Specifications is required.

- Item II.B.3 (Post-Accident Sampling)

  In correspondence from F. J. Borst, Radiation Protection Manager, to G. L. Madsen, Chief Reactor Project Branch 1, dated October 28, 1983 (P-83352, attached), PSC committed to (1) make provisions for the sampling of primary coolant with radiation levels greater than 1 mR/hr at the detector, and (2) provide a procedure to estimate the extent of core damage based on radionuclide concentrations and core temperatures. These two items are beyond the scope of routine primary coolant sampling procedures described in the existing Health Physics and Radiochemistry procedures, and accordingly will be described in new Radiological Emergency Response Plan Implementing Procedures (RERP-IP). The RERP-IPs are referenced in section AC 7.4 of the FSV Technical Specifications. No further action is required.

- Item II.E.1.1 (Auxiliary Feedwater System Evaluation)

  This item was closed in correspondence from Robert A. Clark, Chief Operating Reactors Branch #3, to Don Warembourg, dated March 24, 1982 (G-82084, attached). In the referenced correspondence from the NRC, the following was stated: "The auxiliary feedwater system for FSV consists of two essentially independent systems: the emergency feedwater system and the emergency condensate system ... 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."

  Limiting Condition for Operation 4.3.4 of the FSV Technical Specifications states that "the reactor shall not be operated at power unless the emergency condensate header and emergency feedwater header are operable." We consider this item to be adequately addressed by the existing FSV Technical Specifications.

- Item II.F.1.1 (Accident-Monitoring; Noble Gas Monitor)

Generic Letter 83-37 calls for administrative controls in the event of failure of the referenced noble gas effluent monitor. The administrative controls cited in Generic Letter 83-37, in essence, state that in the event of monitor failure, an alternate method for monitoring the effluent should be initiated as soon as practical, but no later than 72 hours after the identification of the failure of the monitor. Additionally, if the monitor operability is not restored within 7 days after monitor failure, Generic Letter 83-37 calls for submission of a special report to the NRC within 14 days following the event, outlining the cause of monitor failure, compensatory actions taken in the interim, and planned schedule for restoring the monitor operability.

The recently amended FSV Technical Specifications (Amendment No. 37, January 1, 1984) make the following provisions for failure of a noble gas effleunt monitor:

(1) ELCO 8.1.1 g) 3) states that, in the event of failure of both noble gas monitors, gaseous effluent releases from the Reactor Building ventilation system, exclusive of releases from the gas waste holdup system, may continue, provided grab samples are taken at least once per eight hours and analyzed for noble gas activity within 24 hours, or the release is continuously monitored using auxiliary sampling equipment;

(2) ELCO 8.1.1 g) 4) states that if both noble gas monitors become inoperable, gaseous effluent releases from the gas waste holdup system may continue provided that duplicate samples of the gas waste system contents are analyzed in accordance with ELCO 8.1.1 d) and at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve line-up; and,

(3) ELCO 8.1.1 g) 8) states that with one or more of the radioactive gaseous effluent monitoring instruments inoperable (this means from the noble gas, radioiodine, or particulate effluent monitors), best efforts shall be exerted to return the instruments to operable status within thirty days, and, if unsuccessful, the failure to restore operability in a timely manner shall be explained in the next semi-annual Radioactive Effluent Release Report.

PSC maintains that these provisions which provide more conservative compensatory measures in the event of monitor failures, in unison with manual back-up measures for determination of high level effluent release rate referenced in the RERP-IPs (cited in AC 7.4 of the FSV Technical Specifications) more than adequately meet the intent of NUREG-0737 Item II.F.1.1 and the Generic Letter 83-37 sample Technical Specification. It should also be noted that the referenced ELCOs of the FSV Technical Specifications were developed, to the greatest extent possible, in accordance with the current Standard Technical Specifications, and accepted by the Nuclear Regulatory Commission, effective January 1, 1984, as Amendment 37 to the FSV Technical Specifications.

- Item II.F.1.2 (Accident-Monitoring; Iodine/Particulate Sampling)

The sample Technical Specification shown in Generic Letter 83-37 calls for the implementation and maintenance of a program to include (1) training of personnel; (2) procedures for sampling and analysis; and (3) provisions for maintenance of sampling and analysis equipment. Further, it is stated that, "it is acceptable to the staff, if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operations manuals."

FSV has installed radioiodine and particulate on-line effluent monitoring systems which meet the intent of NUREG-0737 Item II.F.1.2 criteria. The means for utilizing the data provided by the effluent monitors is described in the RERP-IPs, referenced in AC 7.4 of the FSV Technical Specifications. Additionally, the training of plant personnel and calibration of related equipment is maintained in appropriate procedures as required by AC 7.4.a.1 of the FSV Technical Specifications. PSC feels that it has met the intent of this item and that revisions to the FSV Technical Specifications are not required.

- Item II.F.1.3 (Accident-Monitoring; Containment High-Range Monitor)

The PSC position on this item is that existing area monitors with a range of up to 10 rad/hr adequately monitor reactor building radiation levels during Design Basis Accident No. 1 (not anticipated to exceed 1.4 rad/hr). Additionally, PSC committed to the installation of a high-range radiation monitor with an upper limit of $10^4$ rad/hr (RT-93250-14). This monitor has been recently installed and is currently in operation.

The PSC status on this item was evaluated in I&E Inspection Report 50-267/82-21 (G-83020, attached), and an open item (267/8221-07) was created. The open item called for the following:

- licensee determination that "during accident situations an adequate number of area monitors would be operating to determine radiation levels in the reactor building;"
- "Procedural changes and/or equipment modifications to be certain the accident could be 'followed' by the area monitors even though more than one monitor is connected to an alarming annunciator;" and,
- "Installation of the ordered high range containment monitor."

PSC has recently completed installation of the referenced high range monitor and will be addressing closure of the open item. If the PSC and NRC efforts determine that revisions to the Technical Specifications are required to close this item, they will be requested at that time.

- Item II.F.1.4 (Accident-Monitoring; Containment Pressure Monitor)

The NRC, in correspondence from G. L. Madsen, Chief Reactor Project Branch 1, to O. R. Lee, Vice President, Electric Production, dated April 27 1983 (G-83166, attached), stated that "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 protection system."

The operability requirement for plant protection system scram parameters for low reactor pressure is stated in Table 4.4.1 of LCO 4.4.1 of the FSV Technical Specifications. We consider this item adequately addressed by the existing Technical Specifications.

- Item II.F.1.5 (Accident-Monitoring; Containment Water Level Monitor) and
- Item II.F.1.6 (Accident-Monitoring; Containment Hydrogen Monitor)

The NRC, in correspondence from G. L. Madsen, Chief Reactor Project Branch 1, to O. R. Lee, Vice President, Electric Production, dated April 27, 1983 (G-83166, attached), stated that "the containment water level and hydrogen concentration monitors are not applicable to FSV;" as such, no amendment to the Technical Specifications is appropriate.

- Item II.F.2 (Instrumentation for Detection of Inadequate Core Cooling)

In correspondence from Themis Speis, Chief Advanced Reactors Branch of the Division of Project Management, to J. K. Fuller, Vice President, Engineering and Planning, dated March 31, 1980 (G-80049, attached), the following was stated:

"The requirement for the installation of indication that would apprise the operator of the margin to saturation of the primary coolant or primary coolant level in the reactor vessel are not applicable to the Fort St. Vrain reactor ... Instrumentation presently available to detect inadequate core cooling consists of helium circulator speed, reactor differential pressure, core outlet thermocouples, ratio of core power to helium flow, and differential pressure across the helium circulators. It should be noted that even though the above instrumentation exists to determine inadequate core cooling, the limiting DBE [sic] for which the plant was analyzed was the loss of all core cooling, primary and secondary. The consequences of this accident as indicated in the FSAR show that upon depressurization, heat from the core will be transferred to the PCRV. The PCRV is cooled by redundant safety grade cooling systems to preserve its integrity. Since direct core cooling is not necessary as indicated by the FSAR analysis; we have determined that the licensee does not need to provide any additional instrumentation to detect inadequate core cooling and, therefore satisfies this requirement."

Additionally, correspondence from George Kuzmycz, Project Manager Operating Reactors Branch #3 of the NRC Division of Licensing, to Public Service Company of Colorado, dated December 23, 1981 (G-81257, attached) and from Robert A. Clark, Chief Operating Reactors Branch #3 of the NRC Division of Licensing to Don Warembourg, Manager, Nuclear Production, dated March 24, 1982 (G-82084, attached), corroborate the PSC interpretation of this item.

FSV Technical Specification surveillance requirements, section 5.4.1, 5.4.3, 5.4.4, 5.4.5, 5.4.6, 5.4.8, and 5.1.7 serve to assure the availability of reliable operating data to inform operators and staff of both core and PCRV cooling conditions. PSC feels that it has met the intent of this item with its existing instrumentation and technical specifications.

- Item III.D.3.4 (Control Room Habitability Requirements)

PSC evaluated the FSV Control Room habitability in correspondence from Don Warembourg, Manager, Nuclear Production, to Darrell Eisenhut, Director of Division of Licensing, Office of Nuclear Reactor Regulation, dated December 20, 1980 (P-80438, excerpt attached). In that correspondence, PSC committed to the installation of a chlorine detector at the Chemical Building, though this detector was not required per NRC Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." Existing radiation detectors and associated alarms and control actions were deemed adequate.

In correspondence from G. L. Madsen, Chief Reactor Project Branch 1, to O. R. Lee, Vice President, Electric Production, dated September 8, 1983 (G-83333, attached), this item was closed. PSC has since determined that the installed detector may be excessively sensitive and, therefore, difficult to clear from alarm. For that reason, PSC is in the process of evaluating the performance of this detector. However, this evaluation does not affect the acceptability of the system function as it now exists.

Automatic control actions on the Control Room ventilation system for airborne radiological contaminants are currently tested in accordance with the FSV Technical Specifications, ESR 8.1.1.b, and are verified prior to each gas waste release, or monthly, whichever is more frequent. No amendments to the existing Technical Specifications are anticipated as a result of item III.D.3.4.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 24 1982

Docket No. 50-267

G-82084
rec'd 3-30-82

Mr. Don Warembourg
Nuclear Production Manager
Public Service Company of Colorado
16805 WCR 19 1/2
Platteville, Colorado 80651-9298

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,

Robert A. Clark

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

Enclosure:
As stated

cc: See next page

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# 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, applicable 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, 1980 (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.

Enclosure 2

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 waived. |
| 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 compare 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 outages but will develop trend analysis at a later 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. |

Public Service Company of Colorado

16805 WCR 19 1/2, Platteville, Colorado 80651

October 28, 1983
Fort St. Vrain
Unit #1
P-83352

Mr. G. L. Madsen
Chief, Reactor Project Branch 1
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

SUBJECT: NUREG-0737, Item II. B. 3,
"Post-Accident Sampling System"
Reference: P-82423, G-83349

Dear Mr. Madsen:

We offer the following comments with respect to the three unresolved criteria as contained in your letter dated September 22, 1983 (G-83349):

Criterion (1)

We stated in our letter P-82423 that we had two gas-driven generators available to provide emergency power for sampling. At the time that letter was written, we were utilizing the generators to obtain air samples in the field, thus the generators were dedicated for that purpose. Currently we do not use the generators to obtain field air samples, and the generators are not dedicated. In the event that generators would be required we would be able to obtain generators and collect and analyze air samples within the three-hour time frame.

Criterion (2)

You recommend that Public Service Company of Colorado "...should provide a procedure... to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data. Guidance ...is attached (Attachment 1)." Unfortunately guidance was not attached, but was graciously provided by Mr. Phil Wagner on October 17, 1983. We agree with this recommendation and will develop an appropriate procedure. Our estimated completion data is June 1, 1984.

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Criterion (8)

Please note that our backup site for sample analyses is Colorado State University, not the University of Colorado as stated in your letter.

Criterion (9)

You recommend that we "...should provide on site capability to measure these higher activities (in primary coolant samples) by means such as sample dilution of collimation of the sample beam." We agree with this recommendation, and feel that the best approach to this recommendation would be to reduce the volume of the samples sufficiently so that they may be counted in the radiochemistry laboratory. Thus far we have determined that primary coolant samples with radiation levels exceeding 1 mR/hr at the detector cannot be counted. We are currently evaluating the possibility of making changes to our sample collection system to accomodate the small (approximately 0.04 scc) sample volume needed. Our anticipated completion date for this item is June 1, 1984.

Criterion (10)

You recommend that we "...provide additional information on the measurement of coolant activities in the time period beyond the first few hours after the onset of a severe accident." As mentioned under criterion (9), we anticipate that we will be able to reduce the volume of primary coolant samples sufficiently such that even at 24 hours post-shutdown, when the primary coolant activity is maximized, we will be able to collect and analyze the samples on site. Thus samples can be collected at any desired times post-shutdown. We plan no further action on this criterion.

Please contact me if you have additional questions on this matter.

Very truly yours,

Frederick J. Borst

Frederick J. Borst
Radiation Protection Manager

FJB/clm

G

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

G-83166
rec'd 4-29-83

April 27, 1983

Docket No. 50-267

Mr. O. R. Lee, Vice President
Electric Production
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

Dear Mr. Lee:

The Commission has issued the enclosed Order confirming your commitments to implement those post-TMI related items set forth in NUREG-0737 for which the staff requested completion on or after July 1, 1981. This Order is based on commitments contained in your letters responding to the NRC's Generic Letters 82-05 and 82-10 dated March 17, 1982, and May 5, 1982, respectively.

The Order references your letters and, in its attachments, contains lists of the applicable NUREG-0737 items with your schedular commitments. As discussed in the Order, several of the items listed in Generic Letter 82-10 will be handled outside of this Order.

The Commission's intention when it issued NUREG-0737 was that items would be completed in accordance with the staff's recommended schedule. However, our evaluation of your proposed schedule exceptions concludes that the proposed delays are acceptable. Among other things, the Order requires implementation of these items in accordance with your proposed schedule.

