DRAFT
For Comments

S. J. Ball
6/8/81

EVALUATION OF TMI ACTION PLAN APPLICATION TO FORT ST. VRAIN

Requirement I.D.2: Plant Safety Parameter Display Console

I. Background

The new NRC requirements for a plant safety parameter display system (SPDS) are outlined as part of the Control Room Design task, I.D, in NUREG-0660 (May 1980). The objective of this task is to improve the ability of the control room operators to prevent and cope with an accident by improving the information provided to them. In conjunction with a design review and upgrade of the control room, all licensees are required to design and install a SPDS which will:

> "display to operating personnel a minimum set of parameters (safety state vector) which defines the safety status of the plant. The system should have the capability of displaying a full range of important plant parameters and data trends on demand. In addition, the system should provide indication of when process limits are being approached or exceeded."

The licensees were originally required to submit a system design for NRR review by January 1981; however, since the action plan clarification (NUREG-0737) was not issued until November 1980, the deadline for the design (for FSV) was reset to July 1, 1981.

The NRC requirements for the SPDS were issued as part of NUREG-0696 in November 1980. The plant functions requiring indication on the SPDS are, at a minimum, reactivity control, reactor core cooling, reactor coolant system integrity, radioactivity containment, and containment activity. The SPDS is to be located in the main control room and have additional displays in the technical support center (TSC) and the emergency operations facility (EOF). The required design unavailability goal of the SPDS is 0.001, or 9 hrs/year.

II. Present Status of Implementation at FSV

Per a June 4, 1981 conversation with the PSC licensing engineer in charge of the emergency response facility implementation, PSC plans to submit a conceptual design for the SPDS to NRC by July 1, 1981. The conceptual design will specify what parameters will be read out on the SPDS, and propose means for presenting the operator with trend data and digested information.

III. Evaluation of Applicability of the SPDS to FSV

The design of a "satisfactory" SPDS would be dependent on reactor type (PWR, BWR, or HTGR), and except for minor details, a design that was satisfactory for one plant would probably be suitable for all reactors of that same type. Thus FSV is at a disadvantage in that the enitre HTGR SPDS development burden would fall on the one plant, while PWR and BWR owners could pool their resources. On the other hand, plant-type standardization efforts might turn out to be more costly and time consuming than individual efforts.

The objectives and the functional requirements for the SPDS for FSV (and others) are clearly vague and therefore allow the licensees considerable latitude in their designs. The objective would be to provide the operators with safety-related information not readily accessible on the main control panels. Such information could include summaries of critical plant parameters, notification of certain combinations of conditions diagnosed to be significant or potentially dangerous, trend information not apparent from control panel observations, and more sophisticated computations that could indicate potential problems (such as mass, heat, and reactivity balances). Because of the relatively slow response times of many HTGR parameters, diagnostic algorithms would also have to account for the dynamics of the plant.

The NRC guidelines also appear to be contradictory in that the SPDS parameter set is to be *minimized* and at the same time be capable of determining the *overall status* of the plant. Considering the large number of vital components and subsystems in FSV, it is possible that an SPDS that was poorly conceived with respect to a given accident could be distracting or misleading to an operator who could otherwise be getting a more complete picture of the situation from the larger and more detailed main control panels. A significant distinction between HTGRs and LWRs with respect to the SPDS is that due to the inherently slower response of the HTGR, its operators would have much more time during an accident sequence to properly assess the plant conditions from the more complex and complete main control panels. Consequently, safety parameter monitoring in FSV equivalent to that in a PWR or BWR could be effected with a less detailed SPDS.

The design unavailability requirement (0.001) appears to be low enough to dictate the need for backup computers instead of just a single computer, and would escalate the costs and complexity of the system considerably, and perhaps unnecessarily. Consequently, a cost-benefit analysis should be done to arrive at a justifiable unavailability goal for an HTGR SPDS.

