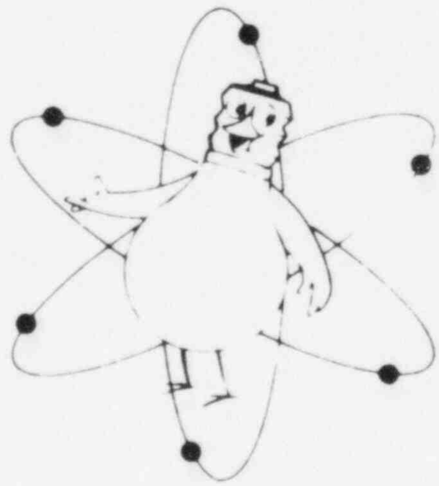
DRESDEN NUCLEAR POWER STATION
UNIT 2

Supplementary Information

to

Special Report
of
Incident of June 5, 1970



Commonwealth Edison
Company

QUESTION

1. On page VII (6) special report of incident June 5, 1970, it is stated that a new design temperature of 320°F has been established for the primary containment. Previously the design temperature had been established at 281°F. Accordingly, provide analyses to show that the various structural elements, including penetrations, can withstand the effects of the high temperatures and provide the results of analyses to show that this new design temperature is the maximum temperature which could be experienced within the primary containment. Include the effect of containment spray operation in the analyses. In addition for equipment within the containment which is required to remain operational during such an incident as occurred on June 5, 1970, provide appropriate test results or other applicable data that demonstrates such components can withstand the incident environment. The effects of a high design temperature on the allowable primary containment leak rate should also be discussed.

   Based on the data provided in the "Special Report of Incident June 5", it appears that the leak tightness of the primary containment could have deteriorated. Accordingly, discuss the measures you have taken or will take to assure that the leak rate of the primary containment is within the limits of the Tech. Spec. Your discussion should include considerations of local tests as well as integrated leak rate test of the primary containment.

ANSWER

Analysis has shown that the established design temperature of 281°F, in conjunction with the other variables such as pressure and duration, protects public safety with adequate design margin even though one specific variable (temperature) exceeds its design value. This conclusion is based on an evaluation of June 5 Dresden-2 event in which the containment atmosphere may have reached 320°F after about 1 hour, and in addition on an evaluation of a more general case in which the containment atmosphere is assumed to reach a maximum of 340°F.

This analysis of the containment at higher temperature does not imply that the design basis temperature of the D2 containment has been increased. The containment design temperature of 281°F was in fact used for containment construction. However, analysis demonstrates that the higher temperature condition imposes no significant compromise on the design margins which were originally based on the design temperature condition of 281°F, and in this sense the higher design temperature has been adopted.

Maximum Atmosphere Temperature

The maximum atmospheric temperature achieved in the D2 drywell will result

from a pure steam leak issuing into the containment. This peak super heat temperature is a function of reactor pressure and approaches a maximum of 340°F at a reactor pressure between 400 and 500 psi (this is readily demonstrated from the Mollier diagram). At rated reactor vessel pressure the maximum super heat temperature approaches 320°F. Needless to say, to approach the peak temperature of 340°F for any duration a very special reactor condition must be postulated where the reactor could remain at ∽500 psia and the drywell temperature at 50 psia. Large steam leaks that depressurize the vessel quickly would approach the peak containment temperature quickly, but in this case required valve action would occur within a short duration. Small steam leaks would depressurize the vessel slowly and result in a slower containment temperature rise as demonstrated during the incident.

The very low heat transfer coefficients associated with superheated steam (h of 1 to 5) will result in lower temperatures, and a significant time lag, in the temperature response of the drywell structure and components.

A general case in which the containment wall is postulated to be 340°F has been analyzed to demonstrate the adequacy of the containment.

SAT ?

Containment

1. Structure

The over temperature transient of 340°F at 20 psig does not produce significant primary containment vessel (PCV) stresses when compared to the design conditions of 281°F at a 62 psig maximum pressure.

The average membrane stress intensity analysis and formulae are dependent on pressure only.

However, the effects of temperature in the vessel calculations are considered separately, and these effects do not result in an appreciable increase in average membrane stress intensity for the temperature increase from 281°F to 340°F.

A. The increased temperature (340°F) case has been analyzed for the condition of expansion against the compressible material in the drywell. The safety factors for critical buckling at the design condition of 281°F and the increased temperature condition of 340°F are shown in the following table:

Temp °F	Pressure psig	Safety Factor
281	0	2.2
340	0.5	2.2
340	0	1.9

These results indicate that thermal induced loads are the same for 340°F with only 0.5 psig as for the design condition of 281°F @ 0 psig. Loads for 340°F at 0 psig are slightly greater and result in a slight decrease in safety factor from 2.2 to 1.9.

Activation of the containment sprays would eliminate the possibility of super heat temperature in the bulk of the containment while the sprays are operating. However, for steam leaks high in the containment above the containment spray sparger the possibility of superheat temperature could exist during spray operation. The sprays, however, will also have the effect of lowering the containment pressure thereby decreasing the peak superheat temperature.

If it is postulated that the containment spray was turned on after the drywell wall had heated uniformally to 320°F, the drywell pressure and wall temperature would respond as shown in Figure 1. After 10 minutes the containment pressure has dropped to about 1 psig and the wall temperature to 300°F. Again this combination of buckling load would be less than the design case. In fact, even if the temperature was still 320°F and the pressure as low as 0.2 psig, the buckling load would be no worse than the design case (281°F @ 0 psig). Finally as indicated in the incident report, at 320°F and 0 psig the safety factor compared to allowable loading is 2.0.