Some of the items set forth in the attachment to the Order are subject to post-implementation review and inspection. Our post-implementation review and/or the development of Technical Specifications may identify alterations to your method of implementing and maintaining the requirements. Any identified alterations will be the subject of future correspondence.

A copy of this Order is being filed with the Office of the Federal Register for publication.

Sincerely,

G L Madsen

G. L. Madsen, Chief
Reactor Project Branch 1

Enclosure: Confirmatory Order

cc w/enclosure: See next page

8305200516

Fort St. Vrain
cc list

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Department of Local Affairs
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Chairman, Board of County Commissioners
of Weld County, Colorado
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Regional Representative
Radiation Programs
Environmental Protection Agency
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Don Warembourg
Nuclear Production Manager
Public Service Company of Colorado
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Platteville, Colorado 80651

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of

PUBLIC SERVICE COMPANY OF COLORADO

(Fort St. Vrain Nuclear Generating Plant)

Docket No. 50-267

ORDER CONFIRMING LICENSEE COMMITMENTS ON POST-TMI RELATED ISSUES

I.

Public Service Company of Colorado (the licensee) is the holder of Facility Operating License No. DPR-34 which authorizes the operation of the Fort St. Vrain Nuclear Generating Station (the facility) at a steady-state power level not in excess of 842 megawatts thermal. The facility is a high temperature gas-cooled reactor (HTGR) located at the licensee's site in Weld County, Colorado.

II.

Following the accident at Three Mile Island No. 2 (TMI-2) on March 28, 1979, the Nuclear Regulatory Commission (NRC) staff developed a number of proposed requirements to be implemented on operating reactors and on plants under construction. These requirements include Operational Safety, Siting and Design, and Emergency Preparedness and are intended to provide substantial

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additional protection in the operation of nuclear facilities based on the experience from the accident at TMI-2 and the official studies and investigations of the accident. The staff's proposed requirements and schedule for implementation are set forth in NUREG-0737, "Clarification of TMI Action Plan Requirements." Among these requirements are a number of items, consisting of hardware modifications, administrative procedure implementation and specific information to be submitted by the licensee, scheduled to be completed on or after July 1, 1981. On March 17, 1982, a letter (Generic Letter 82-05) was sent to all licensees of operating power reactors for those items that were scheduled to be implemented from July 1, 1981 through March 1, 1982. Subsequently, on May 5, 1982, a letter (Generic Letter 82-10) was also sent to all licensees of operating power reactors for those items that were scheduled for implementation after March 1, 1982. These letters are hereby incorporated by reference. In these letters each licensee was requested to furnish within 30 days pursuant to 10 CFR 50.54(f) the following information for items which the staff had proposed for completion on or after July 1, 1981:

(1) For applicable items that have been completed, confirmation of completion and the date of completion, (2) For items that have not been completed, a specific schedule for implementation, which the licensee committed to meet, and (3) Justification for delay, demonstration of need for the proposed schedule, and a description of the interim compensatory measures being taken.

III.

Public Service Company of Colorado (PSC) responded to Generic Letter 82-05 by letter dated March 26, 1982, and provided supplemental information by letter dated June 30, 1982; PSC responded to Generic Letter 82-10 by letter dated June 1, 1982. In these submittals, PSC confirmed that some of the items identified in the Generic Letters had been completed, some were not applicable to Fort St. Vrain, and made firm commitments to complete the remainder. The attached Tables summarizing the licensee's schedular commitments or status were developed by the staff from the Generic Letters and the licensee-provided information.

Generic Letters 82-05 and 82-10 addressed nineteen and sixteen items, respectively. Of the 11 items listed in Generic Letter 82-10 requiring a response, six items are not included in this Order. Item I.A.1.3.2 is part of a separate rulemaking; Items I.C.1, III.A.1.2 (2 items), and III.A.2.2 will be handled separately following Commission actions that would proceed as a result of its consideration of Commission Paper SECY 82-111, as amended; and Item II.K.3.30 and II.K.3.31 (one item) is not required until one year after staff approval of the generic model and staff review of these models has not been completed. Some items are still under review to resolve philosophical differences in reactor technology, and one, the simulator item, is under policy review to account for a one-of-a-kind, unique facility. The staff has reviewed the licensee's submittals and determined that the licensee's response is acceptable based on the following:

I.A.3.1 - Simulator Exams

Since specific requirements for a Fort St. Vrain simulator are not available, and a simulator for a one-of-a-kind, unique reactor 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 NRC and therefore is not included in this Order. In the interim, PSC is providing training in accident analyses and plant behavior during transients as well as more hands-on experience and use of accident simulation codes for its operators. Until the issue of a simulator exam is resolved, PSC is in compliance with the intent of the requirement.

II.B.2 - Plant Shielding
II.B.3 - Post Accident Sampling

The existing methods and instrumentation for monitoring the plant exhaust stack gas and primary activity are acceptable for the previously analyzed design basis accident at Fort St. Vrain. However, it appears that for some worst-case severe accident sequence scenarios beyond the design basis, the on-line monitors could go off scale and personnel access might not be permissible either to obtain a primary coolant grab sample or to read an inplace monitor. Until an NRC ruling becomes available on severe accident analyses, PSC is in compliance with this item.

II.B.4 - Training to Mitigate Core Damage

Procedures and training in place at FSV are satisfactory with respect to prevention and mitigation of core damage. PSC is reviewing procedures for alternate depressurization methods and will possibly include them in a training manual.

II.F.1 - Accident Monitoring

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 protection system. FSV has provisions for continuous sampling of plant effluents for post-accident releases of radioactive iodines and particulates and onsite laboratory capabilities. The ruling mentioned in Items II.B.2 and II.B.3, above, may also affect this item.

The effluent gases at FSV are monitored before release by instrumentation having a continuous recording and control room display. Permanent online monitors have an upper limit reading of 10 μci/cc. Installed semiportable monitors are capable of 10E5 μci/cc.

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 be met with the existing installed instrumentation, (which has a limit of 10E4 rad/h). 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 a reactor scram. Also, the Prestressed Concrete Reactor Vessel (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.

We find, based on the above evaluation, that 1) the licensee has taken corrective actions regarding the delays and has made a responsible effort to implement the NUREG-0737 requirements noted; 2) there is good cause for the several delays; and 3) as noted above, interim compensatory measures have been provided.

In view of the foregoing, I have determined that these modifications and actions are required in the interest of public health and safety and should, therefore, be confirmed by Order.

IV.

Accordingly, pursuant to Sections 103, 161i, and 161o of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED EFFECTIVE IMMEDIATELY THAT THE LICENSEE SHALL:

Implement and maintain the specific items described as complete in the attachments to this Order. Incomplete items shall be completed by no later than the dates shown in the attachments (as described in the licensee's submittals noted in Section III herein) and maintained thereafter.

V.

The licensee may request a hearing on this Order within 20 days of the date of publication of this Order in the Federal Register. A request for a hearing shall be addressed to the U.S. Nuclear Regulatory Commission, Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 1000, Arlington, Texas 76011. A copy shall also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

If a hearing is requested by the licensee, the Commission will issue an Order designating the time and place of any such hearing.

If a hearing is held concerning this Order, the issue to be considered at the hearing shall be whether the licensee should comply with the requirements set forth in Section IV of this Order.

This Order is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Gagliardo

James E. Gagliardo, Director
Division of Resident, Reactor Project
and Engineering Programs
Region IV

Dated at Arlington, Texas,
this 27th day of April, 1983

Attachments:
1. Licensee's Commitments on Applicable NUREG-0737 Requirements from Generic Letter 82-05
2. Licensee's Commitments on Applicable NUREG-0737 Requirements from Generic Letter 82-10

## LICENSEE COMMITMENTS ON APPLICABLE NUREG-0737 ITEMS FROM GENERIC LETTER 82-05

| ITEM | TITLE | NUREG-0737 SCHEDULE | REQUIREMENT | LICENSEE'S COMPLETION SCHEDULE (OR STATUS)* |
|---|---|---|---|---|
| I.A.3.1 | **Simulator Exams | 10/1/81 | Include simulator exams in licensing examinations. | To be determined |
| II.B.2 | Plant Shielding | 1/1/82 | Modify facility to provide access to vital areas under accident conditions. | Complete |
| II.B.3 | Post-accident sampling | 1/1/82 | Install upgrade post-accident sampling capability. | Complete |
| II.B.4 | Training for Mitigating Core Damage | 10/1/81 | Complete training program. | Complete |
| II.E.1.2 | Aux FW Indication & Flow Indicator | 7/1/81 | Modify instrumentation to level of safety grade. | Not applicable |
| II.E.4.2 | Containment Isolation Dependability | 7/1/81 | Part 5 - lower containment pressure setpoint to level compatible with normal operation. | Complete |
| II.E.4.2 | Containment Isolation Dependability | 7/1/81 | Part 7 - isolate purge and vent valves on radiation signal. | Complete |

*Where complete date refers to a refueling outage (the estimated date when the outage begins), the item will be completed prior to the restart of the facility.

**Not part of Confirmatory Order.

LICENSEE COMMITMENTS ON APPLICABLE NUREG-0737 ITEMS
FROM GENERIC LETTER 82-05

| ITEM | TITLE | NUREG-0737 SCHEDULE | REQUIREMENT | LICENSEE'S COMPLETION SCHEDULE (OR STATUS)* |
|---|---|---|---|---|
| II.F.1 | Accident Monitoring | 1/1/82 | (1) Install noble gas effluent monitors. | Complete |
| | | 1/1/82 | (2) Provide capability for effluent monitoring of iodine. | Complete |
| | | 1/1/82 | (3) Install in-containment radiation-level monitor. | Complete |
| | | 1/1/82 | (4) Provide continuous indication of containment pressure. | Complete |
| | | 1/1/82 | (5) Provide continuous indication of containment water level. | Not applicable |
| | | 1/1/82 | (6) Provide continuous indication of hydrogen concentration in containment. | Not applicable |

*Where completion date refers to a refueling outage (the estimated date when the outage begins), the item will be completed prior to the restart of the facility.

LICENSEE COMMITMENTS ON APPLICABLE NUREG-0737 ITEMS FROM GENERIC LETTER 82-05

| ITEM | TITLE | NUREG-0737 SCHEDULE | REQUIREMENT | LICENSEE'S COMPLETION SCHEDULE (OR STATUS)* |
|---|---|---|---|---|
| II.K.3.15 | Isolation of HPCI & RCIC Modification | 7/1/81 | Modify pipe break detection logic to prevent inadvertent isolation. | Not applicable |
| II.K.3.19 | Interlock on Recirculation Pump | 7/1/82 | Install interlocks on recirculation pump loops. | Not applicable |
| II.K.3.22 | RCIC Suction | 1/1/82 | Modify design of RCIC suction to provide automatic transfer to torus. | Not applicable |
| II.K.3.24 | Space Cooling for HPCI/RCIC | 1/1/82 | Confirm the adequacy of space cooling for HPCI/RCIC. | Not applicable |
| II.K.3.27 | Common Reference Level | 7/1/81 | Provide common reference level for vessel level instrumentation. | Not applicable |
| II.K.2.10 | Safety Grade Trips | 7/1/81 | Install anticipatory reactor trips | Not applicable |

*Where completion date refers to a refueling outage (the estimated date when the outage begins), the item will be completed prior to the restart of the facility.

... ON APPLICABLE NUREG-0737 ITEMS FROM GENERIC LETTER 82-10 Attachment 2

| ITEM | TITLE | NUREG-0737 SCHEDULE | REQUIREMENT | LICENSEE'S COMPLETION SCHEDULE (OR STATUS) * |
|---|---|---|---|---|
| I.A.1.3.1 | Limit Overtime | 10/1/82 per Gen. Ltr. 82-12 dtd. 6/15/82 | Revise administrative procedures to limit overtime in accordance w/NRC Policy Statement issued by Gen. Ltr. No. 82-12, dtd June 15, 1982. | Complete |
| I.A.1.3.2 | **Minimum Shift Crew | To be superseded by proposed Rule. | To be addressed in the Final Rule on Licensed Operator Staffing at Nuclear Power Units. | To be addressed when Final Rule is issued |
| I.C.1 | **Revise Emergency Procedures | Superseded by SECY 82-111 | Reference SECY 82-111, Requirements for Emergency Response Capability | To be determined |
| II.D.1.2 | RV and SV Test Programs | 7/1/82 | Submit plant-specific reports on relief and safety valve program | Not applicable |
| II.D.1.3 | Block Valve Test Program | 7/1/82 | Submit report of results of test program. | Not applicable |
| II.K.3.18 | ADS Actuation | 9/30/82 | Submit revised position on need for modifications. | Not applicable |
| II.K.3.30 & 31 | **SBLOCA Analysis | 1 year after staff approval of model. | Submit plant-specific analyses. | Not applicable |
| III.A.1.2 | **Staffing Levels for Emergency Situations | Superseded by SECY 82-111 | Reference SECY 82-111, Requirements for Emergency Response Capability | To be determined |
| III.A.1.2 | **Upgrade Emergency Support Facilities | "" "" | "" "" | "" "" |
| III.A.2.2 | **Meteorological Data | "" "" | "" "" | "" "" |
| III.D.3.4 | Control Room Habitability | To be determined by licensee | Modify facility as identified by licensee study. | Complete |

*Where completion date refers to a refueling outage (the estimated date when the outage begins) the item will be completed prior to the restart of the facility.

**Not Part of Confirmatory Order

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
811 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

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JAN 06 1983

G-83020
rec'd
1-11-83

Docket: 50-267/82-21

Public Service Company of Colorado
ATTN: O. R. Lee, Vice President
Electric Production
P. O. Box 840
Denver, Colorado 80201

Gentlemen:

This refers to the radiation protection operations inspection conducted by Messrs. R. Baer and W. Holley of this office during the period August 30 - September 3, 1982, of the activities authorized by NRC License DPR-34 for Fort St. Vrain Nuclear Power Plant, and to the discussion of our findings held with Mr. E. D. Hill and other members of your staff.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

Within the scope of the inspection, no violations or deviations were identified.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room unless you notify this office, by telephone, within 10 days of the date of this letter, and submit written application to withhold information contained therein within 30 days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

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Should you have any questions concerning this letter, we will be pleased to discuss them with you.