IV. Summary of Conclusions and Recommendations to Date

1. Final recommendations will be made after further review of NRC PSC correspondance.
2. An assessment of the requirements for the safety parameter display system (SPDS) as outlined in NUREGs 0660, 0737, and 0696 has determined that the intended objectives of the action item may not be met because of the vague and somewhat contradictory wording of the NRC reports. Particularly in the case of FSV, which has long response times relative to LWRs, a SPDS which has only a summary of information otherwise available on the main control panels may be distracting or misleading to an operator who would probably have sufficient time to absorb and analyze a more complete set of data. Because of the tight schedules imposed on the licensee for the design and implementation of the SPDS, it is likely that relatively little analytical capability could be incorporated into it. Consequently, it is recommended that: (1) the summary display of critical parameters requirement be waived for FSV; (2) the licensee and NRC jointly develop design criteria and requirments for an analytical capability for the SPDS (such as reactivity anomaly detection, heat and mass balance computations, component performance degradation detection, accident progression prediction, etc.) and a schedule that would allow a reasonable amount of time for development.
3. A cost-benefit analysis should be done to justify the design unavailability goal requirement of 0.001 for an HTGR SPDS.

V. Proposed Further Action By ORNL

1. Review recent NRC and PSC correspondance on the emergency response facilities (ERF), and on the SPDS in particular.
2. Visit FSV to observe the status of the ERF, and discuss the present PSC SPDS design criteria with operations personnel.
3. Make an independent evaluation of HTGR SPDS detailed functional requirements.
4. Investigate cost-benefit features of single vs backup computers.
5. Develop final recommendations.

DRAFT
For Comments

J. C. Conklin
6/11/81

EVALUATION OF TMI ACTION PLAN APPLICATION TO FORT ST. VRAIN

Item II.B.3: Postaccident Sampling Capability

I. Position

The basic statements regarding radionuclide analysis can be directly applicable to Fort St. Vrain (FSV). However, the requirement that the reactor coolant spectrum correspond to a Regulatory Guide 1.3 or 1.4 might need reevaluation in light of the entirely different fuel configuration of FSV as compared to a LWR.

Specifically, the FSV fuel consists of spherical particles of $(Th + U)C_2$, coated with SiC, and dispersed in a matrix of graphite so that there is no direct contact of coolant and fuel particles. Therefore, the path for radionuclide transfer to the coolant is not as direct as the LWR. Plateout of iodine and cesium upon the graphite moderator occurs during normal operation, and might become a source term for these radionuclides during a heatup or moisture ingress accident. A regulatory guide for FSV comparable to Regulatory Guide 1.3 or 1.4, and 1.7 should be developed, based upon appropriate phenomenological considerations.

The chemical analysis requirement for boron and chloride concentration is inappropriate for FSV. Chemical analysis for foreign gases in the reactor coolant system should be substituted. These foreign gases would primarily consist of H_2O, H_2, CO, and CO_2.

II. Clarification Items (See Clarification listed in NUREG-0737, II.B.3)

1. This item is appropriate for FSV. However, as an HTGR has a substantially increased thermal capacity over an LWR, the three hour time limit for sampling and analysis might be increased if this would result in improved accuracy and measurement reliability.
2. Appropriate for FSV, with modifications
   a. unchanged
   b. hydrogen, air, CO, and CO_2 concentrations in the coolant and reactor building
   c. inappropriate
   d. unchanged.
3. unchanged
4. inappropriate
5. inappropriate
6. unchanged
7. inappropriate

8. Should be clarified, so as to preclude use of the plant helium purification system or other system designed for power production use at normal operating conditions to meet the requirements of this action item. The normal and accident coolant sampling systems should be independent.

9. Appropriate, with modifications

   a. FSV versions of Regulatory Guides 1.3, 1.4, and 1.7 need to be developed. The sensitivity requirement for liquid sample analysis is inappropriate, and a value for gaseous sample sensitivity should be substituted.

   b. unchanged

10. Modified as follows:

    Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant system, regardless of system operating conditions. This requirement specifically includes a situation where all coolant circulation has ceased, at any system pressure.

11. unchanged.

Other items unchanged.

III. Proposed Further Work or Action by ORNL

Additional work is needed to identify appropriate radionuclide source terms. A great deal of work concerning this subject has previously been done by many investigators, and the information could produce a preliminary regulatory guide for radionuclide source terms analogous to Regulatory Guide 1.3 or 1.4. A great deal of work also has been done for combustible gases (H_2, CO) that might arise from moisture ingress accidents and a preliminary regulatory guide analogous to Regulatory Guide 1.7 could be produced.