B. The vessel design condition of 281°F did not effect the allowable stress intensity range used in the vessel design because the material has the same value for a temperature range of -20°F to 650°F.

However, when using the yield strength for the design criteria, the material has an allowable stress limit value in 100°F increments. In the containment design, the 300°F limit was conservatively used in establishing the yield stress limit. When considering the 340°F temperature, this results in a 7.5% decrease in allowable yield stress limits. Review of the few areas where this was the limiting criteria in the containment design report indicated that in no case were allowable stress limits exceeded.

It is thus concluded that the Dresden-2 containment structure (design temp. of 281°F) provides adequate safety margin for public safety for the maximum steam superheat temperature of 340°F.

2. Penetrations

A. Piping Penetration

The maximum change in vessel differential expansion for this increase in temperature is 4.6×10^{-3} inches/foot, or 0.14 inches, based on the 31 feet which is the distance from the point of vessel embedment to the top of the CRD penetration bundle and encompasses the majority of the drywell vessel penetrations. This 0.14 inch increase in thermal expansion is not significant enough to cause any increase in stress in the penetration nozzles and is definitely less than the constructional clearance between nozzles and concrete pipe sleeves.

B. As a result of the Dresden incident, the electrical penetration seals saw a maximum temperature of 320°F, a maximum humidity of 100% for a maximum of less than one day. The incident caused a measurable leak on the inboard seal of the penetration seals. This leak was an order of magnitude less than that allowed by the Technical Specification. No leaks occurred on the outboard seal and hence no leaks occurred through the penetration seal and hence complete containment was maintained. This was verified by helium mass spectrometer leak checks.

Answer, Question 1 (continued)

3. Equipment

The safety components in the containment which must function for a LOCA have been tested at temperatures higher than the containment design temperature of 281°F. These are indicated in the following table:

	Component Test
Relief Valve	Actuator @ 300°F for 10 hours.
Main Steam Isolation Air Solenoid Valve	Solenoid valves @ 300°F for 10 hours.
Recirc System Valves (pump suction, discharge bypass, equalizer)	Limit torque valve operators @ 329°F for 3 hours.

Although these components have not been tested at the maximum temperature of 340°F, there is no indication that temperature limits were approached during testing. Indeed these components did survive the D2 incident of May 5 where the peak atmospheric temperature was calculated to be 320°F.

Therefore, there is no indication that a temperature of 340°F will adversely effect these components.
a cute negative statement.

4. Leak Rate

The total allowable leak rate of the primary containment vessel shall be less than that specified in the Technical Specifications for any temperature condition in the containment. The viscosity and ensity of steam at 281°F (saturated condition) do not differ appreciably from 340°F superheat steam at the same pressure. The viscosity remains nearly constant and the density decreases by approximately 10%. It's therefore concluded that the effect of the higher temperature or vessel leak rate characteristic will be negligible.

Conclusions

The design basis temperature of 281°F for the Dresden-2 containment need not be changed. It is concluded that the containment design temperature, pressure and duration provide adequate safety margins for public safety even though the maximum steam temperature could possibly exceed this design temperature.

A visual inspection of all penetrations was made and the only sign of degradation noted was with the electrical penetration. The damage occurred to the internal seal, and no leakage out of the containment has been found. This was verified by helium spectrometer leak checks. All of these seals have been repaired.

The initial integrated leakage rate was only 10% of the maximum allowed. There is no reason to believe that any significant change should be expected. To assure ourselves that the leak rate is still within Technical Specification limits, we are planning the following local leak rate program:

1. All electrical Penetrations.
2. All testable penetrations that have been disturbed as a result of work during the outage
3. Ten percent of all other testable penetrations.

Inspection item
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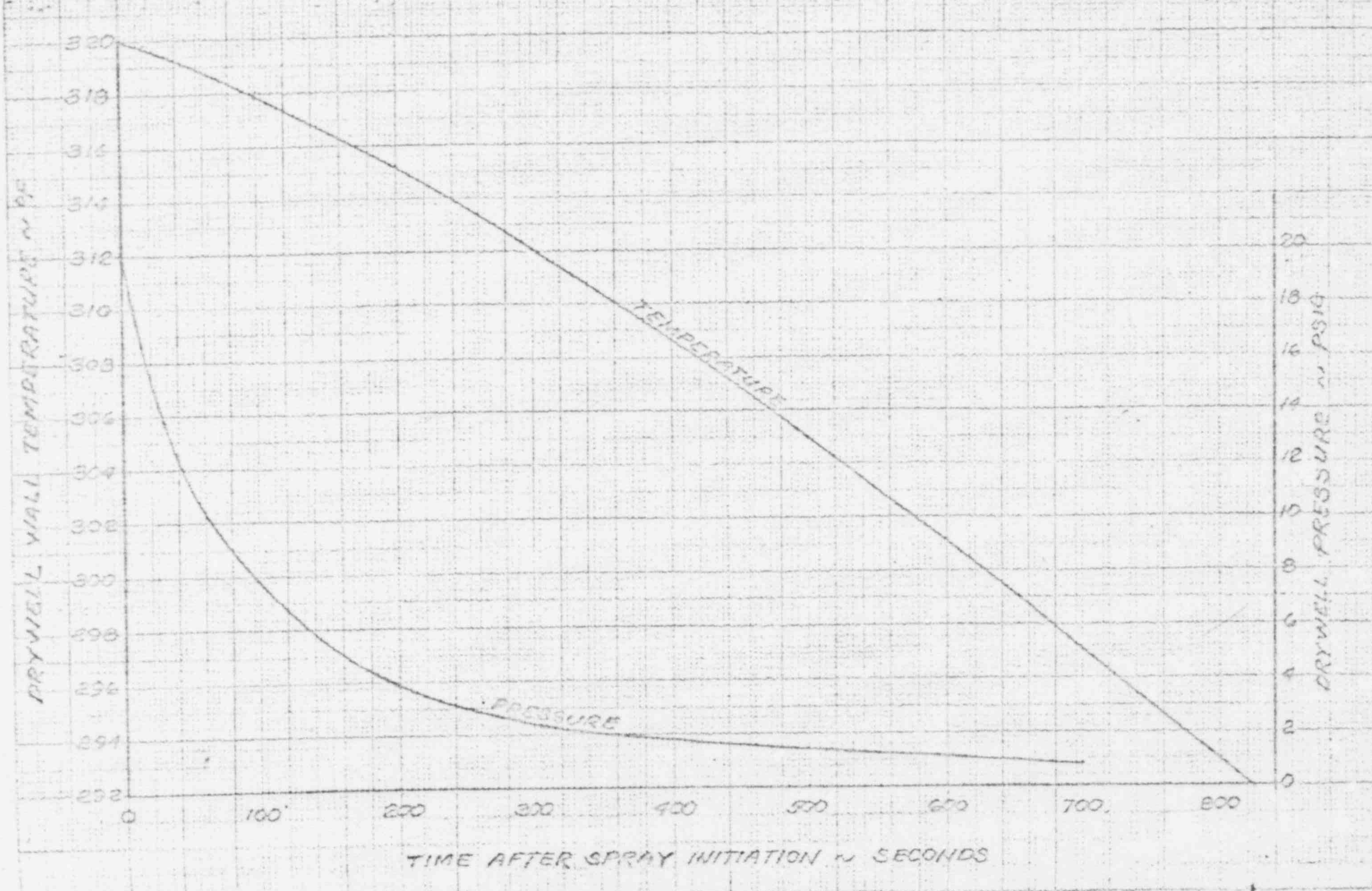
Answer, Question 1 (continued)