Sincerely,

T. R. Westerman for

G. L. Madsen, Chief
Reactor Project Branch 1

Enclosure:
Appendix - NRC Inspection Report 50-267/82-21

cc:
D. W. Warembourg, Nuclear
Production Manager
Fort St. Vrain Nuclear Station
P. O. Box 368
Platteville, Colorado 80651

J. Gahm, Quality Assurance
Fort St. Vrain Nuclear Station
P. O. Box 368
Platteville, Colorado 80651

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

Report: 50-267/82-21

License: DPR-34
Docket: 50-267

Licensee: Public Service Company of Colorado (PSCo)
P. O. Box 840
Denver, CO 80201

Facility: Ft. St. Vrain Nuclear Generating Station (FSV)

Location: Platteville, Colorado

Inspection Conducted: August 30 - September 3, 1982

Inspectors: R. E. Baer, Radiation Specialist — 12/22/82 Date

for W. L. Holley, Radiation Specialist — 12/22/82 Date

Approved by: Blaine Murray, Chief, Facilities Radiation Protection Section — 12/22/82 Date

Inspection Summary

Inspection conducted August 30 - September 3, 1982 (Report 50-267/82-21)

Areas Inspected: Routine, unannounced inspection by regional based inspectors of transportation activities, radiation protection operation and select NUREG-0737 items including: management controls; preparation of packages for shipment; delivery of completed packages to carrier; receipt of packages; incident reporting; indoctrination and training program; audit program; recordkeeping; radiation protection audits; radiation protection training; radiation protection procedures; exposure control; and posting, labeling and control. The inspection involved 66 onsite hours by two inspectors.

Results: No violations or deviations were identified. Six open items are discussed in paragraphs 5.f, 6.e, 7.b, 7.d, 7.e., and 7.f. One unresolved item is discussed in paragraph 6.c.

~~8303180223~~

DETAILS

1. Persons Contacted

a. Public Service Company of Colorado (PSCo)

*E. D. Hill, Station Manager
*T. J. Borst, Radiation Protection Manager
*W. S Franek, Site Engineering Manager
C. F. ller, Technical Services Engineering Supervisor
*J. W. Gahm, Quality Assurance Manager
R. E. Huster, Quality Assurance Auditing Coordinator
V. McGaffic, Radiochemical Supervisor
*L. W. Singleton, Quality Assurance Operations Superintendent
*T. E. Schleiger, Health Physics Supervisor
*R. Wadas, Training Supervisor
W. E. Woodard, Health Physicist

b. Other Personnel

*G. L. Plumlee III, NRC Resident Inspector

The NRC inspectors also interviewed several other licensee employees, including health physics, radiochemistry technicians and administrative personnel.

*Denotes those present during the exit interview.

2. Licensee Action on Previous Inspection Findings

(Closed) Open Item (267/8115-01) - Health Physics Technicians Training and Experience: This item was discussed in NRC Health Physics Appraisal Report 50-267/80-13 and NRC Inspection Report 50-267/81-15 and involved the lack of adequate guidance to ensure that contract health physics technicians training and experience meet ANSI N18.1-1971 criteria. The licensee revised Procedure HPP-46, Section 4.7 and defined 2 years to mean 24 months experience in the specialty. This item is considered closed.

(Closed) Open Item (267/8115-02) - Installation of Personnel Monitoring Equipment: This item was discussed in NRC Health Physics Appraisal Report 50-267/80-13 and NRC Inspection Report 50-267/81-15 and involved the lack of personnel monitoring equipment at the exit from the protected area. The licensee purchased and installed two walk-through portal monitors. One monitor was installed at the exit to the reactor building, the other at the security building. This item is considered closed.

(Closed) Unresolved Item (267/8115-03) - Contractor Health Physics Qualifications: This item is discussed in NRC Inspection Report 50-267/81-15 and involved several technicians who were given credit toward the 2 years of experience through training programs and overtime work on the job.

The licensee stated the technicians in question have terminated and have been replaced by personnel who meet the 24-month experience criteria in accordance with Procedure HPP-46. This item is considered closed.

(Closed) Violation (267/8128-01) - Titanium Sponge: This item was identified in NRC Inspection Report 50-267/81-28 and involved the titanium sponge in the helium purification system which had been out of service for 18 months. The NRC inspectors verified by visual inspection that the titanium sponge had been installed and was in service as required by Technical Specification LCO 4.8.1.c. This item is considered closed.

(Open) Unresolved Item (267/8128-01) - Reactor Building Exhaust Filters: This item was identified in NRC Inspection Report 50-267/81-28 and involved the documentation of filter tests required by Technical Specifications. The licensee stated that the reactor building exhaust filters had been tested to meet the requirements of the Technical Specifications during the system start-up testing. This will be reviewed during a future inspection.

3. Open Items Identified During This Inspection

(Open) Open Item (267/8221-01) - Whole Body Counter Calibration: The licensee had not developed a comprehensive calibration and testing program that satisfies the recommendations of ANSI-N343-1978. See paragraph 6.e for details.

(Open) Open Item (267/8221-02) - Radioactive Waste Retraining Program: The licensee had not developed a formal training/retraining program for personnel involved in the transfer, packaging, and transport of radioactive materials. See paragraph 5.f for details.

(Open) Open Item (267/8221-03) - Primary Coolant Sample Lines: The licensee had not determined potential for line blockage, activity plate-out or sample distortion. See paragraph 7.b for details.

(Open) Open Item (267/8221-04) - Noble Gas Effluent Monitors: The licensee had not determined that the noble gas effluent monitors met ANSI N13.1 design criteria. See paragraph 7.d for details.

(Open) Open Item (267/8221-05) - Reactor Building Ventilation Exhaust Monitor: The license had not determined effect of entrained moisture on iodine sampling capabilities. See paragraph 7.e for details.

(Open) Unresolved (267/8221-06) - Radiation Worker Training Program: The licensee had not revised the radiation worker training program to include the recommendations of Regulatory Guides 8.27 and 8.29. See paragraph 6.c for details.

(Open) Open Item (267/822-07) - Containment High Radiation Monitors: The licensee had not determined operability of the containment radiation monitors during elevated temperature conditions following an accident. See paragraph 7.f for details.

4. Regulatory Documents

The NRC inspectors verified that the licensee had current copies of applicable Nuclear Regulatory Commission (NRC) and Department of Transportation (DOT) regulations so as to be able to comply with their requirements.

The licensee subscribes to Dat-O-Line, Inc., Charleston, South Carolina, radioactive waste management service which provides current copies of 10 CFR Part 71 (NRC), 40 CFR (DOT), 39 CFR (Postal Service), and State and nonfederal regulations. Additional information is provided about Notices, Pending Rules, and Proposed Rules as prepared by the DOT and other authorities and extracted from the Federal Register. All the above categories are updated with a biweekly supplement.

No violations or deviations were identified.

5. Transportation Activities

a. Management Controls

The management control system for radioactive material management is described in general in Administrative Procedure Q-1 and more specifically in Procedure P-3. The health physics supervisor is designated as the individual with the responsibility to insure the proper shipment and receipt of all radioactive material to and from the plant. The health physics department is responsible for the collection, compaction or solidification, preparation of the shipment and loading of the transport vehicle. The licensee generates a minimal quantity of radioactive waste material from maintenance activities and, therefore, does not provide dedicated personnel for radioactive waste activities.

The licensee had developed and implemented procedures for the various processes and details of the radioactive material handling program. These procedures included:

HPP-23, "Receiving Radioactive Materials," Issue 6

HPP-26, "Radioactive Material Control and Handling." Issue 4

HPP-30, "Radioactive Material Classification, Packaging, and Labeling," Issue 0

The quality assurance - operations department is responsible for planned and periodic audits of the radioactive waste management program. The licensee had developed Procedure Q-18, "Quality Assurance Audit and Monitoring Program," Issue 5, to provide guidance in implementing these audits. Audits are scheduled on a biannual frequency.

No violations or deviations were identified.

b. Preparation of Packages for Shipment

The licensee's program for preparation of by-product radioactive material for shipment was reviewed against the requirements of 10 CFR Parts 71.12, 71.31, 71.35, 71.53 and 71.54, 49 CFR Parts 172 and 173, and the following generally accepted codes, guides and standards:

Regulatory Guide 7.1 - Administrative Guide for Packaging and Transporting Radioactive Material.

Regulatory Guide 7.4 - Leaking Tests on Packages for Shipment of Radioactive Materials.

ANSI N14.10.1 - Administrative Guide for Packaging and Transporting Radioactive Materials.

ANSI N14.5 - Leak Tests on Packages for Shipment of Radioactive Materials.

The licensee had developed and implemented procedures for preparation of radioactive materials for shipment. These procedures (see list in paragraph 5) included requirements for visual inspection prior to filling or loading the package; marking of package weight and contents; labeling requirements appropriate for the type of package; and radiation and contamination limits for packages.

The NRC inspectors noted by observation of the radioactive waste compaction and storage facility that the licensee used, for shipments of low specific activity radioactive waste, steel drums manufactured in accordance with DOT specification 17H (49 CFR Part 178.118).

The licensee had not made a shipment of low specific activity radioactive waste since receiving their operating licensee in 1973.

No violations or deviations were identified.

c. Delivery of Completed Packages to Carrier

The licensee's program for delivery of completed packages to a carrier for transport was reviewed against the requirement of 10 CFR Part 71.55 and 49 CFR Parts 100 to 199. Activities for delivery of completed packages to a carrier were governed by previously mentioned Procedure HPP-30.

The NRC inspectors examined this procedure for consistency with regulatory requirements and to determine whether it covered all aspects. The licensee had not shipped any radioactive waste, therefore, records were not available to verify adherence to procedural requirements.

No violations or deviations were identified.

d. Receipt of Packages

The licensee's program for the receipt of packages containing radioactive material was examined against the requirements of 10 CFR Part 20.205 and conformance to Procedure HPP-23.

The NRC inspectors reviewed this procedure for compatability with regulatory requirements and to determine whether it covered all aspects of the work being carried out.

No violations or deviations were identified.

e. Incident Reporting

The NRC inspectors reviewed the licensee's procedures for incident reporting against the requirements of 49 CFR Parts 171.15 and 171.16. The reporting of incidents were not covered by plant procedures. The licensee had not offered for shipment any radioactive waste material, and plans on using a contract carrier when a shipment is made.

No violations or deviations were identified.

f. Indoctrination and Training Program

The licensee's indoctrination and training program, as it pertains to the packaging of low level radioactive waste for transport and burial, was examined against the provisions of IE Bulletin No. 79-19 and the licensee's response to this bulletin.

The NRC inspectors reviewed documentation of training conducted since January 1980 for personnel involved in the transfer, packaging, and transport of radioactive material.

Two members of the health physics staff had attended a vendor conducted workshop on packaging and transportation of radioactive materials. Two additional staff members are scheduled to attend this workshop in the fall of 1982.

The NRC inspectors noted that health physics personnel receive training in radioactive waste systems and processes and is documented in the individual's "Health Physics Technician Training Check-off List," but the licensee had not developed a formal retraining program for personnnel involved in the transfer, packaging, and transport of radioactive materials. The licensee's station training program administrative manual, HPC-2 states in Section 4.4.3.a, "The Health Physics and Radiochemistry Department Retraining program is conducted as considered necessary by the radiation protection manager and the training supervisor." This item is considered open pending implementation of a suitable training and retraining program which details

retraining frequency and subject material to be presented (267/8221-02).

No violations or deviations were identified.

g. Audit Program

The licensee's audit function for the low-level radioactive waste transfer, packaging, and transport activities was examined against the requirements of 10 CFR Part 71 and IE Bulletin No. 79-19 and within the framework of the following generally accepted guidance:

-Regulatory Guide 1.33 - Quality Assurance Program Requirements

-ANSI N18.7-1976 - Administrative Controls and Quality Assurance for Operational Phase of Nuclear Power Plants.

The NRC inspectors reviewed the audits of transportation activities, including the latest audit, conducted by the licensee:

QAA-1501-79-02, dated September 24-26, 1979 QAA-1501-81-01, dated August 19 - September 3, 1981

These audits were conducted in accordance with the licensee's written procedures listed in paragraph 5 and included a checklist for the areas reviewed. Deficiencies identified during these audits, recommendations and comments relating to the areas audited were contained on Form QAA-602. All deficiencies were corrected in a timely manner. The inspectors also reviewed audits QAA-501-80-01, dated June 3 - August 13, 1980, and QAA-501-82-01, dated August 11-31, 1982, which were conducted on spent fuel shipments.

No violations or deviations were identified.

h. Recordkeeping

The licensee had not made a shipment of low-level radioactive waste. No records were available for review to determine compliance with the requirements of 10 CFR Part 71.62.

No violations or deviations were identified.

i. Spent Fuel Shipping Program

1) Responsibility

The responsibility for Special Nuclear Material (SNM) has been delegated to the Technical Services Department in Administrative Procedure G-6, "Control of Special Nuclear Material,"

Issue 6. Section 4.1.3 states, "Technical Services prepares and controls all transmittal forms necessary for the transfer or possession of Special Nuclear Material." The reactor engineer has been assigned the responsibility for SNM documentation.

2) Procedures

The licensee has developed and implemented procedures for all phases of fuel handling; these are designated Fuel Handling Procedures (FHP). Specifically, Procedures FHP-5 and FHP-6 relate to the shipment of spent fuel and cover the handling, loading, and inspection (including checklists) of the spent fuel cask.