DRAFT
For Comments

R. M. Harrington
6/8/81

EVALUATION OF TMI ACTION PLAN APPLICATION TO FORT ST. VRAIN

Item II.B.4 Training for Mitigating Core Damage

I. Background

The occurrence of severe core damage implies that important reactor plant systems have not been available to, or properly used by the operator. However, at some point during a severe accident sequence by optimum use of available systems it is possible that the operator can halt further core damage and/or prevent containment failure, thereby minimizing release of radioactivity. A necessary prerequisite to optimum operator decision making is the ability to diagnose the true condition of the reactor from plant instrumentation.

It is not obvious for every accident and for each reactor plant system what the optimum way to control or mitigate the course of the accident is. The NRC is currently funding at a level of $\$2(10)^6$/year a Severe Accident Sequence Assessment (SASA) program for LWRs. One main purpose of the SASA program is to develop a base of knowledge that will allow operators of LWRs to determine the best way to control or mitigate a given severe accident. Industry has funded (at $\$8(10)^6$ total) an independent program with approximately the same goal. Nevertheless, until these research programs are completed it is important for reactor operators to summarize available knowledge of severe accidents gained from the accident at TMI and knowledge of their own reactor plants in order to provide guidance for severe accidents where none has existed in the past.

II. Present In-plant Status at FSV

FSV has committed to develop a training program outline that meets the intent of the requirement and to have that program initiated by April 1, 1981.

III. Applicability to FSV

While FSV is very different from a LWR, severe fuel damage is possible; therefore, the contingency plans for this unlikely possibility should be made a part of the FSV operational training program.

There are a number of possible ways in which the FSV core could be damaged. The steam generators, helium circulators, and reactor core are all inside the Prestressed Concrete Reactor Vessel (PCRV). An inleakage of steam from the higher pressure steam generators into the hot PCRV would cause oxidation of graphite

in coolant holes in the moderator blocks which hold the fuel. However, this reaction requires heat and therefore tends to be self-limiting. This combined with system design features makes steam ingress an unlikely candidate for severe fuel damage. Air leakage, if possible, would be of much more concern due to the heat liberation of the air + carbon reaction; however, the PCRV is pressurized with helium coolant and the available sources of air are at atmospheric pressure.

The possibility of an Anticipated Transient Without Scram (ATWS) event causing severe core damage seems unlikely due to the unique nature of HTGRs. The backup scram system consists of 1/2-inch diameter poison balls that are dropped into round channels internal to the core. Once actuated, they are in place within seconds. This contrasts to the very slow shutdown achieved by borating the coolant in LWRs. In addition, the FSV core has a very large heat capacity so that there is plenty of time for actuation of a backup scram even for a relatively large discrepancy between heat generation (power level) and heat removeal. The normal full load fuel temperature of about 2000°F is more that 1000°F below the temperature at which the refractory fuel particle coatings begin to fail, allowing volatile fission products to escape. For these reasons ATWS events have a low priority for degraded core considerations at FSV.

Sequences that should receive priority degraded core consideration for FSV involve the loss of forced circulation. The FSV design has steam generators well below the reactor core, therefore even at the full coolant pressure of 700 psia, natural circulation alone will not provide sufficient heat removal. If circulation is available, then adequate heat removal via the steam generators is possible at any helium pressure down to and including atmospheric. Since the steam-driven circulators and the steam generators are normally used for power production, backup safety systems were provided to assure their availability for emergency use. The fire-water system provides both a redundant drive for the circulators (via the Pelton wheel) and a redundant supply of feedwater to the steam generators.

Sequences involving a postulated loss of all 4 circulators are shown in a very schematic fashion on Figure 1. The dominant sequence (No. 1) has already received considerable attention: it is Design Basis Accident No. 1 (DBA-1) and is fully discussed in the FSAR (as amended by PSC submittal P-77250 dated December 1977). When the circulators fail, the reactor trips, and the massive 27 ft. diameter by 23 ft. high reactor core begins to heat up in an approximately adiabatic manner. Figure 2 reproduces the fuel temperature vs time reported for DBA-1 in the FSAR. This figure is included here to illustrate the very slow rate of core temperature rise during unrestricted heatup. With respect to core heatup

