PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

February 27, 1985

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

Public Service Electric and Gas Company presently plans to issue Amendment No. 10 to the Hope Creek Generating Station Final Safety Analysis Report by May 1, 1985. Accordingly, this letter is provided to document the status of Hope Creek Generating Station responses to NRC requests for additional information which were forecasted to be responded to by January 1985.

Attachment I is a tabulation of the Hope Creek Generating Station Final Safety Analysis Report commitments for January 1985, and the corresponding resolution for each commitment. Attachments II through XIV provide responses to commitments forecasted to be responded to in January 1985, which will be included in Amendment No. 10.

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Should you have any questions in this regard, please contact us.

Very truly yours,

R Turbett

Attachment I - Hope Creek Generating Station - FSAR Commitment Status through January 1985
Attachment II - Response to FSAR Section 9.3.2.2.2.11
Attachment III - Response to FSAR Section 9.3.2.3.2
Attachment IV - Response to TMI Item I.A.3.1
Attachment V - Response to TMI Item II.B.4
Attachment VI - Response to TMI Item II.K.3.27
Attachment VII - Response to Question 210.20
Attachment VIII - Response to Question 430.40
Attachment IX - Response to Question 430.94
Attachment X - Response to Question 430.111
Attachment XI - Response to Question 430.133
Attachment XII - Response to Question 440.10
Attachment XIII - Response to Question 460.4
Attachment XIV - Response to Question 630.4

C D. H. Wagner (w/attach)
USNRC Licensing Project Manager

A. R. Blough (w/attach)
USNRC Senior Resident Inspector

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
1. FSAR Section 6.2.3.2.3	This commitment concerns providing the Feedwater Bypass Leakage Analysis. This information will be provided in March 1985.
2. FSAR Section 9.3.2.2.11	This commitment concerns providing procedures for disposal of lab samples. This information will be provided in April 1985. This revised commitment date, provided in Attachment II, will be included in Amendment 10 to the HCGS FSAR.
3. FSAR Section 9.3.2.3.2	This commitment concerns providing a plant specific procedure to access the extent of core damage based on radionuclide concentrations and other parameters. This information will be provided in April 1985. This revised commitment date, provided in Attachment III, will be included in Amendment 10 to the HCGS FSAR.
4. FSAR Section 13.2.2	This commitment concerns providing the Fire Brigade Training Program. This information has been provided in Amendment 8 to the HCGS FSAR.

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
5. FSAR Section 15.8.1	This commitment concerns providing an Emergency Operating Procedure for Reactor Control. This information will be provided in March 1985.
6. Question/Response Appendix: Question 100.6	Re: TMI Item I.A.3.1; This commitment concerns providing the requalification training program. This information has been provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated December 28, 1984. The information provided in Attachment IV will be included in Amendment 10 to the HCGS FSAR.
	Re: TMI Item I.C.8; This commitment concerns providing the Procedure Generation Package. This information has been provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated January 28, 1985. The information in this letter will be included in Amendment 10 to the HCGS FSAR.
	Re: TMI Item II.B.4; This commitment concerns implementing training for mitigating core damage. This information, provided in Attachment V, will be included in Amendment 10 to the HCGS FSAR.

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
6. Question/Response Appendix: Question 100.6 (Cont'd)	Re: TMI Item II.K.3.27; This commitment concerns establishing a common reference water level in the reactor vessel. This information, provided in Attachment VI, will be included in Amendment 10 to the HCGS FSAR.
7. Question/Response Appendix: Question 210.20	This commitment concerns providing an analysis to evaluate structural integrity for other valves parts per the response to DSER Open Item No. 37. This information will be provided in May 1985. This revised commitment date, provided in Attachment VII, will be included in Amendment 10 to the HCGS FSAR.
8. Question/Response Appendix: Question 430.40	This commitment concerns providing procedures for periodic testing of Class 1E safety systems. This information will be provided in April 1985. This revised commitment date, provided in Attachment VIII, will be included in Amendment 10 to the HCGS FSAR.
9. Question/Response Appendix: Question 430.76	This commitment concerns providing DG instruments and controls surveillance frequency. This information has been provided in Amendment 8 to the HCGS FSAR.