3) Spent Fuel Shipments

The licensee had made 12 shipments during 1982 of spent fuel to the Department of Energy (DOE) contractor-operated facility in Idaho. The licensee had a copy of a letter from the Public Service Company of Colorado, dated April 14, 1982, to the DOE contractor requesting license information to receive spent fuel and the reply letter from the Idaho Operations Office DOE, dated May 6, 1982, which stated the contractor was authorized to receive spent fuel.

4) Spent Fuel Shipping Container

All shipments of spent fuel had been made in shipping containers designed USA/6346/B Model FSV-1. A Certificate of Compliance, Number 6346, Revision 4, dated September 25, 1980, which pertained to these containers was available for review. This Certificate of Compliance expires September 30, 1985.

5) Notifications And Reports

The NRC inspectors reviewed records for the advanced Notification of Governors of states through which spent fuel was being transported and Region IV, as required by 10 CFR Parts 71.5b and 73.72. The licensee had made the proper notification prior to scheduled shipments.

The NRC inspectors reviewed select spent fuel shipment documentation for shipments made during 1982.

No violations or deviations were identified.

6. Radiation Protection Operations

a. Radiation Protection Personnel Staffing and Qualifications

The NRC inspectors reviewed the station organization to determine if there had been any changes affecting the radiation protection program and examined the staffing level of the health physics department. The licensee's organization and staffing level are depicted below:

1 - Radiation Protection Manager (1)*

1 - Health Physics Supervisor (1)

1 - Health Physicist (1)

1 - Senior Health Physics Technicians (1)

7 - Health Physics Technicians (6)

*The numbers in parentheses denote the present staffing level.

The NRC inspectors reviewed the resumes and training records of the three supervisory level and all seven of the licensee's staff health physics technicians. All health physics technicians met the selection and qualification criteria of ANSI-N18.1-1971, and the radiation protection manager and health physics supervisor met the recommendations of Regulatory Guide 1.8. The licensee has supplemented station health physics technicians with four contractor-supplied health physics technicians. The NRC inspectors reviewed the resumes and training records of these personnel. Three of these persons did not meet the qualification criteria, but were not assigned to function in positions having senior health physics responsibilities.

No violations or deviations were identified.

b. Radiation Protection Audits

The NRC inspectors reviewed the licensee's audit program relating to radiation protection operations conducted by the quality assurance group. Audits are conducted on a biannual frequency in accordance to written procedures listed in paragraph 5.a. of this report. Quality Assurance Audit, Health Physics QAA-602-81-01, April 20-28, 1981, were reviewed for scope and timely response to the deficiencies identified. The NRC inspectors did not identify any problems in this area.

No violations or deviations were identified.

c. Radiation Protection Training

The NRC inspectors discussed initial and refresher radiation worker training with the training supervisor. The present training program appears to satisy the requirements of 10 CFR Part 19.12; however, did not include all recommendations of Regulatory Guides 8.27 and 8.29. The licensee stated Regulatory Guide 8.27 (dated March 1981) is titled, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants," and they are a high temperature gas cooled plant and not applicable to them. The NRC inspectors referred them to Section D, Implementation, which states that, "In the case of training programs at operating reactors, appropriate modifications to such programs should be made consistent with this guide as soon as practicable and no later than one year after publication of this guide." The NRC inspectors considered the licensee's facility an operating reactor and, therefore, were required to comply with these recommendations.

The licensee stated that they would review their training program against the recommendations of Regulatory Guide 8.27, in addition to the Institute of Nuclear Power Operations (INPO) which has published a proposed standard radiation worker training program. The licensee had planned to conform to the INPO training program and was scheduled to attend a meeting in mid-September on this program. This is considered to be an unresolved item (267/8221-06) pending revision of the training program to meet the recommendations of Regulatory Guide 8.27.

The NRC inspectors reviewed selected training records for new employees, regular plant staff, and health physics technicians. This review indicated requirements of 10 CFR Part 19.12 and the Station Training Program Administrative Manual were being met.

No violations or deviations were identified.

d. Radiation Protection Procedures

The NRC inspectors reviewed the licensee's procedures to determine compliance with 10 CFR Part 20 requirements and recommendations contained in Regulatory Guides 1.33, 8.8, 8.9, 8.15, 8.25, and ANSI Standards, N13.1-1969, N13.11, N13.12, N18.1-1971, N18.7-1976, N322-1977, N323-1978, and N343-1978, and NUREG-0761.

The following procedures have been issued or revised since the previous radiation protection program inspection:

HPP-9, Establishing and Posting Controlled Areas, Issue 5

HPP-14, Analytical Instrumentation Room, Issue 11

HPP-19, Calibration of the Model 315 A-L Beckman CO Analyzer, Issue 4

HPP-20, Calibration of Radiation Detection Instruments, Issue 12

HPP-23, Receiving Radioactive Materials, Issue 6

HPP-27, Personnel Dosimetry, Issue 6

HPP-37, Emergency Kit Checklist, Issue 14

HPP-44, Radioactive Material Spill, Issue 2

HPP-46, Technical Specifications Related to Health Physics, Issue 2

HPP-48, Routine Maintenance, Inspection, and Cleaning of Respiratory Equipment, Issue 5

HPP-56, Reactor Building Exhaust Stack Discharge Activity Calculations, Issue 2

HPP-58, Calibration Procedure for Airflow Measuring Devices, Issue 2 HPP-60, Sampling Procedure for the Reactor Building Sump Effluent, Issue 1

HPP-61, Film Badge and Finger Ring Response Check, Issue 2

HPP-62, Portable Grab Sampler Operation Using 1260cc Marinelli Beaker, Issue 1

RCP-40, Operation and Calibration of the Whole Body Counting System, Issue 1

The NRC inspectors discussed these procedures with the radiation protection manager and noted where procedures were weak or inconsistent with plant operation. Procedure HPP-27, Section VI A.1, stated that personnel would receive a whole body count at the Colorado State Department of Health when terminating employment. However, the licensee had recently installed their own onsite whole body counting system and no longer used the Colorado State Department of Health system. All newly issued or revised procedures had been reviewed, approved, and issued in accordance with Station requirements.

No violations or deviations were identified.

e. Exposure Control

The NRC inspectors reviewed the station bioassay whole body counting operation and calibration program for agreement with the recommendations of ANSI N343-1978. The NRC inspectors discussed with the chief radiochemist, Procedure RCP-40, "Operation and Calibration of the Whole Body Counting System," and RCP-28, Routine Laboratory Functions. Procedure RCP-28 states the normal frequency for energy calibration check is weekly; the licensee performs the calibration check daily. ANSI-N343-1978, Section 15.3.3(3) states that these checks should be performed at least daily while the system is in use, and should be made at approximately 8-hour intervals. The licensee used radionuclides of Cr-51, Co-60, and Cs-137 for the body and lung calibration, and I-131 for thyroid calibration. Only one activity level of each radionuclide is used. ANSI-N343-1978, Section 15.2, recommends a series of measurements on various standard phantoms loaded with known quantities of radioactivity. These measurements shall be for the range of organ burdens of interest, i.e., 60-20,000 nanocuries of Co-60. The NRC inspectors inquired if any effort was being made to participate in an inter-calibration program with other facilities, as recommended in the standard. The licensee stated they would review the ANSI standard and also discuss this with the instrument vendor. This item is considered open (267/8221-01) and will be reviewed during a future inspection.

No violations or deviations were identified.

f. Posting, Labeling, and Control

The NRC inspectors, during a tour of the licensee's facilities on September 1-2, 1982, determined that the licensee was in compliance with the requirements of 10 CFR 20.203b, 20.203e, 20.203f, 20.207, and station procedures for the posting, labeling, and control of radioactive material and radiation areas. No high radiation areas or airborne radioactivity areas were noted.

Radiation work permits were reviewed against licensee surveys and independent measurements made by the inspectors to determine whether they afforded an adequate level of protection to workers.

No violations or deviations were identified.

7. NUREG-0737, "Classification of TMI Action Plan Requirements"

The NRC inspectors reviewed the licensee's progress and commitments in meeting the post-TMI requirements ac[illegible] NUREG-0737 for:

Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operation."

Item II.B.3, "Postaccident Sampling Capability"

Item II.E.4.2, "Containment Isolation Dependability, Position (7), Containment Purge and Vent Isolation Valves Must Close on a High Radiation Signal"

Item II.F.1, "Additional Accident Monitoring Instrumentation"

Attachment 1, "Noble Gas Effluent Monitor"

Attachment 2, "Sampling and Analysis of Plant Effluent"

Attachment 3, "Containment High-Range Radiation Monitor"

Item III.D.3.3, "Improved Inplant Iodine Instrumentation Under Accident Conditions"

Item III.D.3.4, "Control-Room Habitability Requirements"

a. Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operation"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

(c) Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(e) Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(f) Letter, March 2, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(g) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(h) Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

(i) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(j) Letter, October 22, 1981, to S. J. Ball (ORNL) from D. W. Warembourg (FSV)

(k) Memorandum, January 29, 1982, to file (USNRC R4) from T. F. Westerman (USNRC)

(l) Letter, March 19, 1982, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(m) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(n) Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(o) Letter, June 1, 1982, to D. G. Eisenhut (USNRC) from D. W. Waremboug (FSV)

(p) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Waremboug (FSV)

(q) Standard Review Plant 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Features Components Outside Containment"

(r) Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants 19 - Control Room."

(s) Standard Review Plan, Section 6.4, "Habitability Systems"

(t) Calc - FSV Shielding Design Review for DBA - 1, Document No. C-70-002, September 23, 1980

(2) Discussion

An explanation of this item, per NUREG-0737, is given in the following:

"With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas in which personnel occupancy may be unduly limited or safety equipment may be

unduly degraded by the radiation fields during postaccident operations of these systems.

"Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility."

The licensee performed a design review which included a design basis accident where dose rates were calculated at various points in the plant.

(3) Conclusions

NUREG-0737 is written primarily for light water reactors which will not apply in every detail to the Fort St. Vrain High Temperature Gas Cooled Reactor. Therefore, the source terms in Regulatory Guides 1.3 and 1.4 are not applicable. Presently, Oak Ridge National Laboratory is performing a review which compares the source term calculations of the FSAR and the Gulf Atomic fuel model to determine the most conservative source term. The design review was done using the source term used in the FSAR.

During an accident situation, personnel would spend limited periods of time performing tasks in the reactor building. The design review gave dose rates that were acceptable to meet the General Design Criteria (GDC) to perform these tasks.

The control room peak gamma dose rate is less than 6 mR/h in an accident situation and this meets GDC 19 criteria for continuous occupancy.

The technical support center has a calculated dose rate of approximately 1 mrem/h.

The following plant systems, which require postaccident operation capability from the control room, were considered in the design review; reactor plant cooling water system, helium circulator auxiliary system, secondary coolant system, purification cooling water system, fire protection system, and Alternate Cooling Method.

The NRC inspectors determined that this item meets the conditions adequately as set forth in NUREG-0737, and should be considered closed.

No violations or deviations were identified.

(b) Item II.B.3, "Postaccident Sampling Capability"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to all Operating Nuclear Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. F. Swart (FSV)

(c) Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(e) Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(f) Letter, February 20, 1980, from F. E. Swart (FSV)

(g) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(h) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(i) Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

(j) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(k) Letter, October 22, 1981, to S. J. Ball (ORNL) from D. W. Warembourg (FSV)

(l) Memorandum, January 29, 1982, to File (USNRC) from T. F. Westerman (USNRC)

(m) Letter, March 19, 1982, to all licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

(n) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(o) Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(p) Letter, June 1, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(q) Letter, July 28, 1982, to R. A. Clark (USNRC) from D. W. Warembourg (FSV)

(r) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(s) FSV Radiochemistry Procedure - 34, 'Sample Handling and Log-In'

(t) FSV Health Physics Procedure - 14, 'Analytical Instrumentation Room'

(2) Discussion

Briefly, Item II.B.3 of NUREG-0737 requires the following:

"A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

"A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet criteria.

"In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial

sample (Regulatory Guide 1.3 and 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample within a shift)."

(3) Results

The design review, previously mentioned in paragraphs 7.a.(2) and (3), gives results whereby dose rates needed for this item (sample collection, transport, and analysis) to meet GDC-19 criteria is satisfied.

The NRC inspectors determined that samples of the reactor coolant and reactor building atmosphere could be collected in less than 1 hour. Also, the collection and analyses can be made in less than 3 hours. The licensee's computerized analysis system has a radioisotope library that is more than sufficient for the number of isotopes to be determined in an accident situation. In obtaining these samples, no auxiliary system has to be isolated.

In addition to the ability to obtain samples of the primary coolant, FSV has a continuous on-line sampler (RT 9301) that monitors primary coolant activity and provides a continuous indication of fuel degradation. Remote control room readout for this system provides a continuous indication of fuel integrity without the necessity of entering the reactor building.

Since FSV is a high temperature gas-cooled reactor, boron and chloride analyses during the accident are not applicable; hydrogen levels are determined with a gas chromatograph.

The radiochemistry laboratory, analyzing procedures, and equipment restricts background radiation levels to where sample analysis results will not contain objectable error. The Canberra Series 80 multi-channel analyzer with GeLi detectors are used in conjunction with a Digital Equipment Company PDP 11/44 computer to give the necessary accuracy, range, and sensitivity needed for isotopic determination. The offsite facilities of the State of Colorado Public Health and Colorado State University laboratories will be used as backup for sample analysis. The ventilation exhaust from the sampling station is filtered with charcoal adsorbers and HEPA filters.

The NRC inspectors had one area of concern. NUREG-0737 states that consideration should be given to:

Provisions for reducing plate out in sample lines, minimizing sample loss or distortion, and preventing blockage of sample lines by loose material, etc., in the sampling apparatus. These potential problems have not been investigated by the licensee. This item is considered open (267/8221-03) pending the licensee's investigation of the sampling system.

No violations or deviations were identified.

c. Item II.E.4.2, "Containment Isolation Dependability, Position (7) Containment Purge and Vent Isolation Valves Must Close on a High Radiation Signal"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut.