Figure 1: Examples of Severe Accident Sequences for FSV Circulator Failure

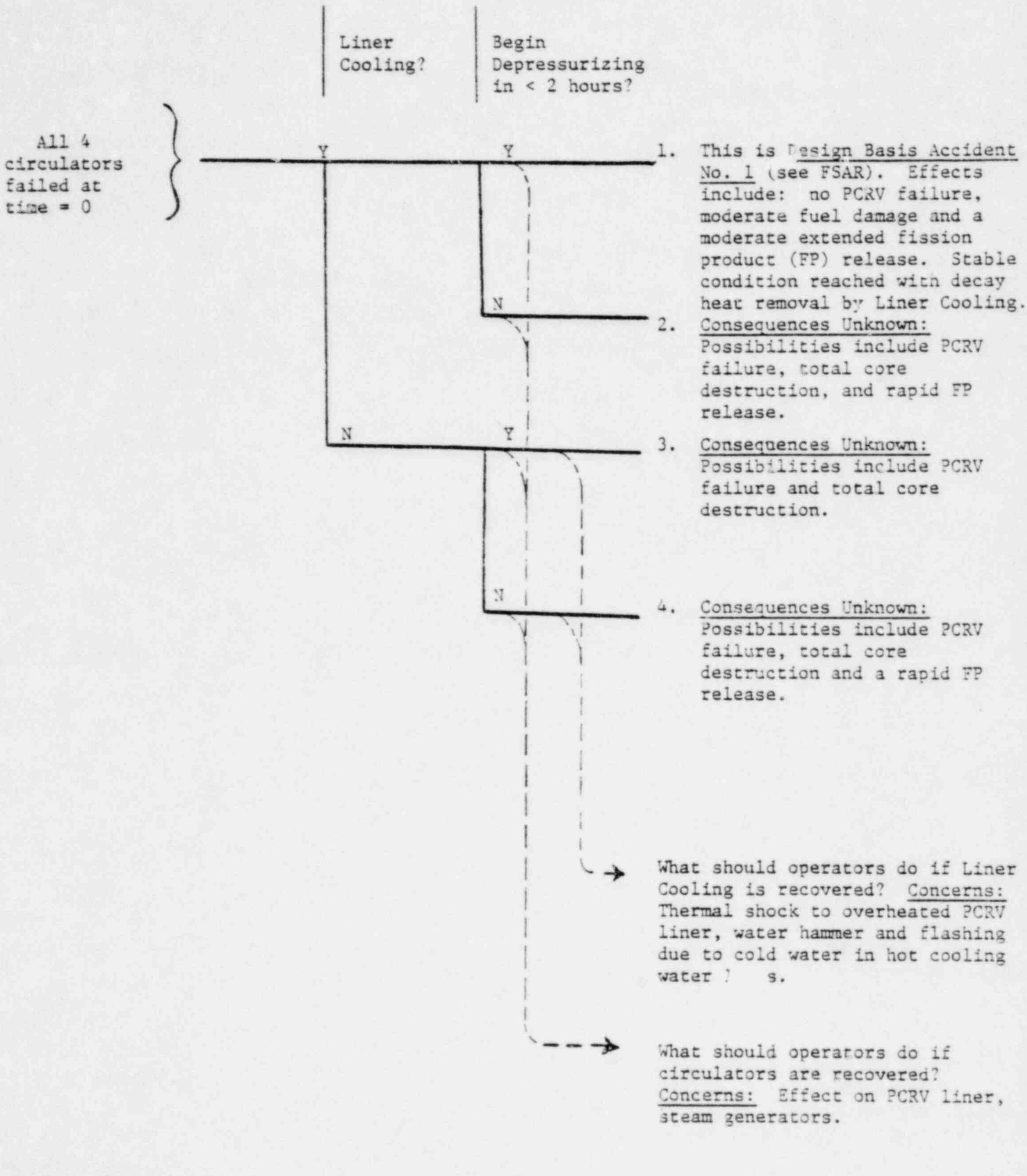

Fig. 2: Core Temperatures During Loss Of Forced Convection Accident (from Fort St. Vrain FSAR)