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
10. Question/Response Appendix: Question 430.94	This commitment concerns providing diesel fuel oil emergency fill procedures. This information will be provided in April 1985. This revised commitment date, provided in Attachment IX, will be included in Amendment 10 to the HCGS FSAR.
11. Question/Response Appendix: Question 430.104	This commitment concerns providing diesel engine cooling system I&C surveillance frequency. This information has been provided in Amendment 8 to the HCGS FSAR.
12. Question/Response Appendix: Question 430.111	This commitment concerns providing an emergency DG operating procedure including no-load operation. This information will be provided in April 1985. This revised commitment date, provided in Attachment X, will be included in Amendment 10 to the HCGS FSAR.
13. Question/Response Appendix: Question 430.127	This commitment concerns providing diesel engine lube oil system I&C surveillance frequency. This information has been provided in Amendment 8 to the HCGS FSAR.
14. Question/Response Appendix: Question 430.133	This commitment concerns providing local diesel operation procedures. This information will be provided in April 1985. This revised commitment date, provided in Attachment XI, will be included in Amendment 10 to the HCGS FSAR.

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
15. Question/Response Appendix: Question 430.140	This commitment concerns providing diesel engine combustion air intake and exhaust I&C surveillance frequency. This information has been provided in Amendment 8 to the HCGS FSAR.
16. Question/Response Appendix: Question 440.10	This commitment concerns providing trip settings for the leak detection system. This information is provided in Attachment XII and will be included in Amendment 10 to the HCGS FSAR.
17. Question/Response Appendix: Question 460.4	This commitment concerns providing information on laboratory tests conducted under the process control program. This information will be provided in September 1985. This revised commitment date, provided in Attachment XIII, will be included in Amendment 10 to the HCGS FSAR.
18. Question/Response Appendix: Question 630.4	This commitment concerns incorporating core damage training into respective training programs. Technician training has been incorporated and management training will be provided prior to fuel load. This information, provided in Attachment XIV, will be included in Amendment 10 to the HCGS FSAR.
19. Question/Response Appendix: Question 630.12	This commitment concerns providing the Fire Brigade Training program. This information has been provided in Amendment 8 to the HCGS FSAR.

ATTACHMENT I
HOPE CREEK GENERATING STATION
FSAR COMMITMENT STATUS THROUGH JANUARY 1985

FSAR COMMITMENT LOCATION	COMMITMENT RESOLUTION
20. Question/Response Appendix: Question 640.17	This commitment concerns providing a description of confirmatory in-plant tests. This information has been provided in Amendment 8 to the HCGS FSAR.
21. Question/Response Appendix: Question 640.27	This commitment concerns providing the PGP. This information has been provided in letter; R. L. Mittl (PSE&G) to A. Schwencer (NRC), dated January 28, 1985. The information in this letter will be included in Amendment 10 to the HCGS FSAR.
22. DSER OI Nos. 170 and 171	This commitment concerns providing the PGP. See resolution above for Question 640.27.

ATTACHMENT II

The small volume (diluted) liquid sample cask is a cylinder with a lead wall thickness of about 2 inches. The cask weighs approximately 50 pounds and has a handle which allows it to be carried by one person.

The 10 milliliter undiluted sample is taken in a 700 pound lead shielded cask which is transported and positioned by a four-wheel dolly. The sample is shielded by about 5-1/2 inches of lead.

9.3.2.2.2.10 PASS Power Supply

The PASS isolation and control valves, sample station control panels, isolation valve control panels, and auxiliary equipment are connected to a non-1E battery backed power source. The safety auxiliaries cooling system, which is needed for the sample coolers, is powered from the emergency diesel generators following a loss of offsite power. Power for the gas sample line heat tracing is supplied from a diesel backed source.