(b) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

(c) Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart

(e) Letter, February 20, 1980, to S. A. Varga (USNRC) from F. E. Swart

(f) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(g) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg

(h) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg

(i) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(j) Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(k) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(2) Discussion

NUREG-0737 is written for Light Water Reactors (LWR) and states that the containment purge and vent isolation valves must close on a high radiation signal. To clarify this further, NUREG-0737 stipulates that these valves must be closed during operation of the reactor, and to implement this, the sealed-closed purge isolation valves shall be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

At Fort St. Vrain (FSV) the "containment" consists of the Prestressed Concrete Reactor Vessel (PCRV) and the interspaces between the primary and secondary closures at PCRV penetrations. The "containment" pressure in the interspaces is always maintained above primary coolant pressure to ensure that no primary coolant helium can flow into "containment" if a leak develops in the primary coolant boundary, or into the environment if a leak develops in the secondary closure. The normal operating containment pressure is 710 psig and the normal reactor coolant pressure is about 5-15 psi lower. Also, the FSV reactor building is not considered to be containment and there is not any way to isolate it. The reactor building louver system releases to the environs for two minutes wherever the pressure in the building increases to 2.5 inches of water.

(3) Conclusions

The design of Fort St. Vrain (FSV) does not require provisions to purge and vent any secondary containment space, thus this item is only applicable to Light Water Reactors. Therefore, the NRC inspectors considers this closed.

No violations or deviations were identified.

d. Item II.F.1, "Additional Accident Monitoring Instrumentation

(1) Attachment 1, "Noble Gas Effluent Monitor"

(a) Documents Reviewed

i. Letter, June 15, 1979, to G. Kuzmycz (USNRC) from D. W. Warembourg (FSV)

ii. Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

iii. Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

iv. Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

v. Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

vi. Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

vii. Letter, February 20, 1980, to S. A. Varga (USNRC) from F. E. Swart (FSV)

viii. Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

ix. Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

x. Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

xi. Memorandum, January 29, 1982, to File from T. F. Westerman (USNRC)

xii. Letter, March 19, 1982, to all Licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

xiii. Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

xiv. Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

xv. Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

xvi. ANSI N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities"

xvii. FSV Radio Chemistry Procedure 30, "Isotopic Calibration of Gaseous Activity Monitors"

xviii. SR 5.8.1 cd-Q, "Radioactive Gaseous Effluent System Calibration"

xix. FSV RERP-DOSE, "Offsite Dose Calculation Methodology"

xx. FSV Health Physics Procedure 56, "Reactor Building Exhaust Stack Discharge Activity"

(b) Discussion

NUREG-0737 position for this item is that the noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions. Multiple monitors are considered necessary to cover the ranges of interest.

Noble gas effluent monitors with an upper range capacity of E+05 uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of E+05 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest.

Licensees shall provide continuous monitoring of high-level, postaccident releases of radioactive noble gases from the plant.

The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. Externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

The design description shall include the following information:

System description, including:

instrumentation to be used, including range or senitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and,

the source of power to be used.

Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown is needed.

(c) Conclusions

Because FSV is a HTGR and the above NUREG-0737 position is for LWRs, some appropriate consideration should be given to this fact when reviewing the noble gas effluent monitors. In the meeting referenced in the memorandum of January 29, 1982, (7.d.(1)(a)(xi)), it was decided that the Oak Ridge National Laboratory (ORNL) will determine the upper limit for the Reactor Plant Ventilation Exhaust Stack monitor and that the upper limit values for this instrumentation should be based on the physical properties of the reactor instead of the fact that high level radiation monitors are commercially available. The NRC inspectors were unable to determine when ORNL would finish making the

upper limit determination. Also, in the letter of March 24, 1982, (7.f.(1)(a)(viii.)) it was stated by the NRC that the licensee has met the intent of this item (Item II.F.1, Attachment 3) in a qualitative sense, and that the upper limit of the monitors specified in this item may be appropriate for LWR's only.

Due to this item's (II.F.1, Attachment 1 of NUREG-0737) stipulation that the noble gas effluent monitor must have an upper range capacity of E+05 uCi/cc, the licensee has met the intent of this requirement by designing an emergency stack monitor even though this is for light water reactors and not high temperature gas cooled reactors. The purpose of this monitor is to provide an estimate of noble gas activity released from the reactor building exhaust stack. This monitor consists of a lead shielded collimator (located on level 10 of the turbine building) and two portable radiation detection instruments, an Eberline E-500 detector with a GM probe and a ion chamber rate meter. Procedure HPP-56 describes how the readings from these instruments can be converted to exhaust concentration in uCi/cc. This system has a range of E-01 to E+05 uCi/cc.

The NRC inspectors determined that the maximum noble gas activity expected in the exhaust stack gas during an accident situation is 5E-02 uCi/cc which is approximately an order of magnitude below the maximum of the range (6.3 E-01 uCi/cc) of the Reactor Plant Ventilation Exhaust Stack monitor RT 7324-1 and approximately three orders of magnitude below the maximum of the range (1.5 E+01 uCi/cc) of radiation monitor RT 7324-2. The ranges of these inline monitors, RT 7324-1/2, are 9.5E-07 uCi/cc to 6.3E-01 uCi/cc and 2.3E-05 uCi/cc to 1.5E+01 uCi/cc, respectively, which give good range overlap and the necessary continuous range to cover normal operations (ALARA) through postaccident accident situations. These monitors are checked and calibrated on a monthly and quarterly schedule, respectively, according to RCP-30 and SR 5.8.1 cd-Q and are located on the turbine deck at elevation 4829 feet. Their readout modules (RIS 7324-1/2) and recorder (RR 93256) are calibrated on an annual basis and readout continuously in the control room.

The procedure to convert these monitor readings to release rates for offsite dose calculations is given in station procedure, RERP-DOSE, "Offsite Dose Calculation Methodology," Issue 1.

These monitor systems are on the essential power bus which provides uninterrupted power from the emergency diesel generators upon loss of normal power.

The licensee was unable to determine if these monitors were designed per ANSI N13.1 criteria. This item is considered open (267/8221-04) pending:

- the licensee's determination that monitors meet ANSI N13.1 criteria.
- the completion of ORNL Reactor Plant Ventilation Exhaust Stack monitor upper limit determination.

No violations or deviations were identified.

e. Item II.F.1, "Additional Accident Monitoring Instrumentation

(1) Attachment 2, "Sampling and Analysis of Plant Effluents"

(a) Documents Reviewed

i. Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

ii. Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

iii. Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

vi. Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

vii. Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

viii. Letter, March 20, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

ix. Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

x. Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

xi. Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

xii. Letter, August 25, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

xiii. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities"

xiv. SR 5.8.1cd-Q, "Radioactive Gaseous Effluent System Calibration"

xv. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants 19 - Control Room"

xvi. FSV Health Physics Procedure - 53, "RT 7325-1 and RT 73437 Filter and Cart Removal (Emergency Accident Conditions)"

(b) Discussion

The clarifications for this item (Item II.F.1, Attachment 2) in NUREG-0737 states that the licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates. Licensees shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition.

The sampling system design shall be such that plant personnel could remove samples, replace sampling media, and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC 19 of 5 rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air

flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the NRC will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ±20 percent. Further departure from the isokinetic condition need not be considered in design. Corrections for nonisokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969, may be considered on an ad hoc basis.

Effluent streams which may contain air with entrained water, e.g., air ejector dis[illegible], shall have provisions to ensure that the adsor[illegible] not degraded while providing a representative [illegible], e.g., heaters.

(c) Conclusions

The particulate and iodine monitors continuously draw the effluent through a filter assembly and observe the radioactive buildup on the filter by means of a gamma scintillation detector. The paper-type (Whatman GF/A 47 mm) filter traps particles down to 0.3 micron with an efficiency greater than 95 percent. The filter is backed up a silver zeolite cartridge (RADeCo "Radioiodine Sampler" Model GY-130) which collects iodine in gaseous form with an efficiency greater than 90 percent. Both the filter and charcoal are monitored continuously by the gamma scintillation detector.

The plant gaseous effluents are sampled isokinetically for the above mentioned monitors (RT 7325-1 and 2). Monitor RT 7325-1 is used in conjunction with RT 7325-2 and both are located in the turbine building access bay on the north wall above the deaerator tanks at elevation 4921 feet. Monitor RT 7325-2 is a G-M detector which provides a high range capability for the system. Both of these monitors sample the reactor building ventilation exhaust and are read out in the control room on a multipoint strip chart recorder. These monitors have control actions of shutting down the turbine building ventilation system and placing the control room ventilation system on recirculation, wherever the setpoints are reached. These monitors are tested monthly and calibrated quarterly according to the procedures in SR 5.8.cd-Q.

The reactor building ventilation exhaust stack monitors (Eberline stack monitor, RT 73437-1, 2, and 3) monitor the effluent from the reactor building ventilation for beta particulate and iodine-131 radioactive

contaminants. This monitoring system has isokinetic sampling with the same filters and collection efficiencies as previously dated for monitors RT 7325-1 and 2. It is comprised of two separate units. The detector and sampler unit located on EL 4912 feet of the turbine side and the readout unit is located in the control room. The system detectors are scintillation type detectors and their signals are sent to the readout unit in the control room where they are displayed. The readout consists of individual meter readouts, NORMAL, ALERT, and HIGH light indications, and a common chart recorder.

The licensee has the capability to remove, replace, and transport samples to the radiochemistry laboratory and meet the criteria of GDC-19 of 5 rem whole body and 75 rem dose equivalent to extremities during the duration of an accident. The procedure to perform this task is found in HPP-53. A shielding analysis study for transporting a loaded silver zeolite cartridge via a 2" thick lead pig determined that the unshielded cartridge has a contact dose equivalent rate of 20 mrem/h, and when contained in the pig the dose equivalent rate would be 1.3E-02 mrem/h at the surface.

The transported cartridges are analyzed in the radiochemistry laboratory outside the reactor building. A GeLi detector is used with a Canberra Series 80 Multichannel analyzer to determine the iodine content of the cartridge.

The vent stack airborne iodine concentration is continuously displayed, alarmed, and recorded in the control room. Two control room alarm functions are provided; the first being a trouble alarm on the iodine detector to indicate loss of background signal, loss of power, or an increased level of detected radiation above background but below the instrument setpoints, and the second being the high radiation alarm.

The NRC inspectors could not determine if any provisions had been made in the sampling systems to ensure that the adsorbers (zeolite cartridges) could not be degraded by entrained moisture in the effluent stream. The licensee had not considered this potential problem, therefore, this item is considered open (267/8221-05) pending the licensee's study of this item and the solving of the problem if any are found.

No violations or deviations were identified.

f. Item II.F.1, "Additional Accident Monitoring Instrumentation"

(1) Attachment 3, "Containment High-Range Radiation Monitor"

(a) Documents Reviewed

i. Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

ii. Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton

iii. Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

iv. Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

v. Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

vi. Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

vii. Letter, July 30, 1982 to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

viii. General Atomic Company Document Number C-70-002, "Calc-FSV Shielding Design Review for DBA-1.

ix. SR 5.4.9-A3, "Area and Equipment Monitors Calibration"

(b) Discussion

For this item (Item II.F.1, Attachment 3), NUREG-0737 stipulates that the containment high-range radiation monitoring system must provide two radiation monitors in containment.

It specifies that the monitors have a maximum range of E+08 rad/h which includes both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments, but gamma-sensitive instruments can be so qualified. In order to follow the

course of an accident, a containment monitor that measures only gamma radiation is adequate if it has upper range of E+07 R/h.

The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

The monitors are required to respond to gamma photons with energies as low as 60 keV, and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV. Monitors that use thick shielding to increase the upper range will underestimate post-accident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

The monitors must have the capability to detect and measure the radiation level within the reactor containment during and following an accident.

(c) Conclusions

Again, one must be reminded that the stipulation given for the high-range containment monitors are for light water reactors and FSV is a high temperature gas cooled reactor. The power density and fuel configuration are different for light water reactors and FSV. FSV's power density is lower and the fuel is encapsulated with a multilayered ceramic coating having a high temperature capability. This coating will delay the release of fission products after a reactor accident. Also, the prestressed concrete reactor vessel has a minimum thickness of nine feet. FSV does not have a containment building and the maximum gamma dose rate expected during a design basis accident is 1.4 rad/h in the reactor building. After 1000 hours into the accident, a maximum dose rate of 600 rad/h is expected from the main stack filters.

FSV is using the existing area radiation monitors (RT 93250, 93251, and 93252) to meet the requirements of NUREG-0737 containment high range radiation monitors.

These monitors are capable of reading up to 10 rad/h gamma dose rates. Although 10 rad/h is much less than E+07 rad/h specified for gamma radiation in NUREG-0737, FSV maintains that an appropriate radiation upper limit for their reactor building environment monitoring should be lower than that specified for light water reactors. An NRC letter (7.f.(1)(a) viii.) to FSV states that ORNL will determine the upper limits of the radiation level appropriate for the reactor building of FSV, and another letter (7.f.(1)(a) ix.) from FSV to the NRC states that a containment high radiation monitor is on order and should be installed by the end of 1982.

There are 17 area monitors located in the reactor building.

Each area monitor has a halogen-quenched G-M detector and an approximate sensitivity of 2 cps/mR/h. The monitors have a energy response of ±15% between 80 KeV and 2.5 MeV and a range of 0.1 mR/h to 10 R/h. The monitors are Gulf General Atomic Area and Equipment Monitor Detector Assemble RT-1. Each one has a local alarm (except for RT 93250-14) and the electronic equipment, recorders, and alarms are located in the control room.

The area monitors are tested on a weekly schedule and are calibrated quarterly according to SR 5.4.9-A3. They are source calibrated at 30-70 mR/h and 1.0 R/h. The area monitors are connected to essential power busses.