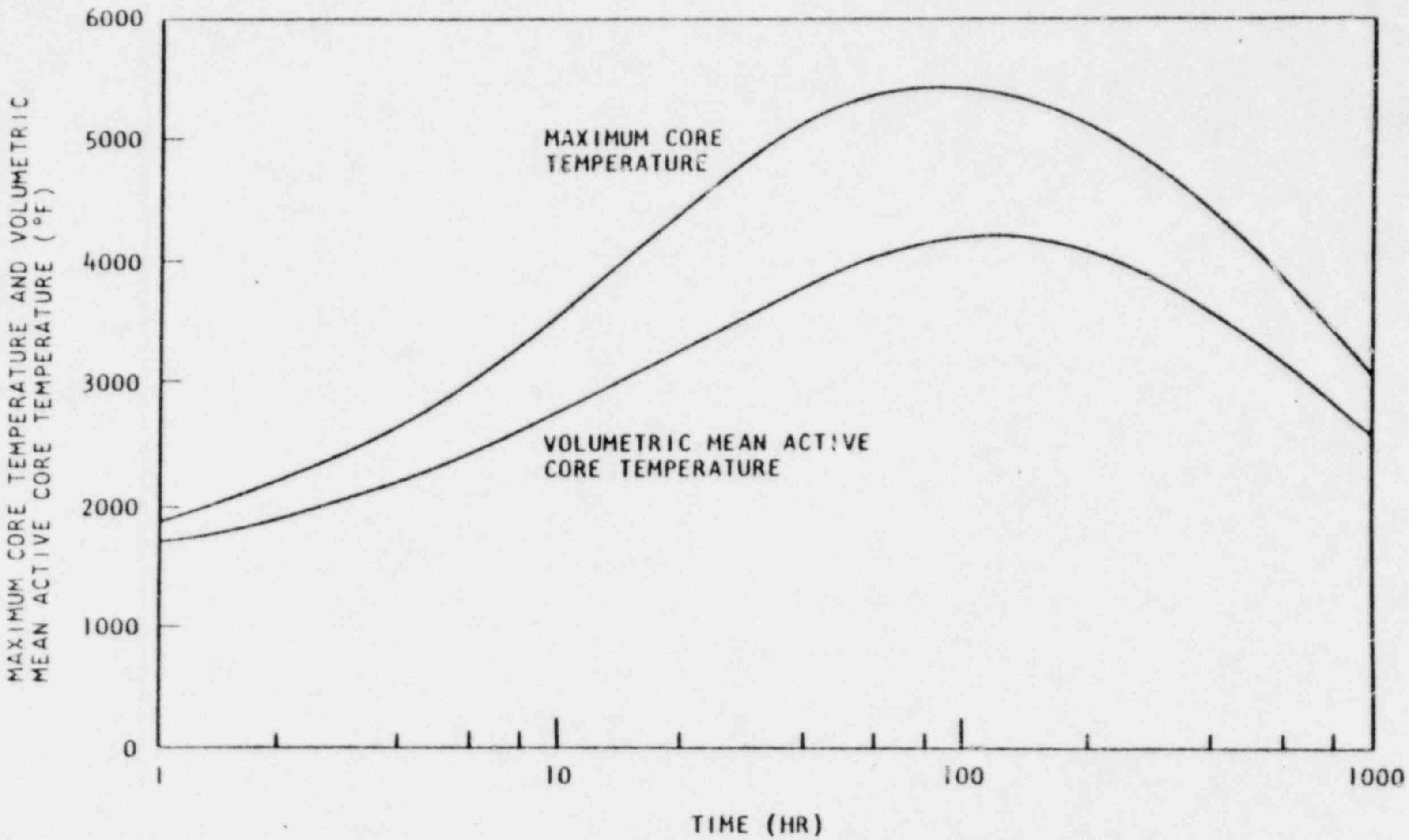

Fig. 14.10-1--Maximum and volumetric mean active core temperature during LOFC accident

Fig. 3: PCRV Thermal Barrier Arrangement (from FSAR)