9.3.2.2.2.11 Storage and Disposal of Sample

Short-term sample storage areas will be provided in the chemistry laboratory and counting room facilities. An area for long-term storage of the samples will be designated prior to core load. Low level wastes generated by routine chemistry evolutions will be flushed to radwaste. Procedures addressing the ultimate disposal of the samples will be prepared by ~~January~~ 1985.
April

9.3.2.3 Safety Evaluations

9.3.2.3.1 Process Sampling System Safety Evaluation

The process sampling system has no safety-related function. Failure of the system will not compromise any safety-related system or component, or prevent a safe shutdown of the plant.

The process sampling lines, connected to the reactor coolant pressure boundary (RCPB) through the first isolation valve outside containment, are designed to seismic category I requirements, as defined in Section 3.7. Sample lines that penetrate the containment are provided with isolation valves in accordance with 10 CFR 50, Appendix A, GDC 55, as described in Section 6.2.4.

9.3.2.3.2 Post-Accident Sampling System Safety Evaluation

The PASS has no safety-related function. Failure of the system will not compromise any safety-related system or component, or prevent a safe shutdown of the plant.

ATTACHMENT III

However, NUREG 0737 Section II.B.3.1 requires that the PASS meet the following:

a. The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

   The following is a conservative time sequence for sampling, transport, and analysis to demonstrate that samples can be obtained and analyzed within the specified 3-hour period:

   1. Recirculate sample, install sample vial/or cartridge -- 15 min.

   2. Operate sample station -- 15 min.

   3. Transport sample to lab -- 20 min.

   4. Analyze sample --- 30 min.

   Sample points and sample gathering methods are discussed in Section 9.3.2.2.2.

b. The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

   1. Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and nonvolatile isotopes);

      A generic procedure to assess the extent of core damage based on radionuclide concentrations and other parameters has been prepared by the BWR Owners Group (FSAR Section 1.8.1.97). A HCGS plant specific procedure based on this methodology will be prepared by ~~January,~~ April 1985.

   2. Hydrogen levels in the containment atmosphere;

      At greater than 15% power, the primary containment atmosphere is maintained under a nitrogen blanket. Hydrogen and oxygen concentrations are monitored by chemical analysis of gas samples drawn from various points in the drywell and torus. During post accident conditions, hydrogen and oxygen

ATTACHMENT IV

Clarification

The clarification does not alter the staff's position regarding simulator examinations.

The clarification does provide additional preparation time for utility companies and NRC to meet examination requirements as stated. A study is under way to consider how similar a nonidentical simulator should be for a valid examination. In addition, present simulators are fully booked months in advance.

Application of this requirement was stated in June 1, 1980 to applicants where a simulator is located at the facility. Starting October 1, 1981, simulator examinations will be conducted for applicants of facilities that do not have simulators at the site.

NRC simulator examinations normally require 2 to 3 hours. Normally, two applicants are examined during this time period by two examiners.

Utility companies should make the necessary arrangements with an appropriate simulator training center to provide time for these examinations. Preferably these examinations should be scheduled consecutively with the balance of the examination. However, they may be scheduled no sooner than 2 weeks prior to and no later than 2 weeks after the balance of the examination.

Response

Requalification program requirements have been ~~will be~~ outlined in the Nuclear Department Training Center procedure, TP-305. ~~(to be revised for Hope Creek by January 1985).~~ 1 -HC

The Hope Creek simulator is scheduled for delivery in the summer of 1984 and is expected to be operational for training by the fall of 1984. Until the Hope Creek simulator is operational, any necessary simulator training will be conducted at the Susquehanna training center or other simulator training facilities.

See Section 13.2 for further discussion.

ATTACHMENT V

liquid outside containment; importance of using leak-tight systems.

(b) Expected isotopic breakdown for core damage; for clad damage.