NUREG-0737 states that the containment high-range radiation monitor shall have the capability to detect and measure the radiation level within the reactor containment during and following an accident. The operating temperature limits are -58 to 167°F for the FSV area monitors. In the FSV FSAR update, Figures 1.4-1, 2, and 4 show temperatures for accident situations in the reactor building that are greater than 167°F for for periods of time up to 30 minutes. This would indicate that some of the area monitors would be inoperative under these conditions; therefore, these monitors would be unable to function properly continuously during an accident. This item is considered open (267/8221-07) until the licensee determines:

- During accident situations an adequate number of area monitors would be operating to determine radiation levels in the reactor building.

- Procedural changes and/or equipment modifications to be certain the accident could be "followed" by the area monitors even though more than one monitor is connected to an alarming annunciator.

- Installation of the ordered high range containment monitor.

No violations or deviations were identified.

g. Item III.D.3.3 "Improved Inplant Iodine Instrumentation Under Accident Conditions"

(1) Documents Reviewed

(a) Letter, June 15, 1979, to G. Kuzmyca (USNRC) from D. W. Warembourg (FSV)

(b) Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(c) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

(d) Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

(e) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(f) Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(g) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(h) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg

(i) Letter, December 30, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(j) Letter, August 6, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(k) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(l) Letter, October 22, 1981, to S. J. Ball (ORNL) from D. W. Warembourg (FSV)

(m) Memorandum, January 28, 1982, to file (Region IV) from T. F. Westerman (USNRC)

(n) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(o) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(p) FSV Health Physics Procedure - 12, "Portable Air Sample Collection and Analysis"

(q) General Atomic Company, Document No. C-70-002, "Calc-FSV Shielding Design Review for DBA-1.

(r) FSV Health Physics Procedure - 57, "Radiation and Airborne Radioactivity Monitoring During Abnormal Releases in the Plant"

(2) Discussion

The NUREG-0737 stipulates that each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

The effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

(a) The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

(b) Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

(c) Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

(d) The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

Each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

(3) Conclusions

For air sampling of radioiodines, the licensee uses a portable system weighing approximately 10 pounds that can be used in any area of the plant. This system includes a Radeco model H-809V air sampler with a Whatman GF/A filter and a Radeco silver zeolite "Radioiodine Sampler" model GY-130 cartridge, which has collection efficiency for iodine greater than 95 percent. These silver zeolite cartridges require no flushing with clean air or inert gases since they will not collect any of the noble fission gases.

The samples are collected for 5 minutes per procedures HPP-12 and -57, and taken to the radiochemistry laboratory for analysis on the multichannel analyzer with GeLi scintillation detectors previously described in this report. The analysis is performed according to procedure HPP-12.

The radiochemistry laboratory has a projected background dose rate of approximately 2 mrad/h from the reactor in an accident situation. The radiochemistry laboratory is on the ground level of the Technical Support Center which is outside of the reactor building and this complex has monitor RIT 7937 on the intake ventilation system. The high alarm setpoints on Monitor RIT 7937 are set to 3 E+04, 3 E+04, and 3 E+03 cpm for the gas, particulate, and iodine, respectively. These setpoints close the louvers routing the air through a filter system.

The NRC inspectors determined that the associated training for this item (Item III.D.3.3) could be improved to the extent that specific training for collection and analyzing of the iodine in emergency situations be given instead of relying upon the routine training in these areas. The added emphasis on the accident situation during specific training would be more beneficial. In addition to the routine training, the health physics and radiochemistry personnel participates in the two emergency drills annually where the necessary procedures are involved. The NRC inspectors inspected a sampling of the routine training and found it adequate.

The licensee also has two cart mounted iodine monitors (Eberline PING 1A) which has a single channel analyzer as part of each monitor. These monitors have very limited portability and are not easily moved to any position in the plant on a timely basis.

If needed, the licensee has a 2" lead pig, as previously mentioned in 7e.(3), to transport cartridges to the radiochemistry laboratory.

This item meets satisfactorily the intent of NUREG-0737 and should be considered closed.

No violations or deviations were identified.

h. Item II.D.3.4 "Control Room Habitability Requirements"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton (USNRC)

(c) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(d) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(e) Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

(f) Letter, August 26, 1981, to J. R. Miller (USNRC) from D.W. Warembourg (FSV)

(g) Memorandum, January 29, 1982, to File from T. F. Westerman (USNRC)

(h) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(i) Letter, June 1, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(j) Letter, June 10, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(k) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(l) 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants - 19, Control Room"

(m) Standard Review 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity"

(n) Standard Review Plan 2.2.3, "Evaluation of Potential Accidents"

(o) Standard Review Plan 6.4 "Habitability System"

(p) Regulatory Guide 1.78, "Assumptions for Evaluating

(q) Regulatory Guide 1.95, "Protection of Nuclear Power plant control room operators against an Accident Chlorine Release"

(r) General Atomic Company, Document No. C-70-002, "Calc-FSV Shielding Design Review for DBA-1.

(2) Discussion

In accordance with this item (NUREG-0743 Item III.D.3.4) and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shutdown under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50).

All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plans (SRP) sections. The new clarification specifies that licensees that meet the criteria of the

SRP's should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability, but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment, as described in Appendix A and B of Standard Review Plan Chapter 15.6.5.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems.

(3) Conclusions

In the various documents reviewed above, the licensee has made submittals to the NRC that provide a basis for their conclusion. In correspondence 19(a)(1)(g) and (h), it is implied that the licensee has met the requirements of this item, but a human factors study is needed. Also, correspondence 19(h)(1)(i) and (j) states that ORNL still has this item under review.

The NRC inspectors determined that the licensee's submittal addresses all the subjects entailed in this item (Item III.D.3.4) of NUREG-0737. Again, realizing that NUREG-0737 SRP 2.2.1-2.2.2, 2.2.3, 0.4, Regulatory Guides 1.78 and 1.95, respectively, for light water reactors and FSV is a high temperature gas-cooled reactor, it appears that the

licensee has met the intent of these requirements. Therefore, this item (III.D.3.3) is considered closed.

No violations or deviations were identified.

8. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. The unresolved item disclosed during this inspection is discussed in paragraph 6.c.

9. Exit Interview

The NRC inspectors met with the licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on September 3, 1982. The NRC inspectors summarized the scope and findings of the inspection.

G

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

REC'D MAR 3 1 1980
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Docket No. 50-267

Mr. J. K. Fuller, Vice President
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

Dear Mr. Fuller:

Enclosed is the staff's evaluation of the implementation of "Category A" Lessons Learned requirements (excluding 2.1.7a) at Fort St. Vrain. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on January 21 and 22, 1980.

Based on our evaluation, we conclude that the implementation of the "Category A" requirements at Fort St. Vrain is acceptable. Certain items, identified in the evaluation, will be verified by the Office of Inspection and Enforcement.

This evaluation does not address the Technical Specifications necessary to ensure the limiting conditions for operation and the long-term operability surveillance requirements for the systems modified during the "Category A" review. You should be considering the proposal of such Technical Specifications. We will be in communication with you on this item in the near future.

Sincerely,

Themis P. Speis

Themis P. Speis, Chief
Advanced Reactors Branch
Division of Project Management

Enclosure:
As stated

cc: D. Ross
D. Eisenhut

XA

8004040713

9pp.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# FORT ST. VRAIN

## EVALUATION OF "CATEGORY A" LESSONS LEARNED

## IMPLEMENTATION

### Introduction

By letters dated October 29, December 12, 18 and 28, 1979 and February 20, 1980, Public Service Company of Colorado (licensee) submitted documentation of the actions taken at Fort St. Vrain (the plant) to implement the requirements resulting from TMI-2 Lessons Learned. To facilitate our review of the licensee's action, members of the staff visited the plant on January 18 and 19, 1980.

### Evaluation

Each of the "Category A" requirements and the associated acceptance criteria are documented in NUREG-0578 and NRC letters dated September 13 and October 30, 1979. The number designation of each item in this evaluation is consistent with the identification used in NUREG-0578.

2.1.1 Emergency Power Supplies for Pressurizer Heaters, PORVs, Block Valves and Pressurizer Level Indication

Since the HTGR does not use a pressurizer or PORVs these requirements are not applicable.

2.1.2 Performance Testing for BWR and PWR Relief Valves

The licensee has committed to abide by the applicable recommendations of the EPRI Qualification Program. Any recommendations of the EPRI Program arising from either two phase flow or liquid flow through safety valves are considered by the licensee to be non-applicable in view of the design of the HTGR. We find this position to be acceptable.

2.1.3.a Direct Indication of PORV and Safety Valve Position

The Fort St. Vrain reactor does not use PORVs since it is a gas-cooled reactor and pressure changes due to system transients are relatively small. This reactor utilizes two code safety valves, however, they are separated from the reactor by normally locked open block valves and rupture discs. The pressure between the rupture disc and the safety valve is monitored by safety grade instrumentation and will alarm in the control room. Lifting of the safety relief valves will be indicated by the safety grade pressure alarm and also a high radiation alarm in the discharge piping. Should a safety valve fail to reseat after actuation the operator would be able to determine this by a continually decreasing pressure in the PCRV as indicated by three safety grade pressure instruments that read out in the control room. Plant Technical Specifications

do not permit the operators to close the block valves. Furthermore, the maintenance of primary coolant inventory and pressure is not essential to cooling down the reactor and can result in more severe consequences.

Based on the above, we have determined that the existing safety valve position indication instrumentation adequately meets this requirement.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling

The requirement for the installation of indication that would apprise the operator of the margin to saturation of the primary coolant or primary coolant level in the reactor vessel are not applicable to the Fort St. Vrain reactor. This reactor design utilizes helium gas instead of water as primary coolant. The critical temperature of helium (-268°C) is such that the primary coolant is always single phase. Instrumentation presently available to detect inadequate core cooling consists of helium circulator speed, reactor differential pressure, core outlet thermocouples, ratio of core power to helium flow, and differential pressure across the helium circulators. It should be noted that even though the above instrumentation exists to determine inadequate core cooling the limiting DBE for which the plant was analyzed was the loss of all core cooling, primary and secondary. The consequences of this accident as indicated in the FSAR show that upon depressurization, heat from the core will be transferred to the PCRV. The PCRV is cooled by redundant safety grade cooling systems to preserve its integrity. Since direct core cooling is not necessary as indicated by the FSAR analysis; we have determined that the licensee does not need to provide any additional instrumentation to detect inadequate core cooling and, therefore, satisfies this requirement.

2.1.4 Containment Isolation

The NRC requirements are that the licensee is to: (a) carefully reconsider the determination of which system should be considered essential or non-essential for safety; (b) modify systems as necessary, to isolate all non-essential systems by automatic, diverse, safety-grade isolation signals; and (c) modify systems as necessary, to assure that the resetting of the containment signals does not cause the inadvertent re-opening of containment isolation valves.

The Fort St. Vrain gas cooled reactor design, the containment design, and the design of systems associated with accident mitigation and plant shutdown are such that these requirements cannot be directly applied to the Fort St. Vrain plant. The plant does not include a conventional containment building. The Fort St. Vrain primary coolant system is completely contained within the PCRV. Secondary closures on the PCRV penetrations and the PCRV concrete structure constitute a secondary containment. Furthermore, the PCRV and the reactor plant associated systems are located within a reactor building. This building provides vented, filtered tertiary confinement. The licensee has addressed the NRC containment isolation requirements considering the differences in the Fort St. Vrain design from light water reactor plant designs.

The licensee's February 20, 1980 submittal included a table of the essential and non-essential containment penetrations. The "essential" system included those systems required to perform an active role in various safe shutdown functions. The bases for the classification of these systems was provided in that submittal.

The Fort St. Vrain design is such that non-essential systems are either isolated automatically by isolation signals, closed, or contained within a tertiary containment. As discussed in the licensee's February 20, 1980 submittal, automatic isolation is initiated by diverse containment isolation signals including external radiation detection, high pressure, or high flow rates as appropriate to the individual system purpose and design. A few systems do not close on diverse signals, these systems close automatically on a single isolation signal. An important difference in the Fort St. Vrain gas cooled reactor design versus a light water reactor is the relatively long time for accident conditions and core damage to develop. Thus, a significantly greater time (i.e., several hours) is available to perform manual operations to isolate the "containment". Considering the time available to take manual action we find this design acceptable.

The licensee has reviewed their isolation control circuitry with regard to the resetting of the isolation signal and the potential for automatic loss of containment isolation as discussed in their December 12, 1979 submittal.

To prevent inadvertent reopening, the system design utilizes three position spring-return-to-neutral switches. In addition, each valve has "seal-in" relays which maintain the valve in the closed position following isolation reset. Therefore, the operator must deliberately turn each individual hand switch to the "open" position after the isolation signal is reset.

The "containment" isolation design for the Fort St. Vrain plant has been reviewed considering the unique features of a gas cooled reactor and the isolation problems identified in NUREG-0578 Section 2.1.4. We conclude that the requirements of Item 2.1.4 have been properly addressed and that the Fort St. Vrain "containment" isolation design is acceptable.

2.1.5.a & c Dedicated Penetrations for External Recombiners or Post-Accident External Purge System and Recombiner Procedures

The NRC's position is that dedicated containment isolation systems should be used for the external recombiners or purge systems that meet redundancy and single failure requirements and that the procedures for use of the recombiners be reviewed considering shielding requirements and personnel exposure limitations.

These requirements do not apply to the licensee since their design does not include requirements for recombiners or purge systems for post-accident combustible gas control of the containment atmosphere. The Fort St. Vrain

reactor incorporates a ceramic core cooled by an inert gas. This is in contrast to Zirconium clad water cooled light water reactors. Loss of coolant accident conditions involving Zirconium water reactions and the disassociation of water are the significant sources of combustible gases in light water reactors and there is no comparable source of hydrogen in the HTGR. Therefore, this requirement is not applicable.