If any individual penetration test exceeds 25 percent of the Technical Specification for limit, necessary repairs will be made, and all remaining testable penetrations will be tested.

In view of the pressure and temperature conditions experienced and the test program planned, there appears to be no reason to delay the return to service of the unit by running a full scale integrated leak rate test.

FIGURE 1.

DRYWELL WALL TEMPERATURE RESPONSE



QUESTION

2. Following the incident the primary containment atmosphere was vented through the standby gas treatment system. The pressures and temperatures that the standby gas treatment system experienced during the venting operation are not stated in your report. These data should be provided and compared to the conditions for which the system was designed. In the event the design parameters were exceeded discuss the inspection and maintenance actions that were performed that assure that the system is now capable of performing its design function and that design parameters will not again be exceeded.

   Also describe the revisions to procedures or equipment that have been implemented to prevent use of this system until the containment atmosphere is known to be within design conditions for the SGTS. The desirability of appropriate interlocks on the vent line isolation valve should also be included in the discussions as well as your plans to require operability of necessary instrumentation to evaluate the containment atmosphere to incident conditions. Redundancy aspects of such instrumentation should also be discussed.

ANSWER

During the Dresden 2 incident of June 5, 1970 the normal reactor building ventilation exhaust and supply valves isolated and one train of the standby gas treatment system (SGTS) automatically started (the other train was in its normal standby status) as a result of high drywell pressure 6 minutes and 3 seconds after the start of the incident. Air was withdrawn from the reactor building (secondary containment) and a negative pressure was maintained in the building to minimize exfiltration.

20 min

About 30 to 40 minutes after the start of the incident, the second SGTS train was manually placed in service with the total flow through both systems controlled at 4000 CFM (approximately 2000 CFM per train). Shortly after this action, the 6" butterfly valve (2-1601-63) and the 2" bypass globe valve (2-1601-62) were opened and the drywell atmosphere was vented to the SGTS as shown on Figure 1. Calculations show that with steam in the drywell at 20 psig, critical flow occurred in the bypass valve so that the water vapor flow from the drywell was about 650 CFM. (The 1200 CFM reported in the incident report would only result if the design basis accident pressure of 48 psig had been reached). Since the total SGTS flow was about 4000 CFM, the water vapor from the drywell was being diluted with about 3350 CFM of air from the reactor building. The pressure entering the trains of the SGTS was near atmospheric and the maximum temperature after the SGTS heaters was less than 180F assuming the reactor building air temperature was 100F or less.

The piping leading from the drywell to the SBGT 24" header (line 7503) consists of piping which is rated according to the pipe diameter. The 2" bypass line is schedule 80, the 6" line section is schedule 40, the 18" section is 3/8" wall. All valves, fittings, etc. associated with these sections are 150 lb. rated. The 24" header section is 10 gage spiral welded light weight pipe meeting ASTM A211 and specified to be

leak tight at 10 psig. The associated valves, etc. are 125 lb. rated. Pressure in this later section was near 0 psig. Hence, no excess pressure was experienced.

The SGTS train was designed to operate at near atmospheric pressures, although it has been successfully pressure tested to 1-1/2 psig. All components in the SGTS are designed for continuous service at temperatures of 250F or higher.

Our analysis of the previously described events indicate that neither the design pressure or temperature of the SGTS was exceeded. The high efficiency particulate air (HEPA) filters were tested in-place using air generated DOP (dioctyl phthalate) particles about two weeks after the incident. Efficiencies measured were greater than 99.9% for all tests indicating that no damage had occurred to these filters. Because of the complexity of charcoal filter testing, it was elected to replace all charcoal filters in the interest of saving time. Following the installation of new charcoal filters, they were tested in-place using Freon 112. Efficiencies measured were greater than 99.9% for both tests.