(c) Corrosion effects of extended immersion in primary water; time to failure.

(5) Radiation Monitoring

(a) Response of process and area monitors to severe damage; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

(b) Methods of determining dose rate inside containment from measurements taken outside containment.

(6) Gas Generation

(a) Methods of hydrogen generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of noncondensibles.

(b) Hydrogen flammability and explosive limit; sources of oxygen in containment or reactor coolant system.

Managers and technicians in the instrumentation and control, health physics, and chemistry departments shall receive training commensurate with their responsibilities.

Response

A program for training of all plant ~~operating~~ operations staff in mitigating core damage ~~will be~~ has been implemented~~, by January 1985.~~

ATTACHMENT VI

hours. These results are applicable to the Byron-Jackson pumps used at HCGS. The normal or emergency controls for reactor water level could easily accommodate this small leakage rate.

- II.K.3.27 PROVIDE COMMON REFERENCE LEVEL FOR VESSEL LEVEL INSTRUMENTATION

Position

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

The applicant is to submit documentation by January 1, 1981 and implement action by April 1, 1981.

Response

The Hope Creek position on TMI issue II.K.3.27 was the current BWROG study, NEDO-24951, which stated that the current BWR water level indication system is fully adequate to allow plant operations to respond properly under all postulated reactor conditons and that there are no required design changes based on any plant safety considerations.

This evaluation was rejected by the NRC as explained in the letter from D.G. Eisenhut to D.B. Waters, dated April 6, 1981. In this letter, the NRC stated its position that "... all level instruments should be referenced to the same point. The selection of the reference point for any specific reactor has been left to the discretion of the licensee..." In light of this situation, HCGS has established ~~will establish a~~ the bottom of the dryer skirt as the common reference point for instruments measuring water level in the reactor vessel. ~~Appropriate design modifications will be implemented by December 1984.~~

- II.K.3.28 VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

Position

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. General Electric has also stated that the emergency core cooling (ECC) systems are designed to

ATTACHMENT VII

QUESTION 210.20 (SECTION 3.6.2)

Provide the basis for assuring that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside containment.

RESPONSE

The feedwater line break outside the containment will be simulated using the RELAP5 computer code. A check valve model which has been developed specifically for calculations of this nature will be used for obtaining the valve dynamics. From this thermal hydraulic analysis, the peak pressures upstream and downstream of the valve disc as well as the maximum disc angular speed will be obtained. As part of this analysis, a sensitivity analysis will be performed to determine the break location and feedwater check valve selection that yields the most conservative stress results.

The stress analysis for the feedwater swing check valve will be performed for fluid transient loads induced by the pipe break event. Elastic and/or inelastic analysis will be performed to determine the primary stress intensities and/or strains at critical locations. For the pressure retaining boundaries such as valve body and valve disk, the calculated primary stress intensities shall not exceed Level D, Service Limit (3.0 Sm). For other valve parts such as seat ring, hinge, and actuator shaft, structural integrity will be evaluated based on strain criteria determined by material properties. The results of this analysis will be provided in ~~January~~ 1985.

May

ATTACHMENT VIII

QUESTION 430.40 (SECTION 8.3.1 and 8.3.2)

Section 6.3 of IEEE standard 308-1974 requires that periodic equipment tests shall be performed to detect the deterioration of equipment toward an unacceptable condition. In Section 8.1.4.6 of the FSAR you state that these periodic equipment tests will be performed to detect deterioration within practical limits. Define the terminology "within practical limits". Describe periodic test to be performed to demonstrate calibration and adjustment of metering and protective devices are not deteriorating toward an inacceptable condition.

RESPONSE

The statement, "within practical limits" cannot be readily defined. Section 8.1.4.6 has been revised to read "within prescribed limits". Prescribed limits are minimum performance requirements, such as response time, set point accuracy, and other performance requirements as stated in the design basis.