2.1.6.a Systems Integrity

The entire primary system of the Fort St. Vrain reactor is contained in the PCRV which is inside the reactor building. The only system which processes primary coolant is the helium purification system, most of which is also in the PCRV. The hydrogen removal and regeneration equipment and primary coolant sampling lines are located outside the PCRV within the reactor building. Leakage from this system would be collected and discharged through filters by the reactor building ventilation system. Leakage into the reactor building will be detected by various area and process radiation monitors located throughout the building.

In addition, the licensee has stated that the total helium inventory is determined and leakage calculated daily and that unanticipated departures from established leakage rates are investigated and corrected on an as-needed basis.

Therefore, we conclude the licensee has met the requirements of Item 2.1.6.a as they apply to his system.

2.1.6.b Plant Shielding Review

The licensee's December 28, 1979 submittal includes a design review of plant shielding and environmental qualification of equipment for the worst design basis accident. The design review was performed assuming the source term as specified in the October 30, 1979 letter is uniformly distributed throughout the free space of the reactor building or the PCRV. The licensee has identified vital areas which will require further shielding in order to assure necessary functions can be performed. The licensee has also evaluated operator actions which may be required and determined further shielding or design modifications are not necessary. The licensee has evaluated the adequacy of equipment and instrumentation and determined that the radiation levels pose no hazard to this operation.

A detailed evaluation of the licensee's submittal will be performed at a later date. We conclude that the licensee has met the "Category A" requirements for this item.

2.1.7.b Auxiliary Feedwater Flow Indication to the Steam Generators

The Fort St. Vrain reactor has two feedwater headers each of which supplies six steam generator modules. Auxiliary feedwater flow from any of the various sources passes through these feedwater headers. Each of the feedwater headers has two safety grade flow transmitters which record and indicate in the control room. In addition, the flow to each of the six steam generator modules on each header has safety grade flow instrumentation that feeds to a multipoint recorder in the control room. We conclude that the licensee meets the requirements of NUREG-0578.

2.1.8.a Post-Accident Sampling

The licensee's December 28, 1979 submittal contains a design review of the plant sampling capability for primary coolant and containment air samples assuming a source as speci ied in NUREG-0578. The licensee has concluded that samples can be obtained throughout the accident without incurring excessive personnel radiation exposures. Therefore, no modifications to the existing sampling station are necessary.

The licensee has incorporated minor modifications into the sampling procedure to assure sampling of primary coolant can be accomplished throughout an accident.

The licensee has indicated that a new radiochemical analysis facility will be located in a concrete building to be erected adjacent to the reactor building. The new facility will contain all existing analysis equipment and will have appropriate ventilation and waste disposal facilities. This will assure the capability to provide onsite analysis of samples following an accident. The licensee has incorporated procedures for analyzing samples onsite and has the capability to ship the samples offsite if the existing analysis facility becomes uninhabitable.

Based on the above information, we conclude the licensee has met the "Category A" requirements for this item.

2.1.8.b High Range Radiation Monitors

The licensee has implemented interim procedures and installed portable equipment for the quantification of noble gas effluents released from the stack as a result of an accident.

The licensee currently has the capability to continuously monitor gaseous iodine releases from the reactor building exhaust by way of a detector which monitors a charcoal cartridge. The monitor has a remote readout in the control room. The licensee also has procedures in effect for removing the cartridge to the analysis facility for spectroscopic analysis. The licensee has incorporated procedures for estimating particulate releases in plant effluents.

The licensee has not incorporated interim procedures for monitoring of steam dump and relief valves. However, the only potential for primary to secondary leakage exists in the reheat loop of the steam system. This loop is monitored upstream of the steam dump valves and will automatically isolate the steam generator in the event of primary to secondary system leakage. Therefore, monitoring of the steam dump valves is not necessary at Fort St. Vrain.

Based on the above, we conclude that the licensee has met the Category "A" requirements for this item.

2.1.8.c Improved Iodine Instrumentation

The licensee has indicated that portable air samples will be taken utilizing charcoal absorbers which will be counted using a multi-channel

analyzer. The licensee has located the analyzer in a low background area to permit counting of the samples. The licensee has developed procedures for obtaining the air samples and has identified those areas requiring continuous habitability. The licensee has indicated that the samples can be analyzed in ten minutes which will allow adequate time for protective actions. The licensee has stated that all procedures will be in place prior to resumption of power operation. Our Office of Inspection and Enforcement will verify that the procedures are in effect.

Based on the above, we conclude that the licensee has met the requirements for this item.

2.2.1.a Shift Supervisor Responsibilities

The NRC requirement for this item is to revise, as necessary, the responsibilities of the Shift Supervisor (SS) such that he can provide command oversight of operations and perform management review of ongoing operations that are important to safety.

During the staff's site visit we reviewed the licensee management directives and administrative procedures associated with this position. We have determined that these directives and procedures, along with the modifications noted in the licensee's February 20, 1980 submittal satisfy the requirements of NUREG-0578 Item 2.2.1.a for delineation of SS responsibilities.

2.2.1.b Shift Technical Advisor (STA)

The NRC requirement is for the licensee to provide an on-shift advisor to the SS to serve the two functions of accident assessment and operating experience assessment. As a supplement to the operating staff, the STA must be available to the control room to assist in diagnosing an off-normal event.

The program that the licensee has implemented to satisfy the STA accident assessment function utilizes three engineers who are placed on-call to respond to potential accident conditions at the plant. The licensee has committed to a one-hour response time for the on-call STA, in contrast to the staff's position for a 10 minute response time. We have considered the licensee's argument for the relatively long times (i.e., hours) for accident conditions to develop in a gas cooled reactor versus a significantly shorter time for light water reactors. Considering this unique feature of a gas cooled reactor the staff finds the one-hour response time acceptable for Fort St. Vrain. The three STA engineers will also fulfill the required operating experience assessment function required by NUREG-0578.

We have reviewed the licensee's October 29 and December 12, 1979 submittals describing their STA program. In addition, during the site visit we discussed the program with the licensee and determined that a satisfactory STA program is in operation. We find that their STA program satisfies the staff's requirements described in Section 2.2.1.b of NUREG-0578 and is therefore acceptable.

2.2.1.c Shift and Relief Turnover Procedures

The NRC requirement is for the licensee to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the oncoming shift.

The licensee conducted a review of their turnover procedures, as discussed in their December 12, 1979 submittal. They determined that for the most part their existing procedures and logs contained the required information regarding critical plant parameters, availability of essential systems and limiting conditions of operation. Several modifications discussed in their February 20, 1980 submittal were made to provide better continuous monitoring of the conditions of all facility systems.

Further, their Q/A surveillance and audit program provides a routine evaluation of effectiveness of the shift turnover procedures.

During our site visit we discussed the shift turnover procedures with the licensee. We conclude that the licensee has satisfied the requirements of Item 2.2.1.c related to shift turnover procedures. Adequacy of the checklists and logs will be performed by the Office of Inspection and Enforcement and will be documented in appropriate Inspection Reports.

2.2.2.a Control Room Access

The licensee has amended their procedures to authorize the Shift Supervisor, the Superintendent of Operations or plant management to restrict access to the control room. Emergency procedures have been revised to establish lines of authority and responsibility in emergency situations. The licensee has satisfactorily implemented this requirement.

2.2.2.b Onsite Technical Support Center (OTSC)

The OTSC proposed by the licensee to meet the Category A requirements of NUREG-0578 will be located adjacent to the control room. The OTSC is part of the control room complex and as such its atmosphere is controlled by the same ventilation system as the control room and has the same shielding as the control room. The licensee has provided dedicated communications in the OTSC to the NRC, the control room, and the Emergency Operations Facility. In addition, the licensee has provided plant technical data in the OTSC. This data includes P&I diagrams, single line electrical schematics, FSAR, Technical Specifications, and Emergency Procedures. The licensee has committed to install a closed circuit TV system in the control room to transmit plant parameters to the OTSC. This closed circuit TV system will be installed on an expedited basis and should be operational in approximately 4 to 5 weeks. In the interim, the licensee can utilize the designated Technical Advisors in the control room and the OTSC to relay plant parameters. We find that the licensee has implemented this requirement in an acceptable manner.

2.2.2.c Operational Support Center (OSC)

The licensee has designated an emergency station to which operational and support personnel report in the event of an emergency. This station is reflected in the Emergency Procedures and has communication with the control room. We find this acceptable in meeting the requirement for establishment of an OSC.

NRR Reactor Coolant System Venting

Since Fort St. Vrain is a gas cooled reactor, this requirement is not applicable.

Conclusion

Based on the above, subject to our Office of Inspection and Enforcement verification as noted, we find that implementation of the Category "A" Lessons Learned requirements at Fort St. Vrain is acceptable.

Dated: March 20, 1980

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Y-81257

DEC 23 1981

Docket No. 50-267

LICENSEE: Public Service Company of Colorado

FACILITY: Ft. St. Vrain

SUBJECT: SUMMARY OF MEETING HELD ON DECEMBER 3, 1981.

On December 3, 1981 a meeting was held at the site of the Fort St. Vrain reactor among representatives of Public Service Company of Colorado and the Nuclear Regulatory Commission. The object of the meeting was to discuss the results of an Oak Ridge National Laboratory report on the applicability of NUREG-0737 requirements to Fort St. Vrain. The list of attendees is included as Enclosure 1.

The following is an itemized listing of the requirements of NUREG-0737 and their resolution as discussed at the meeting.

I.A.1.1 Shift Technical Advisor.

PSC has a training program for STAs that includes the major equipment of plant systems. Also implementing porcedures are in place governing STA presence in the plant and contingencies. Implementation of technical specifications as per PSC comments (P-81263) is acceptable. Use of accident simulation codes by STA in analyzing plant transients and postulated accidents is acceptable. The PSC proposal for keeping the STA position (long term) and upgrading SROs, but using college level expertise as nonshift assistance is acceptable.

I.A.1.3 Shift Manning.

As per previous I&E RIV SER, the 16 consecutive day cycle for operators is acceptable. Table B.1 will be addressed in the January review by Emergency Planning Licensing Branch. PSC will generate a new Table B.1 that will be applicable for FSV with appropriate times and supporting studies. A specific request will be made by PSC to have the H.P. technician cover two functions.

I.A.2.3 Administration of training

PSC is in compliance.

I.A.3.1 Simulators Exams.

Paul Collins will hold a meeting with PSC to resolve this issue. 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. PSC provides training in accident analyses and behavior during transients and will, in the future, provide more hands-on experience and accident simulation codes.

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I.C.1 Accident and Procedures Review.

PSC issued a set of Emergency Procedures on November 16, 1981. These procedures will be reviewed to determine their completeness and comprehensiveness to the plant operators.

I.C.5 Feedback of Operating Experience.

As per SER written by I&E, R IV, PSC is in compliance.

I.C.6 Verify Correct Performance of Operating Activities.

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 agrees and will provide input to Region IV.

I.D.2 Plant Safety Parameter Display Console.

PSC is reviewing the recommendation made by ORNL and will continue their dialog for proper resolution.

II.B.2 and II.B.3 Plant Shiedling 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.

II.B.4 Training for Mitigating Core Damage.

ORNL recommended several items that might be useful for severe accident mitigation and control at FSV. PSC will review the items 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 Relief and Safety-Valve Test Requirements. Valve Position Indication.

PSC will rely upon the EPRI qualification testing program recommendations as they may apply to FSV.

II.E.1.1 and II.E.1.2 Auxiliary Feedwater Systems Evaluation. Auxiliary Feedwater System Initiation and Flow.

These two items are not applicable to Fort St. Vrain.

II.E.4.2 Containment Isolation Dependability.

PSC is in compliance.

II.F.1 Accident Monitoring.

The upper limits of $10^5$ MC/cc and $10^8$ rad/hr cannot be met with existing instrumentation. 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.

PSC is in compliance.

II.K.3.17 ECCS System Outages.

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

II.K.3.18 ADS Actuation. Not Applicable.

II.K.3.30 SB LOCA Methods. Not Applicable.

II.K.3.31 Compliance with 10 CFR.50.46. Not Applicable.

II.D.1.1 Primary Coolant Outside Containment.

PSC is in compliance

II.D.3.3 Inplant Radiation Monitoring.

PSC is in compliance.

III.D.3.4 Control Room Habitability.

PSC response will be evaluated from a human factors viewpoin Most other aspects have been incorporated by PSC.

III.A.2 Early Warning Alert System.

Exemption from the 15 minute public notification has been denied. This was a general action and was published in the Federal Register Vol. 46, #182, September 21, 1981 page 46587. PSC will evaluate the option of formal appeal.

I.A.2.1 We recommend 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 STA to be on callrather than on shift, the requirements for college level equivalent training for shift personnel be waived.

George Kuzmycz

George Kuzmycz, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. List of Attendees
cc: See next page

ENCLOSURE 1

MEETING ATTENDEES

NRC

T. M. Novak, AD/OR/DL
. A. Clark ORB#3/DL
G. Kuzmycz ORB#3/DL
D. M. Rohrer, IE/DEP/EPLB
T. F. Westerman, IE/R IV
M. W. Dickerson, Sr. Resident Inspector
G. L. Plumlee, Resident Inspector

ORNL

S. J. Ball
R. M. Harrington

PSC

| | |
|---|---|
| Don Warenbourg, Mgr - | Nuclear Production |
| Ed Hill | FSV Station Manager |
| H. L. Brey, Mgr - | Nuclear Engineering |
| J. R. Reesy, Mgr - | Nuclear Design |
| L. M. McBride, Mgr - | Tech/Admin. Services |
| Ted Borst, Mgr - | Radiation Protection |
| W. Franek, Mgr - | Nuclear Site Engineering |
| M. H. Holmes | Nuclear Engineering |
| C. Fuller | Tech. Services Engineering |
| J. M. Sills | " " |
| Ms E. Niehoff | Nuclear Projects |

Ft. St. Vrain
cc list

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Senior Vice President
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Mr. W. Dickerson
NRC Resident Inspector
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Director, Division of Planning
Department of Local Affairs
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Denver, Colorado 80203

Chairman, Board of County Commissioners
of Weld County, Colorado
Greeley, Colorado 80631

Regional Representative, Radiation Programs
Environmental Protection Agency
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Denver, Colorado 80203

Mr. Don Warembourg
Nuclear Production Manager
Public Service Company of Colorado
16805 Weld County Road 19 1/2
Platteville, Colorado 80651

MEETING SUMMARY DISTRIBUTION

Licensee: Public Service Company of Colorado

*Copies also sent to those people on service (cc) list for subject plant(s).