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Inspection item

A procedure has been issued titled "Primary Containment Post Incident Cooling and Depressurization." This procedure specifies the conditions under which the containment may be vented after an incident and also details the procedures to be followed for post incident cooling and depressurization. This procedure states that:

(a) No venting of the drywell can be initiated by shift personnel action if <u>any</u> of the following exist:

   (1) Drywell pressure is greater than 2 psig.
   (2) Low-low reactor water level.
   (3) Drywell gross air sample activity exceeds Technical Specification stack release limits.

(b) Venting through the SGTS is permissable if <u>all</u> of the following conditions are met:

   (1) Reactor water level and make-up are normal.
   (2) Drywell pressure is less than 2 psig.
   (3) Drywell gross air sample activity is determined to be within Technical Specification release limits.

In addition, sections in existing Procedure 1600-II "Primary Atmospheric Control" relating to venting of the containment have been revised to refer to the new procedure mentioned above.

The containment isolation system prevents venting of the containment through the 18-inch line whenever the containment pressure is in excess of 2 psig. The 2-inch vent line is not included in the containment isolation system but is restricted in operation by the operating procedures as modified above.

The answers to question 15 specifies the instrumentation within the containment which monitors the containment atmosphere during an incident condition. The operability of the instruments will be verified prior to startup of Dresden-2. Redundancy of each instrument is not required,

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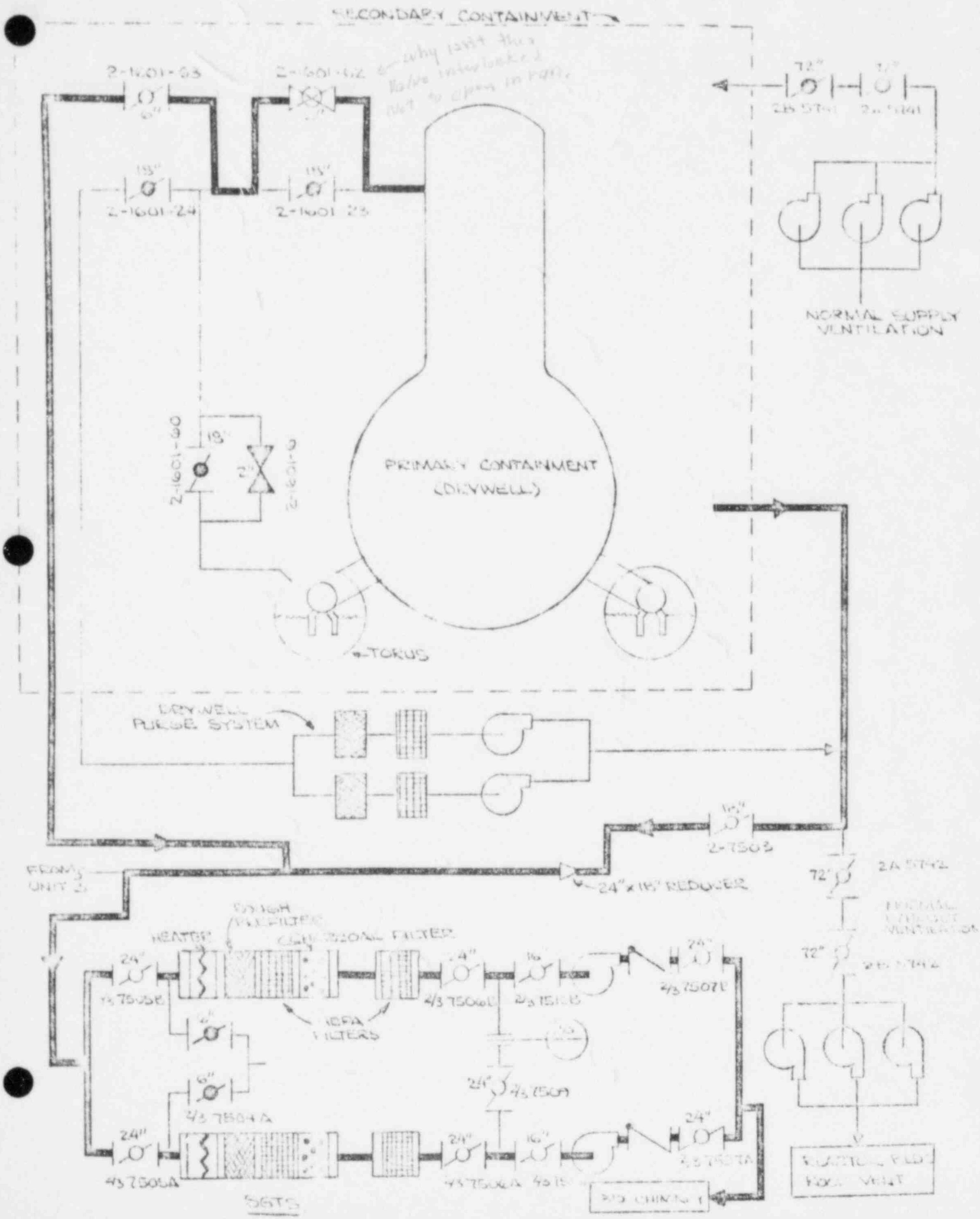
Inspection item

Answer, Question 2 (continued)

since operability checks can be made by comparing instrumentation in the torus with instrumentation in the drywell.