Periodic testing requirements of Class 1E safety systems are in the process of being identified. These tests will be performed using written procedures, each of which will be designed to produce the objective data necessary for assessing the performance and the availability of the equipment being tested. These written procedures will be available by ~~January~~ April 1985.

ATTACHMENT IX

QUESTION 430.94 (SECTION 9.5.4)

In Section 9.5.4.2.6 of the FSAR you state that the emergency flood protected truck fill connection for the fuel oil storage tanks is located inside the auxiliary building at Elevation 102 feet (plant grade level). In Section 3.4, Table 3.4-1 you state that the design flood elevation for the D/G building is 120.4 feet with the still water height at 113.8 feet. Provide the following:

a. Describe or provide adequate drawings to show the location of the emergency fuel oil storage tank fill connection.

b. Assuming the emergency fill connection must be used to refill the fuel oil storage tanks. Describe how fuel oil will be delivered to the site during flood conditions and describe the procedures that will be used in refilling the storage tanks during flood conditions and non-flood conditions. The procedures should include fuel hose routing and fire watcher.

c. Describe how flood water is prevented from entering the building during refueling operations. (SRP 9.5.4, Parts I, II & III)

RESPONSE

The diesel fuel oil emergency fill line is located in the auxiliary building at floor elevation 102 feet-0 inches and a center line elevation of 106 feet-6 inches. The emergency diesel fuel oil fill connection is located in an area which is flood protected by the auxiliary building main service entry doors. The location of the diesel fuel oil connection is shown on Figure 430.94-1, reference Figure 1.2-35 for location relative to watertight door.

System operating procedures (SOP) will address the refilling of the storage tanks from any fill connection and include proper fuel hose routing and the establishment of a fire watch, when necessary. Abnormal operating procedure, acts of nature, shall provide direction to the operator as to which SOP is to be used, dependent upon environmental conditions. These procedures will be in place by ~~January~~ April ~~March~~ 1985. Refer to response to Question 430.88 for information pertaining to fuel oil delivery to HCGS during flood conditions.

INSERT "A"

Leakage through the door seals is removed by drainage systems in the building. The flood doors are capable of withstanding the flood height as described in Section 3.4 and Table 3.4-1.

ATTACHMENT X

QUESTION 430.111 (SECTION 8.3 & 9.5.5)

The diesel generators are required to start automatically on loss of all offsite power and in the event of a LOCA. The diesel generator sets should be capable of operation at less than full load for extended periods without degradation of performance or reliability. Should a LOCA occur with availability of offsite power, the diesel generator running in an unloaded (standby) condition for an extended period of time, should not result in degradation of engine performance or reliability. In Section 9.5.5.1 or the FSAR you state that the diesel generator should "remain operational after 8 hours of no-load operation, provided that the SDG runs up to a minumum of 25% of full load for 1 hour immediately after such no load operation." Verify the following:

a. Verify that the statement conforms to the manufacturer's recommended no-load operation for this diesel or justify non-conformance.

b. Verify that the conditions for no-load operation will be included in the plant operating procedures. (SRP 8.3.1, Parts II and III and SRP 7.5.5, Part III)

RESPONSE

a. In conformance with the manufacturer's latest recommendation, Section 9.5.5.1.e. has been revised to state that the diesel generators should remain operational after 12 hours of no-load operation, provided that the SDG runs up to a minimum of 50% of full load for 1 hour immediately after such no load operation.

This exceeds the recommendation by Colt Industries for the PC2 type of engine, reference letter to Mr. Clemerson (NRC) from Mr. V. T. Stonehocker (Colt Industries), dated September 11, 1975, para 2.a.

b. The conditions for diesel generator no-load operation will be included in OP-SO.KJ-001(Q), Emergency Diesel Generator Operation. Available ~~January~~ 1985.

~~MARCH~~
April

ATTACHMENT XI

required, directly to the engine sump via the gravity fill line on the side of the crankcase.