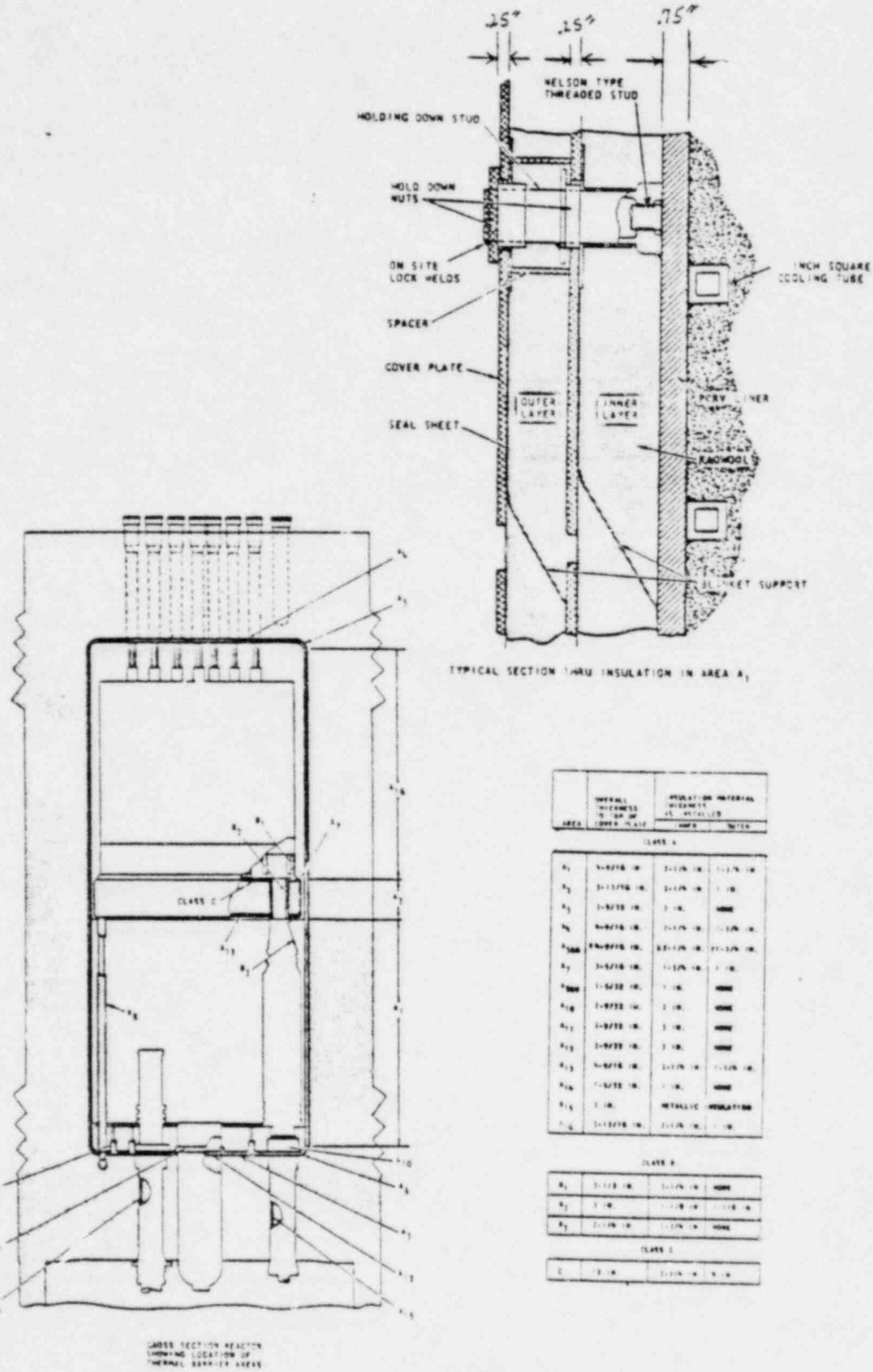

Fig. 5.9-3--Thermal barrier arrangement

Table 1 PCRV Material Properties*

Component	Property	Temperature (°F)
Cover Plates (o.25" thickness steel)	failure	1500 to 2000
Kaowool Insulation	Melting	3200
Liner (0.75" thickness steel)	failure	2000
Concrete	design temp.	150
"	free water loss	190 to 275
"	decarboxylation (CO_2 production)	1630 to 1710
"	zero strength	1800
"	melting point	2000 to 2700

*Reference: HTGR AIPA Status Report: Phase II Risk Assessment, GA-A15000, UC-77, April 1978.

rate, the initial portion of Figure 2 would be applicable to all the sequences shown on Figure 1. Also shown by Figure 2 is the fact that the reactor can possibly reach a stable operating condition after DBA-1 without ever regaining the use of the circulators. Although the system was depressurized during the early part of DBA-1 to limit heat transfer to the liner, after about 100 hours the outer surface temperatures of the core are high enough such that radiant heat transfer directly to the liner exceeds the core heat generation rate, which by 100 hours is diminished by natural decay and by loss of volatile fission products (i.e. when temperature exceeded the 3100°F fuel particle coating rupture temperature). Liner cooling flow is assumed available throughout DBA-1. Radiological consequences of DBA-1 could be very light if the PCRV depressurization is completed before a significant number of fuel particles begin to rupture.