FIGURE 1. DRESDEN 2 DRYWELL VENTING [illegible] JUNE 5TH 1970



QUESTION

3. Page VII (7) of the June 5 incident report prov... discussion of the mechanisms which could have caused the safety valve to lift. It is stated that if waterhammer were the responsible mechanism for lifting the safety valve a pressure rise of 225 psi above system pressure is calculated to have occurred. If the other mechanism discussed on Page VI - 7, i.e., the possibility of a pressure pulse in the steam lines due to rapid condensation of trapped steam were responsible, what pressure pulse would occur? What is the maximum pressure pulse that would occur by either, or a combination of, these postulated events. Compare these pressures to the design and hydrostatic test pressures of the main steam piping system.

ANSWER

Condensation of trapped steam would not generate a significant pressure pulse during this transient because the required heat transfer surface conditions are not present.

If the steam and water are mixed such that a large heat transfer area is available, then all steam bubbles would not collapse at the same instant. The result would approximate the affect of cavitation, that is, considerable noise and vibration would occur but not a large amplitude pressure pulse.

If the steam and water are not mixed, then insufficient heat transfer area is available to cause condensation of a trapped pocket of steam at a rate that is fast enough to generate a significant pressure pulse. For this mechanism, the pressure pulse amplitude would be only 5 psi.

If a combination of steam/water mixing and water hammer had occurred, the pressure rise would be less than 225 psi because steam would increase the compressibility of the mixture which would reduce the surge pressure rise. The calculated maximum pressure rise of 225 psi could result in a maximum pressure of 1200 psi. Design pressure for the main steam line piping is 1250 psi, and the piping is hydrostatically tested at 1565 psi.

It is concluded that the maximum pressure experienced by the steam line was not only well within the hydrostatic test pressure, but was less than the piping design pressure.

QUESTION

4. Transient and accident analyses presented in the FSAR do not consider the compressible effects of the steam volume within the reactor vessel. Provide the results of analyses that show that such effects will not result in unacceptable consequences for the various accident and transient conditions.

ANSWER

Accident analyses in Section 6 of the FSAR do include the compressibility effects of the steam volume within the reactor vessel. Likewise the analysis of this incident included the effects of steam compressibility and such effects do not result in unacceptable consequences for this incident or for any other condition analyzed.

QUESTION

5. Describe the operational test program conducted for the isolation condenser. Discuss the results of this test with regard to demonstrating that the isolation condenser meets its design characteristics.

ANSWER

The isolation condenser was operationally tested at 10% and 25% reactor power. Both tests were made by manually initiating the isolation condenser and observing various parameters for the purpose of performing a heat balance. The heat balance verified that the isolation condenser heat removal rate exceeded the design rate by 150%.

During the test of the main steam line isolation valve, the automatic actuation of the isolation condenser was tested. The valves and alarms functioned properly.

During the startup of Dresden 2 additional tests will be conducted to insure the proper operation of the isolation condenser. These consist of the following:

1. Automatic initiation of the isolation condenser will be accomplished and a heat balance will be made to verify the heat removal rate.

   Inspection item

2. A simulated high flow signal will be inserted to verify the isolation valve automatic closure on both the steam supply line to the condenser and the condensate return line from the condenser.

QUESTION

6. During the incident the steam discharged through the safety valves impinged on various components within the primary containment. Provide a sketch that shows which components were exposed to the steam jet and provide resulting pressure loadings to which such components were subjected. Relate these loadings to those included in design evaluations. Include in this evaluation the change (due to a heated valve and springs) in set points that could have resulted from the safety valves that were subjected to the steam jet.

ANSWER

A sketch that shows which components were exposed to the steam jet is shown in Figure 6-1.

The various components which were subject to the jet stream have been evaluated to determine the resulting pressure loading to which these components were subjected. The maximum stresses developed on any of the components occurred on the relief valve which was subject to a 4000 pound impingement load. This loading results in a maximum stress of 80% of code allowable.

An evaluation was made to determine the effects of fluids impinging on the safety valves relief set points. The study is based upon the spring force varying with the torsional modulus of elasticity which varies with temperature.

In the event the original setting is at 200°F for 1220 psig, and the maximum temperature of the spring is 320°F as noted in the June 5 incident report, the set point changes to 1195 psig or a 2% set point change.

In general, the result of impinging fluids is to raise the spring temperature and reduce the set point which tends to be in a safe direction. Also, it is noted that the valve discharges have been rotated at Dresden 2 to assure that impingement will not occur as it did during the June 5 incident.