Docket File
NRC PDR
L PDR
NSIC
TERA
ORB#3 Rdg
JOlshinski
JHeltemes, AEOD
BGrimes
RClark
Project Manager
Licensing Assistant
ACRS (10)
Mtg Summary Dist.
NRC Participants

0737

Public Service Company of Colorado

December 20, 1980
Fort St. Vrain
Unit No. 1
P-80438

Mr. Darrel G. Eisenhut, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Fort St. Vrain Unit No. 1 TMI Action Plan Requirements NUREG 0737

REFERENCE: NRC Letter Dated 10-31-80

Dear Mr. Eisenhut:

We have completed our review of the subject NUREG transmitted by the above referenced letter. The attachment contains our response to each of the action requirements that are applicable to Fort St. Vrain. Our response to the various action requirements generally falls into four (4) categories.

1. Those action requirements for which we have provided previous response which we feel is still applicable in light of the clarification provided by NUREG 0737. Other than previous commitments that may have been made as a part of our response, we do not plan on any further action.

2. Those action requirements and schedules which for various reasons we will be unable to meet, or which for various reasons we have taken exception as to the applicability of the requirements to gas cooled technology as opposed to water cooled technology.

3. Those action requirements and schedules which we intend to meet.

4. Those action requirements which are clearly not applicable to Fort St. Vrain.

As we have pointed out in previous correspondence we have had a difficult time applying the criteria, guidance and requirements to Fort St. Vrain, and in many cases have had little if any guidance that was clearly applicable to gas cooled technology. In addition, we were consistently excluded from receipt of various letters, bulletins, and orders resulting from the TMI action requirements, and in this respect, we find that we were not afforded the same time schedule to plan and complete various activities by comparison to the water reactors.

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We met with the Special Projects Division along with other members of the staff on December 10, 1980, in an attempt to obtain some further clarification of the applicability of many of the action requirements to gas cooled technology. While we were able to obtain some general clarification we found that we were unable to obtain any clarification with reference to the technical differences involving gas cooled reactors versus water cooled reactors which is a continuing problem that has been with us since the onset of TMI, and which because of a lack of guidance, has impeded our progress in many area..

We have continued in our efforts to justify certain exceptions from the various criteria on the basis of distinct differences between a gas cooled reactor and a water cooled reactor. We believe that adequate technical justification has been provided in many areas, but it is obvious that the technical justification provided is not being considered in the various staff reviews, and it is also obvious that various technical justifications which were reviewed by one group in the past are not being considered as new review groups are formed. As a result we appear to be in a continual education process and in our opinion, continue to be penalized with the inapplicable criteria.

Very truly yours,

Don W. Warembourg

Don W. Warembourg
Manager, Nuclear Production
Fort St. Vrain Nuclear
Generating Station

DWW/alk

Attachment

ATTACHMENT 4
REFERENCE ITEM III.D.3.4

The control room emergency zone at Fort St. Vrain consists of the control room and adjacent areas such as the kitchen and washroom.

The control room under normal conditions is staffed with two (2) people at all times. Various other people are entering and leaving as required for plant operations.

During an emergency, the Personnel Emergency Response Plan (Administrative Procedure G-5) requires five (5) people to be in the control room full time. Additionally, a health physics person is assigned to the control room while four (4) other Operations Department Personnel will be in and out as required. During an emergency, which requires the personnel to use the Breathable Air System, the control room occupancy is limited to six (6) to match the number of air supply connections. Other personnel will be requested to use Scott Air Pacs if they are in the control room.

In addition to the protection offered by self-contained Scott Air Pacs and the breathable Air System, the control room ventilation system utilizes the Control Room Makeup Ventilation Filter (F-7502) which is of the CBR type and is rated at 1500 cfm but operates at a flow of 480 cfm. The filter consists of a particulate filter in series with a gas absorber containing activated charcoal. The filter is designed to meet all the requirements of the U.S. Army Chemical Corps Specification MIL-F-50052. In addition, Control Room Ventilation Filter (F-7503) has a filter efficiency of 45% by the NBS atmosphere dust spot test, and is rated at 21,160 cfm. The filter is equipped with an upstream prefilter to trap large particles. In conjunction with filters F-7502 and F-7503, Control Room Charcoal Filter (F-7504) has elements with a nominal 1" thickness and is rated at 21,160 cfm. Particulate matter is removed by ventilation filter (F-7503) before passing through the charcoal. Figure 1 (attached) is a schematic of the Control Room Ventilation System.

The following is a detailed comparison of existing FSV conditions to Standard Review Plans 2.2.1, 2.2.2, 2.2.3 and 6.4:

A. SRP's 2.2.1-2.2.2 and 2.2.3 - Hazard Identification

1. Guidelines

These SRP's address the identification of potential hazards and accidents within 5 miles of the plant.

ATTACHMENT 4

REFERENCE ITEM III.D.3.4

2. FSV Existing Condition

On site: Chlorine is stored in liquid form in 1-ton bottles outside the Chemical Building, about 360 feet from the Control Room. Also, various chemicals such as 29% concentrated ammonia, 93% sulfuric acid, and 50% caustic are stored in the demineralizer room on the ground floor of the turbine building. There is an indoor turbine lube oil storage tank and outdoor underground storage tanks for gasoline, diesel fuel, and No. 2 fuel oil; these could produce hazardous combustion products if they were ignited. In addition, there is Halon and $CO_2$ for the fire protection systems.

Within 5 miles: There is a Union Pacific RR track about 3 miles east that is the main north-south line between Denver, Colorado and Cheyenne, Wyoming; it carries LP Gas and occasionally liquid chlorine. Another tract 3/4 mil West of FSV carries mostly coal. Also, there are two oil lines, one 3.1 miles and one 4.7 miles from FSV, and a 4" to 6" medium pressure (140-150 psi) natural gas transmission line about 3/4 mile south of FSV. There are numerous anhydrous ammonia tanks used for fertilizer storage on adjacent farms, but there are no industrial activities that use chemicals or toxic materials.

3. Comments

a. As will be discussed with the specific guidelines, chlorine storage and the proximity to the railroad tracks are in accordance with Regulatory Guide 1.95.

b. Chemicals are properly stored at the Plant inside closed systems in a room with outside vents two floors below the Control Room. The Control Room ventilation intake is 60' above the elevation of the demineralizer (chemical storage) room vent; with ammonia's toxicity level of 100 ppm and its acrid smell that is detectable at a much lower level, it is concluded that the intake dampers could be closed and the respirators donned before personnel injury.

c. The fire protection systems are designed to minimize fires in the petroleum tanks and to alert personnel so that breathing apparatus can be used, if necessary.

d. The oil and gas line hazards due to explosions and fires are far enough away that there can be adequate warning to control ventilation as required.

B. SRP 5.4 - Control Room Habitability

1. Breathing Apparatus

a. Guideline

Paragraph 6.4.II.4 states that self-contained breathing apparatus for an emergency team (at least 5 men) should be on hand in the Control Room. Also, a six-hour on site bottled air supply, 30 man-hours, should be available with unlimited off-site replenishment capability from nearby locations.

b. Existing FSV Condition

There are 6 Scott Air Pacs in or immediately outside the Control Room, with 12 spare air bottles. There is also a Breathable Air system with 2 independent compressors and purifiers, each of which can provide 20 scfm to 5 masks in the Control Room. This system will remove chlorine and other noxious gases. There is also a 1140 scf, 2400 psig storage volume that can recharge 2 Scott Air Pacs and supply 5 respirators for 45 minutes without recharging. This is about 24 man-hours of available air, in addition to which there is about 10 man-hours of air in reserve air pacs located in the rest of the plant.

It is noted here that the Breathable Air System compressors have a suction point about 8' above grade. This keeps out dust and minimizes the amount of heavy, dense gases (like chlorine) that get drawn in.

The filter canisters of the Breathable Air Compressors are rated for 40,000 $ft^3$ of air, minimum. At the normal 20 CFM, each set of canisters could filter for at least 33 hours and in a dry environment, 40 hours could be expected. FSV monitors the compressor elapsed time meters to insure that there is sufficient remaining capacity to handle accidents and replacement cartridges are available locally.

2. Emergency Team Support

a. Guideline

Paragraph 6.4.II.2 states that food, water and medical supplies should be sufficient to maintain the emergency team for 5 days.

b. Existing FSV Condition

FSV does not have this material stored. For the FSV facility all analyzed accidents are of short duration so that these supplies will not be required. In the event of a long term accident, these materials are available nearby and could be obtained as required.

C. Regulatory Guides 1.95 and 1.78 are referenced by SRP 6.4 and they provide the following:

1. Regulatory Guide 1.95 postulates two basic chlorine accident types: a long-term low-leakage-rate release, or a short-term puff release. For the first type, only breathing apparatus is necessary to protect the control room operator, if he is given warning. For the second type, the control room should be automatically isolated. For the low-leakage-rate accident, FSV has adequate breathing apparatus (see B.1 above) and is installing a chlorine leak detector at the chlorine storage facility. Although this leak detector is inside the builidng while the chlorine storage bottles are outside, most of the connections are inside so most slow leaks will be detected. The puff release would most likely occur during loading and unloading the cylinders, which occurs about 360 feet from the Control Room ventilation intakes. There is not a direct path between the chlorine bottle storage area and the Control Room intake, chlorine gas is heavy and would have to rise 75' to the Control Room air intakes, and significant diffusion would take place over this distance. For these reasons, the puff release is not considered to be a significant Control Room hazard. FSV meets the guidelines as discussed below.

2. Material Storage

a. Guidelines

Liquified chlorine should not be stored within 100 meters of a control room or its fresh air inlets. Also, the largest container should have an inventory of 2000 lbs, and there should be a capability for manual isolation of the ventilation system. For large quantities as would be in RR tankers, they should be over 2000 meters (6560') away. Specific criteria is not provided for other substances.

ATTACHMENT 4

REFERENCE ITEM III.D.3.4

It is noted that the design concept for the Control Room ventilation system includes a full flow activated carbon filter that is normally bypassed but would be put on line when the inlet dampers are closed. Also, the makeup filter includes an activated carbon section that is rated for chemical, biological and radiological service. The full flow carbon filter has the capacity to absorb about 20 pounds of chlorine or about 100 $ft^3$ of pure chlorine gas at STP. Since the filter is not placed in service until after an accident, this capacity is considered adequate for cleanup of the initial concentrations of chlorine in the control room that entered before the area could be isolated.

4. Breathing System Assurance Level

a. Guidelines

The emergency air supply should meet single failure criteria and be Seismic Category I. For self-contained apparatus, there should be one extra unit for every three required.

b. Existing FSV Conditions

The Breathable Air System has two compressor/purifier trains, and it was designed and installed to Class I requirements. There are six Scott Air Pacs installed at the Control Room where, for five men, there should be seven. There are other units in the plant, so this is not considered a deficiency.

5. Emergency Procedures

a. Guidelines

Emergency procedures to be initiated in the event of a hazardous chemical release should be written. Also, the Control Room leakage characteristics should be periodically verified.

b. Existing FSV Conditions

FSV procedure G-5 covers Personnel Emergency Responses to various accidents, including chemical spills. This procedure essentially designates an emergency coordinator who will provide direction in the event of a hazardous chemical release. There is no periodic control room leakage test program. However, the amount of leakage is not considered critical to habitability because of the breathable air system and because of the charcoal filters on the ventilation system.

b. Existing FSV Condition

Liquid chlorine is stored in one-ton containers approximately 130 meters from the control room or its air intakes. All air inlet and outlet dampers can be closed with switch HS75184. Per the Union Pacific Traffic Agent, the RR track that carries liquid chlorine tankers is about 3 miles from FSV; the RR track 3/4 miles West of FSV carries mosty coal and miscellaneous freight but no chlorine.

3. Automatic Isolation & Ventilation System Design

a. Guidelines

The control room should be protected by quick response chlorine detectors located in the fresh air inlets that will automatically close the ventilation dampers. Also, the normal fresh air makeup rate should be less than 0.3 air change per hour and the fresh air inlet should be at least 15 meters above grade. Finally, the room should be of low leakage construction, with an equivalent exchange rate of less than 0.06 $hr^{-1}$, and low leakage dampers should be located upstream of recirculation fans or at other negative pressure locations.

b. Existing FSV Conditions

There are no chlorine detectors in the Control Room Ventilation System fresh air inlets; however, chlorine and other toxic materials at FSV have a strong odor that can be detected before they build up to toxic concentrations. Chlorine is toxic at about 15 ppm, and can be smelled before 5 ppm. With the ventilation system bringing 11,400 cfm of makeup into a 40,000 cubic foot control room, it would take over three minutes to replace all the clean air with chlorinated air. Since the dampers can be isolated in 5 seconds and the respirators can be donned in less than 2 minutes, it is concluded that there would be adequate time to manually isolate the ventilation system, don respirators, and switch the ventilation system to recirculate air through charcoal filters, so that plant control would not suffer. With the control room vent inlet located 75' above grade (22 meters), it is hard to envision an accident that would introduce highly toxic chlorine concentrations into the Control Room.

Also, the fresh air makeup rate with outside dampers closed is .39 $hr^{-1}$, the leakage rate is .09 $hr^{-1}$, and the dampers are bubble tight with an 8" water differential. These flow rates are slightly greater than recommended but meet the intent of the recommendations.

CONTROL ROOM HVAC SYSTEM - MODIFIED DESIGN

ATTACHMENT 4

REFERENCE ITEM III.D.3.4

FLOW SHOWN IN CFM FOR NORMAL MODE

FIGURE 1