In addition to high temperature effects on the reactor core, a severe accident analysis for FSV must consider also the high temperature effects on the PCRV. Concrete is clearly not a high temperature material. As shown on Figure 3, the concrete is protected from the high temperature helium environment by a 3/4 inch thick carbon steel liner and varying thicknesses of insulation (depending on design heat load for each location). The liner is cooled by water flowing through tubes welded to the outer surface of the liner. Table 1 gives high temperature properties of selected PCRV components.

The PCRV is depressurized* to 5 psig during DBA-1 in order to protect the liner and PCRV. The concern is that, without depressurization, natural circulation convective heat transfer from the hot coolant would cause failure of the 1/4 inch thick insulation cover plates in the top head region. The temperature from hotter refueling regions can exceed 2000°F after LOFC. If the cover plates failed, the insulation could then drop away, exposing the liner to excessive temperature and possibly causing liner failure and degradation of the PCRV concrete.

This situation would not be self-limiting. Any degradation products could fall directly onto the top of the core, further complicating the chances for recovery. If PCRV degradation were sufficient to cause failure of the vessel, there would be a sudden release of the PCRV contents. This is the cause of the much more severe sequences shown on Figure 1: in order to prevent severe consequences it is necessary to protect the PCRV. If the PCRV failure occurs at high pressure then there is a much greater chance of a large rapid release.

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*To minimize radiological consequences and liner damage, the depressurization must start within 2 hours of the loss of forced convection, and be completed over a time period of about 7 hours.

Although the chance of severe accidents as shown on Figure 1 may be diminishingly small (especially for sequences 2,3, and 4), these are examples of the sequences that must be considered in the requirement for training for control and mitigation of degraded core accidents at FSV. The quantitative aspects of each major sequence must be investigated sufficiently such that time to reach various branch points is known, and temperatures of critical system components are known from best estimate computer analyses. For any given sequence written operational recommendations should be available concerning:

1. Desired (or likely) stable end condition
2. Recommended recovery strategy as a function of time when failed systems become available.

Recovery strategies are important because it is more likely that capabilities will be restored during an accident, especially for a slow-responding HTGR system. In addition, it is very improtant that restored systems be brought on in such a manner that damage is not caused by thermal shock or other unanticipated effects.

In order for the operators to make proper mitigation and control decisions they must be able to accurately assess the condition of the reactor core and the PCRV. Operational training and written procedures should specify how to do this. Topics covered should include:

1. How the core outlet thermocouple readings relate to interior core temperatures during severe heat-up transients.
2. Which radiation moniotrs can be used to estimate release of fission products from the fuel into the helium coolant within the PCRV.
3. Effects of high radiation levels on instrumentation readings.
4. How the condition of the PCRV liner and concrete can be determined. Clearly, the liner cooling water tube outlet temperatures are valuable, especially if there is forced cooling water flow. If forced flow were interrupted, these outlet temperatures might provide indication of liner heat-up by measuring the amount of superheat in steam produced as the contents of the tubes boil off.
5. The effect of liner heat-up on the nuclear instrumentation indications as a function of PCRV temperature.

IV. Summary of Conclusions and Recommendations-Short Range

The requirement for training for control and mitigation of degraded core events can best be met by development of contingency procedures and background information on:

1. Response of plant instrumentation during severe accident sequences, including the effects of accident conditions such as high radiation and temperature.
2. Use of plant instrumentation to assess the condition of the core and PCRV and to determine the amount of fission products released from the core to the reactor coolant.
3. Optimum use of plant systems as a function of time in severe accident sequences.
4. Necessary special procedures for startup of failed plant systems if they do become available at some point in a potentially severe accident sequence.

Long Range

A sufficient data base of calculational and other investigative results does not exist to permit adequate consideration of the full range of severe accident sequences possible at FSV. Therefore, a task of assessment of accident sequences specific to FSV should be initiated. The AIPA study conducted at GA[1] should provide an excellent starting point for such a study, but will not of itself satisfy this requirement because it:

1. Is specific to the 3000 MW HTGRs and not to the 842 MW FSV HTGR.
2. Is concerned more with the calculation of overall statistically expected radiological consequences of severe and non-severe accidents rather than consequences of severe accident sequences.