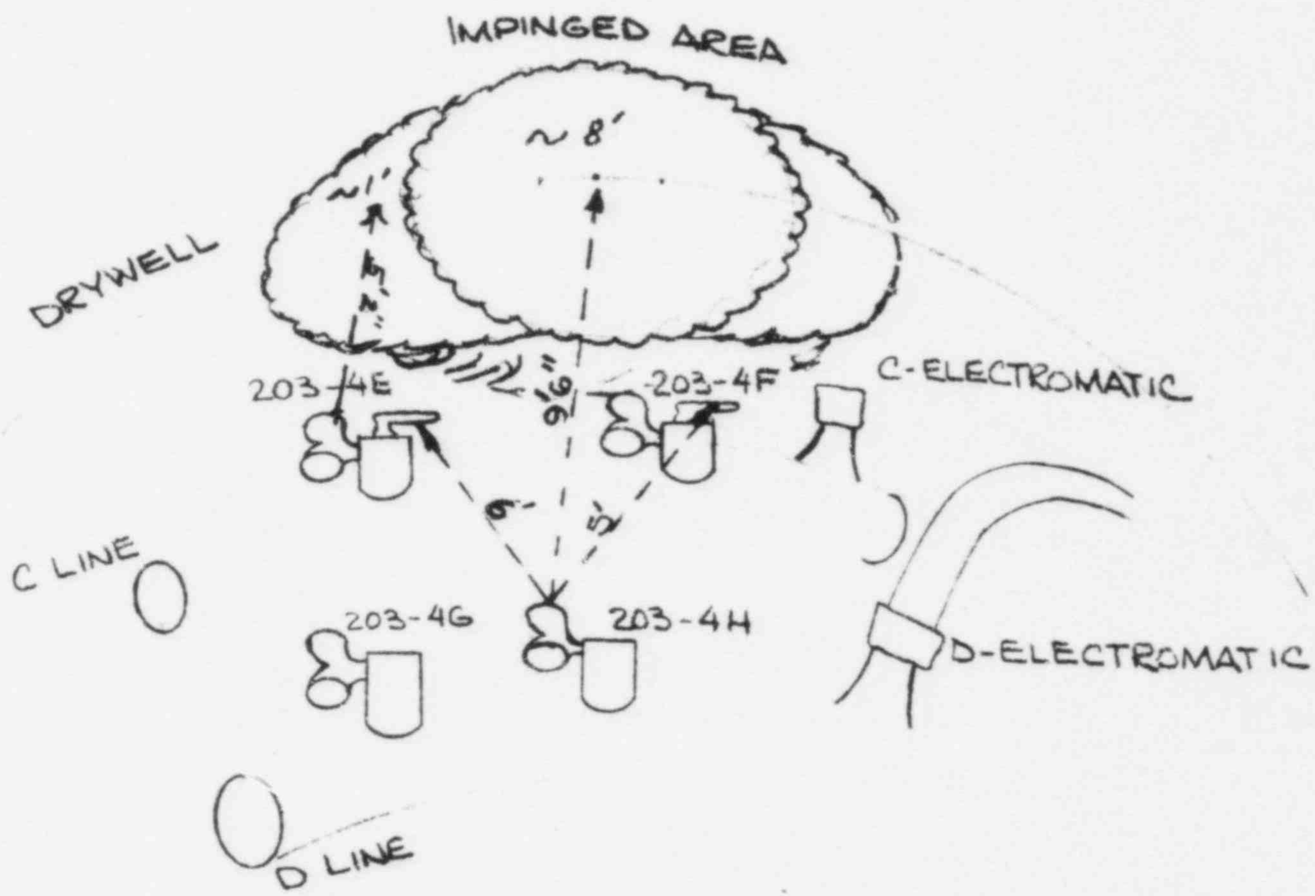

IMPINGED AREA
~ 8'
~1'
DRYWELL
203-4E
203-4F
C-ELECTROMATIC
6'
5'
C LINE
203-4G
203-4H
D-ELECTROMATIC
D LINE

FIGURE 6-1

QUESTION

7. Provide an evaluation of the feedwater controller operation during the incident conditions for both the automatic and manual modes. What is the minimum condition for which automatic (and manual) operation is possible?

ANSWER

As stated in the June 5 Special Report on the D-2 incident, the feedwater control was on automatic operation when the turbine bypass valve opened fully. About one minute into the incident, the operator noticed the low water indication and transferred the control to manual and opened the feedwater valves. The system was operating on an automatic load as per design up to this point. It must be noted that a higher than normal feedwater control valve leak rate with the valve fully closed had been experienced during other shutdowns. The operator placed the control in manual as a precautionary measure to open the valve to its full open position when he noticed the low water level indication without any indication of level being increased. If the control had been left in the automatic mode the system would have responded by opening feedwater valves, increasing the feedwater flow until the vessel level reached the control set point, then closing down to a minimum flow condition, which is the valve leakage rate.

In either mode, the valve leakage would have caused the vessel level to increase. During the present outage, the leakage through the 14-inch valves will be minimized by maintenance to the valve seats.

To prevent future high water problems of the nature experienced during the June 5 Incident, the Operating Procedures have been revised to include the following precautions:

(a) If the reactor water level cannot be maintained at +60 inches or less, the Feedwater Isolation Valves will be closed or the Feedwater Pumps will be tripped.

(b) The above precautions will be exercised individually or in combination as the situation requires.

In addition, the low suction pressure trip on the Feedwater Pumps has been changed from 140 psig to 120 psig in order to prevent spurious trips such as were experienced during the Incident.

The minimum flow condition with the small 4" bypass valve is almost zero, with a 300 gallon/min. design leakage through each 14" valve. These minimum flow conditions are effective in either automatic or manual modes. The system is being redesigned so that under low flow conditions automatic control will be achieved through the small 4" by-pass valve, which is being trimmed to give a maximum flow of 3780 gallons/minute. When the operator reaches the maximum control of the small bypass valve, he would then switch over to one of the 14" control valves, which will give the rated feedwater flow. The other 14" control valve will be in a standby condition with its isolation valve closed. The large 14" valves automatic

Answer Question 7 (continued)

control is generally limited to flows above 3000 gallons/minute. Therefore, it is a recommended procedure to operate on the 4" bypass valves at flow lower than 3000 gallons/minute.

QUESTION

8. Provide the results of the evaluation of the temperature transient experienced by the primary system during the June 5 incident with regard to any deviations from allowable cooldown rates and discuss the effects on subsequent usage factors.