2. Operations Department Document Control Administrative Procedure will include a distribution list for field approved procedures. The diesel generator rooms will be listed for containing a controlled copy of the procedures required for local diesel operation. (Procedure ID is OP-AP.ZZ-~~005~~ 003). Available ~~January~~ ~~March~~ April 1985.

3. The level will be verified by the level indicator on the lube oil make-up tank or if the oil is added directly to the sump it will be verified by the non-recurrance of the low level alarm after its acknowledgement and a dipstick from the sump.

4. Diesel engine lubricating oil fill points will be clearly labelled to identify the fill point, type of lubricating oil required, and the number of the applicable operating procedure. A controlled copy of the procedures required for diesel engine operation will be posted in the diesel generator area.

b. Operations Department Document Control Administrative Procedure will include a distribution list for field approved procedures. The diesel generator rooms will be listed for containing a controlled copy of the procedures required for local diesel operation. Procedure ID is OP-AP.ZZ-005. Available January 1985.

c. Operations personnel will be trained in the use of diesel generator operating procedures during the operator training programs and during the preoperational test program. Operator requalification will require demonstration of the proper use of the procedures.

d. The points for adding lube oil to the diesel generator will be tagged. Also, the diesel-generator area is a "controlled access" area. Access will be limited to authorized plant personnel, trained in maintaining the diesel-generators.

ATTACHMENT XII

QUESTION 440.10 (SECTION 5.4.6)

BWR operating experience has shown that HPCI and RCIC systems have been rendered inoperable because of inadvertent leak detection isolations caused by equipment room high differential temperature signals. The events occurred when there was a relatively sharp drop in outside temperature. As noted in Section 5.4.6.1.1.1, HCGS incorporates this type of RCIC and RHR (steam) isolation. Provide a discussion of the modifications that have been or will be made to prevent inadvertent isolations of this type which effect the availability and reliability of the RCIC and the RHR systems.

Secondly, provide the trip settings for isolation of the RHR and RCIC systems due to high temperature in terms of degrees above ambient temperature.

Also, discuss the method of specification that would be applied. Show that the setting could not be set too low and cause inadvertent isolation when the system is needed.

RESPONSE

General Electric is not aware of any inadvertent isolations attributable to the leak detection differential temperature monitors sensing a sudden drop in outside temperatures. The HVAC system would have to fail to allow a sudden change in outside temperature to result in the sudden change in inside temperature. It is highly unlikely that both the temperatures would drop suddenly and the HVAC would fail at the same time that the RCIC or RHR system was required to function.

The trip settings for the plant leak detection system have not yet been established. They will be available by ~~January 1985.~~

In order to prevent inadvertent isolation when the RHR and RCIC systems are needed, HCGS will consider the potential for changes in outdoor temperatures causing system isolation and consider minimum room air supply temperature in establishing the differential temperature setpoint.

Delete and replace with attached Insert.

The trip settings for the plant leak detection system have been established based on the need to alarm and/or isolate a specified system when required while at the same time avoiding an inadvertent isolation when the system is needed. The calculations performed to determine the leak detection instrumentation setpoints considered, among other things, room volume, leakage rate, normal and minimum room temperatures and differential temperatures, and purpose of the setpoint (i.e.; alarm function only, or alarm and isolation).

The instrument setpoints for the RHR and RCIC systems are shown on the attached table. As shown on the table, the leak detection system provides alarm and isolation functions for the high energy RCIC system. For the ~~high~~ moderate energy RHR system an alarm function only is provided. FSAR Section 5.2.5 and Table 5.2-11 have been revised to reflect that high temperatures or differential temperatures in the RHR area are alarmed in the control room; no isolation of the RHR system occurs.