V. Proposal for Future Work at ORNL

Short Term

Future work at ORNL should consist of a review of the procedures for training for control and mitigation of degraded core accidents developed at FSV. After this review, and after consultation, as necessary, with NRC, PSC and GA personnel then ORNL can make final recommendations about the adequacy of the training procedures.

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[1]"HTGR Accident Initiation and Progression Analysis Status Report," Volumes I thru VII, GA-A13617 and GA-A15000, October 1975 through April 1978.

Long Range

As indicated in a previous se.tion, there is a need for continuing study of FSV severe accident sequences. ORNL can contribute to this study in a number of ways:

1. Futher study of severe accident sequences possible at FSV (part of an existing RSR-Sponsored program).
2. Review of GA models used for calculation of thermal-hydraulic and radiological consequences of severe accidents.
3. Modification of ORNL thermal-hydraulic codes to allow calculation of severe accident effects on the PCRV and liner cooling system.
4. Study of and development of computer methods for prediction of radiation transport during severe HTGR accidents.
5. Performance of calculations to check predictions of severe accident consequences calculated by GA for FSV.

DRAFT
For Comments

S. J. Ball
6/8/81

EVALUATION OF TMI ACTION PLAN APPLICATION TO FORT ST. VRAIN

Requirement II.K.3.17: Report on Outages of Emergency Core Cooling Systems

I. Background

The NRC requirement for licensees to report historical data (for the last 5 years) on cumulative outage times for ECCS components is part of a systems reliability analysis task, II.K, in NUREG-0660 (May 1980). The objective of this task is to improve emergency operating procedures and operator training to improve the capability of plants to mitigate the consequences of loss-of-coolant accidents (LOCA) and loss-of-feedwater events. The present requirement was one of the items recommended by the NRR Bulletins and Orders (B&O) task force. The purpose is to provide NRC with data to evaluate critical component unavailability, and thus determine if tech specs are needed on cumulative outage times. The requirements of the licensee report are given in NUREG-0737, which specifies a Jan. 1, 1981 deadline. PSC responded (P-80441, Dec. 26, 1980) by giving outage histories for the PCRV cooling systems (System 46) and the standby diesel generators (System 92), noting that these are the only two ECCS-related systems for which the tech specs permit substantial outage times. Other FSV ECCS systems, such as the circulators and steam generators, are part of the normal plant operation cooling systems. Hence PSC claims that their reliability and availability is continuously demonstrated by plant operation, and so no reports on their unavailability are necessary.

II. Evaluation of the Applicability of Item II.K.3.17 to FSV

The Diesel generator emergency power supply system and the PCRV liner cooling system are clearly critical parts of the FSV ECCS. PSC's report on their unavailability stated that neither of the two (redundant) liner cooling systems had any downtime in the last 5 years, so its historical unavailability is zero.

Analysis of PSC data for the two (redundant) diesel generators showed an average single-unit unavailability due to all causes (including routine maintenance downtimes) of 0.0088. The average single-unit unavailability due to forced outages was 8.2×10^{-5}. Hence an approximate historical unavailability for emergency diesel power would be the product of the above numbers, 7.2×10^{-7}, or less than one in a million. Other observations about the PSC data: the forced outage times for the diesels ranged from $\sim$2 to 3 hours; and there was no apparent significant deterioration with time.

It is recognized that the PCRV liner cooling system is the ultimate ECCS for FSV, and that, except for the use of the firewater system as a backup cooling system, the diesel generators are needed as the emergency power source for the PCRV coolant circulator pumps. However, the main helium circulators and steam generators are also part of the ECCS, and even though they are used for normal pant operation, they have not demonstrated a zero unavailability history during post shutdown periods. Thus in order to more completely evaluate total ECCS unavailability, historical data on other subsystem and component performance would be useful, including the circulators, the emergency feedwater and firewater systems, and the steam generator normal and emergency cooling water supply systems.

III. Summary of Conclusions and Recommendations

1. Final recommendations will be made after further discussions with NRC and PSC.
2. The last-five-year historical record indicates excellent (low) unavailability for the PCRV liner cooling systems and the emergency diesels. No tech spec changes appear to be necessary if the good records persist.
3. To evaluate the total ECCS unavailability, historical downtime data is needed for the circulators and steam generators and those subsystems used (including backups) to operate them during decay heat removal periods.

IV. Proposed Further Action by ORNL

1. Discuss the need for more data with NRC and PSC.
2. Draw up final recommendations.
3. Analyze unavailability data.