ANSWER

During the incident of June 5, 1970, it is concluded that the reactor vessel was not subjected to any more severe temperature tansients than those considered in the vessel thermal stress analyses. Subsequent vessel usage is not affected to any greater extent than that which occurs during a normal vessel cooldown. These conclusions are based on the following:

The reactor vessel components which are most significantly affected by rapid changes in coolant temperature are the vessel nozzles, and the transition sections at the main closure flanges and vessel supports. During the pressure and temperature cycling and sequence of events shown on Fig. VI-1, Special Report of Incident of June 5, 1970, Dresden Nuclear Power Station Unit 2, the feedwater nozzles and control drive nozzles were not subjected to greater temperature changes than normally occur during reactor scram and normal shutdown operations. The significant difference (insofar as the vessel in concerned) is that the vessel water level was raised to steam nozzle elevation or above, thus possibly affecting vessel main flange transition temperatures during the initial incident filling operation, or subsequently during vessel cooldown. Referring to Fig. VI-1, the reactor recirculating pumps were in operation during the first 6 minutes of the incident which includes the period of initial vessel filling. During this time, the temperature of the coolant coming into contact with the main flange transition region of the vessel was at reactor recirculation loop temperature or higher. As indicated by the Fig. VI-1 plot, the coolant temperature change was only 40°F (from 540°F to 500°F) during the 6 minute interval when the vessel was first flooded. Subsequently, the vessel was depressurized to 275 psig (415°F sat.) at the end of 34 minutes. Vessel saturation temperature is indicative of the coolant temperature in the vessel flange region during this period. Thus the total change in coolant temperature in the vessel flange region during the 34 minutes of most rapid change was 125°F. This is less rapid than the temperature transient that occurs during normal shutdown cooling when the vessel is filled from normal water level to flange elevation and above during the vessel flooding period. During normal vessel flooding, the vessel flange internal surfaces are cooled by 100°F (from 390°F to 290°F) by the rising water in 14 minutes, which is at lease as severe a temperature transient as that encountered during the reported incident. With respect to the

vessel support transitions, the incident cooldown effects are not significant. Thus there is no reduction in vessel usefulness as a result of this incident beyond that which would be encountered during a normal shutdown.

In addition, the technical specification limits (for cooldown) an average rate of reactor coolant temperature change of 100°F/hour when averaged over a one hour period or allows a step reduction in coolant temperature of 240°F, provided shell flange to shell temperature differential does not exceed 140°F. The maximum shell flange to shell differential temperature was 90°F. Therefore, the technical specification was not exceeded. Since the temperatures remained within those analyzed in the vessel design there is no effect on subsequent outage factors.

QUESTION

9. We understand that you plan to perform additional leakage testing of each main steam lines isolation valve prior to Unit-2 startup. Describe and discuss your plans and programs for the conduct of these tests.

ANSWER

Tests are being conducted on the Main steam line isolation valves to assure that the valve leakage is acceptable. These tests consist of two separate phases. The first phase has been completed and was intended as a check for determining that no gross leakage existed. Each valve was subjected to a 94 foot head of water (approximately 41 psig). Results of this test verified that no gross leakage was being experienced; in fact no measurable leakage was found on seven valves and the other valve, 203-1B had a leak rate of 3/4 gallon per hour.

The second testing phase using air is now in progress. The test procedure is basically to close both the inner and outer isolation valves and establish a water head, approximately 40 psig, on the upstream side of the inner valve. Air pressure is then applied between the two valves to 24 psig and the pressure decay time is recorded. The lines are then drained and the valves again closed and air pressure applied between the valves and the pressure decay again recorded. The first step provides the outer valve leakage rate and the second step provides the total leakage rate of the two isolation valves in each of the four lines. The difference in values is the inner valve leak rate. Results of this air test to date are provided in Table I. In addition the water test results and the original preop test results are shown. As can be seen from the results to date, the leak rates are within those allowed by the Technical Specifications. It is therefore concluded that the leakage of all valves tested since the incident is satisfactory. It is not expected that excessive leakages will be found on those valves still to be tested.

TABLE I

Valve No.	Preop air test, CFH @ 50 psig	Water test gal/hr. @ 41 psig	Wet, outer valve air test, CFH @ 24 psig	Dry, combin. outer & inn valves air test, CFH @ 24 psig	Equivalent leakage ea. valve, CFH @ 25 psig
2-203-1A	< 1	0	N.A.	X	X
2-203-1B	2.82	3/4	N.A.	<.197	<.202
2-203-1C	<1	0	N.A.	X	X
2-203-1D	<1	0	N.A.	X	X
2-203-2A	<1	0	.135	X	.139
2-203-2B	<1	0	(1)	<.197	<.202
2-203-2C	2.5	0	5.29	X	5.42
2-203-2D	<1	0	.359	X	.368

(1) The single outer valve data was inadvertently not determined during wet, air tests. The combined leakage has been determined, however, and as shown in last column is < .202 CFH at 25 psig. Hence, due to this low value, no further testing required on 2B valve.

X - Test not complete.

N.A. - Not applicable to inner valves.

QUESTION

10. We have been made aware of certain operational difficulties due to temperatures encountered with the main steam line isolation valves and the corrective actions taken to assure their operability. Discuss the effects of temperature on valve operability including the reasons why you do not consider establishing a maximum temperature as a limiting condition of plant operation to be required.

ANSWER

Consideration was given to establishing a maximum temperature limitation for plant operation. However, it was concluded that such a limitation was not necessary for the following reasons:

1. Means are presently available in the control room for monitoring the steam tunnel and drywell air temperatures. Both areas have a temperature annunciator trip set point of 150°F. The steam tunnel area has 4 sensors with indication on panel 902-21 and annunciation on 902-3. The drywell has 6 sensors with indication on panel 902-21 and annunciation on 902-4. Hence, any approaching temperature problems can be noted.

2. Temperature tests within the steam tunnel area were conducted after the original pilot valve problem was determined to be partially related to temperature and after additional insulation was installed on several previously non-insulated components. Results of these tests indicate that the air temperatures in the area can be maintained below the 150°F region.