RHR/RCIC LEAK DETECTION INSTRUMENTATION SETPOINTS

SYSTEM	AREA	SETPOINT	FUNCTION
RHR	Equipment Room High Temperature	130°F	Alarm
RHR	Equipment Room Differential Temperature	40°F	Alarm
RCIC	Equipment Area High Temperature	125°F	Alarm
RCIC	Equipment Area High Temperature	160°F	Alarm/Isolate
RCIC	Equipment Area Differential Temperature	70°F	Alarm/Isolate
RCIC	Pipeway High Temperature	160°F	Alarm/Isolate
RCIC	Torus Area High Temperature	130°F	Alarm/Isolate

located in the area of the main steam, HPCI, and RCIC lines. The remaining dual elements are used in pairs to provide measurement of differential temperature across (inlet to outlet) the tunnel area and pipe routing area vent system. All temperature elements are located or shielded so as to be sensitive to air temperatures, and not to the radiated heat from the hot equipment. One thermocouple of each differential temperature pair is located so as to be unaffected by pipe routing or tunnel temperature. High ambient or high differential temperature in the main steam tunnel will alarm in the control room and provide a signal to close the main steam line and drain line isolation valves. High ambient or high differential temperature in the HPCI and RCIC pipe routing areas will alarm in the control room and provide signals to close the appropriate HPCI and RCIC steam line isolation valves. A high main steam tunnel temperature or differential temperature alarm may also indicate leakage in the reactor feedwater line that passes through the main steam tunnel.

e. Temperature monitors in equipment areas - Dual element thermocouples are installed in the equipment areas and in the inlet and outlet ventilation ducts to the RCIC, RHR, HPCI, and RWCU system equipment rooms for sensing high ambient or high differential temperature. These elements are located or shielded so that they are sensitive to air temperature only and not radiated heat from hot equipment. High ambient and high differential temperature are alarmed in the control room and provide trip signals for closure of isolation valves of the respective system, ~~in the monitored area.~~ for RCIC, HPCI, and RWCU. For the RHR system, ~~only~~ high ambient and high differential temperature are alarmed in the control room.

f. Intersystem leakage monitoring - Intersystem leakage monitoring is included in the process radiation monitoring system to satisfy the requirements of that system, as described in Section 11.5.

g. Large leaks external to the primary containment - The main steam line high flow, HPCI/RHR steam line high flow, RCIC steam line high flow, reactor vessel low water level monitoring, and RHR excess flow discussed in Section 5.2.5.2.1, can also indicate large leaks from the reactor coolant piping external to the primary containment.

TABLE 5.2-11

SUMMARY OF ISOLATION/ALARM OF SOURCES MONITORED AND THE LEAK DETECTION METHOD USED

(Summary of Isolation Signals and Alarms (Note 3) and System Isolation vs Variable Monitored)

System Isolated \ Variable Monitored	Reactor vessel water level (Note 4) (Note 2)	Reactor pressure	Turbine building leak detection	MS tunnel ambient temp, high	MS tunnel differential temp, high	Flow rate, high	Drywell pressure, high	RHR equipment area ambient temp, high	RHR equipment area differential temp, high	RCIC equipment area ambient temp, high	RCIC equipment area differential temp, high	RCIC exhaust diaphragm pressure, high (Note 2)	RHR/HPCI steam supply differential pressure (High flow)	RCIC steam supply differential pressure (High flow)	RWCU process piping differential flow, high	RWCU equipment area ambient temp, high	RWCU equipment area differential temp, high	HPCI equipment area ambient temp, high	HPCI equipment area differential temp, high
Main Steam	2		I	I	I	I													
Recirculation (Sample Line)	2																		
RHR	3	I				I		~~I~~ A	~~I~~ A				I						
RCIC				I	I	I				I	I	I		I					
RWCU	2			I	I	I									I	I	I		
Balance of Plant	2						I												
HPCI				I	I	I		I	I				I					I	I

Note 1 I = Isolate alarm, and indicate (or record). A = Alarm and indicate only

Note 2 These leak detection signals are provided by other systems.

Note 3 An alarm is associated with each isolation ssignal.

Note 4 Numerals in this column correspond to reactor water levels as shown on condensate and feedwater Specification MPL-C34 and are levels at which isolation valves of the related system are closed.

ATTACHMENT XIII

QUESTION 460.4 (SECTION 11.4)

Acceptance Criterion II.2 of SRP 11.4 requires a process control program (PCP) for the solidification of solid radwaste. Section 11.4 of the FSAR states that the topical report WPC-VRS-001 (Rev. 1) dated May 1978, which has been approved by the NRC, will be used. Page 77 of this topical report states that the feedstream chemical composition and percent of solids in the feedstream are required to determine the end product. Tables V-1, V-2 and V-3 provide maximum radioactivity limits for the feedstream. In Section 1.8 of the FSAR, you take exception to Regulatory Guide 1.21 for determining the quantity and composition of solid waste. Provide the information on laboratory tests that will be conducted under the PCP to assure that the feedstream will be within the parameters in WPC-VRS-001 (Rev. 1) for chemical, physical, and nuclide quantity and composition; the frequency of feed sample testing; and the method that will be used to establish feed conditions. What type of laboratory/field instruction record sheet will be used within the PCP at HCGS. Address the 1% of oil limit on the feedstream. Specify the program to be used whenever a batch fails to solidify. Address the fire protection measures recommended by the topical report.

RESPONSE

In clarification of Tables V-1, V-2 and V-3 of the Topical Report WPC-VRS-001 (Rev 1) dated May 1978, the values provided are typical chemical and weight percent compositions, and maximimum estimated curie contents, and are not maximum limits.

The information on laboratory tests that will be conducted under the PCP to assure that the feedstream is within the parameters in WPC-VRS-001 (Rev 1) for chemical, physical, and nuclide quantity and composition and the frequency of feed sample testing will be provided by ~~January~~ September 1985.

To assure that the feedstream will be within the parameters of the PCP, a grab sample will be taken prior to processing each batch of liquid waste. This sample will be analyzed by the system operator for weight percent solids and specific gravity. The initial waste processing conditions are then determined by knowing the waste type (e.g., resin slurry or concentrates), the specific gravity and weight percent solids.

The laboratory/field instruction record sheet to be used within the PCP at HCGS will be provided by ~~January~~ September 1985.

The one percent oil limit on the feedstream is addressed in WPC-VRS-001 (Rev. 1) dated May 1978, pages 88-89.

ATTACHMENT XIV

The Hope Creek simulator is to play a vital role in the training of operations personnel in determining the extent and mitigating core damage. ~~It is expected to be delivered and accepted in the fall of 1984~~. Plant operating procedures and emergency operating procedures also play a vital role in adequate training on core damage mitigation.* ~~These are expected to be available by January 1985, therefore full implementation of mitigating core damage training is expected by January 1985.~~

Other Plant Personnel

NUREG 0737 Section II.B.4 requires that managers and technicians in the Instrumentation and Control, health physics and chemistry departments receive training in the use of installed instruments and systems to control and mitigate accidents involving a degraded core commensurate with their duties and responsibilities. Enclosure 3 to the H. R. Denton letter dated 3/28/80 and INPO document STG-01, Rev. 1 dated 1/15/81 identify those topics that should be included in these training programs. Plant operating and emergency procedures play a vital role in the training of core damage mitigation. ~~These procedures are expected to be available by January 1985.~~ Training to recognize and mitigate the consequences of core damage for the I&C, chemistry and health physics ~~managers and~~ technicians has been ~~will be~~ incorporated into the respective technician training programs. ~~by January 1985.~~

* Emergency Operating Procedures will be developed and implemented in accordance with the Procedures Generation Package which is to be submitted for NRC staff review in January 1985. The Procedures Generation Package includes a description of the training to be provided on the Emergency Operating Procedures.

Training for the I&C, chemistry and health physics management will be incorporated into the respective technical and management training programs ~~by September 1985.~~ prior to fuel load.