3. The actual temperature within the pilot and energizer valve bodies is the temperature of concern. Simultaneously with the tests of item 2 the actual valve bodies were instrumented and the resulting measured temperatures were found to be acceptable.

4. To assure that the plant does not have to be brought down due to higher than expected temperature, added blowers are being placed in the area. These blowers will substantially further reduce operating temperatures in the tunnel area.

5. The technical specifications already require testing the main steam line isolation valves twice per week. These tests should assure that any problems that might develop, no matter whether the temperature is the cause, or any other factor is the cause, will be noted by such tests.

A more detailed discussion of the operability problems encountered at Dresden 2 was submitted to the AEC by Commonwealth Edison Company letter on July 9, 1970. Briefly, the problem was attributed to a three facet situation. The air supply was contaminated, temperatures of the pilot valve bodies were found to be excessive and the pilot valves at Dresden 2 had very tight tolerances. The latter problem, when associated with either or both of the other problems, resulted in the pilot valve spools sticking in the air ported sleeves on occasion. The contamination, mostly oil, was removed from the system and subsequent filter tests show that a clean system now exists. The valve body temperatures have been reduced within the continuous operating design temperature by changes to the tunnel system, insulation of previously non-insulated components, and primarily by insulating the pilot valves from the main valve which removed the conductive heat path. All pilot valve spools and sleeves have been changed since the June 5 incident. The spool-sleeve clearances were all checked to assure that a later design version of the valves was provided and the valves operate freely and are clean. These are the installed valves.

In summary, all of the problems not only the temperature but the cleanliness and design problems as well, have been corrected. Means are available for determining temperatures in the areas of concern and means are available for annunciation of the temperatures in the event some problem should occur. The main valves are presently tested twice per week in accordance with the technical specifications. Should any problems begin to develop these present tests should adequately expose the problems.

QUESTION

11. As a result of an evaluation of data obtained from the Unit 2 vibration test program we understand that additional bracing in the jet pump risers has been incorporated into subsequent similar BWR plants. You have indicated that such action is not necessary, however, for Unit 2. Accordingly discuss the basis and justification that safe plant operation can be assured without the additional bracing. Any consideration to further action including inspection should be fully discussed.

ANSWER

The design change to the jet pump riser brace assembly that is being incorporated in other BWR plants was initiated for two reasons. First, in December 1968, a degree of difficulty was encountered during the installation of the vessel internals. The problem was basically one of alignment of the braces due to the vessel concentricity and riser pipe alignment. Preliminary design changes to consider means to alleviate the field difficulties started shortly thereafter. Secondly, during the Dresden-2 startup test program in late 1969 and continuing through mid-May 1970, it was noted that the jet pump riser assembly was getting somewhat higher vibration amplitudes than had been anticipated. Although the noted vibrations to that point were within the original criteria set forth for this component, it was decided to make changes on other plants in event that at sometime in the Dresden-2 program the amplitude should become unacceptable. This decision solved the original alignment problem and provided insurance against a more severe vibration problem developing at Dresden-2. The field disposition instruction for Dresden-3, for example, was issued in April 1970 and other plants were directed to incorporate the modified riser brace configuration at about this same period. In addition, a review of the vibration criteria for the component was started and a vibration test program was initiated using full scale brace assemblies. The criteria review showed that the riser brace amplitude limit was set low due to extremely conservative stress distribution assumptions. Specifically, a stress distribution corresponding to a prismatic bar was assumed when, in actuality, a simple linear distribution is more appropriate. The linear assumption resulted in revising the amplitude criteria more than twice the original 3.2 mil criteria.

Subsequent to the analysis review and the test program initiation the Dresden 2 tests continued. On May 20, 1970, a one pump trip test at 75% power and 100% flow was performed. One recirculation pump continued to operate at 100% flow while the other pump coasted down. The vibration data from this test showed that the original amplitude criteria had been exceeded during this one pump coast down transient with the maximum random vibration reaching 7 mils peak-to-peak at the time the shutdown pump passed through the speed range of 30-25% speed. The 7 mils attained the revised criteria amplitude.

Room temperature fatigue tests noted previously are currently in progress where five prototype brace assemblies will be tested to fatigue failure.

The first test has reached 10^7 cycles at 21 mils peak to peak amplitude without failure. Since 10^7 cycles is generally considered to be close to the endurance limit for the material, this initial test strongly suggests that the actual endurance limit of the part is at least as high as 21 mils. If one assumed a factor of 2 safety factor for this preliminary result, then the criteria could be raised to approximately 10 mils peak to peak. The tests of other assemblies will further substanitiate this evidence when they are completed.

The 7 mils peak to peak vibration observed during the Dresden-2 vibration test was a random amplitude rather than a steady state sinusoidal amplitude. This means that in the bench test with sinusoidal motion, 21 mils of peak to peak amplitude corresponds to greater than 21 mils peak to peak amplitude in the reactor since reactor motion is random rather than sinusoidal.

Upon completion of the cold flow test program which included operation at 100% mass flow conditions, a dye penetrant test was made of all ten of the riser brace assemblies. This inspection was done in accordance with Appendix IX, paragraph 360 of ASME Boiler Code Section III. No further inspections are presently scheduled since the vessel has been irradiated and the test results noted are demonstrating that the braces can sustain more cycles at a higher amplitude than could be attained in the transient mode of operation.

Based on the test results discussed, it is our engineering opinion that safe plant operation can be assured without additional bracing.

QUESTION

12. The consequences of failure of certain furnace sensitized stainless steel components were discussed in your June 9, 1970 letter. Our preliminary review of this information has indicated that certain safety aspects related to failure of the furnace sensitized stainless steel components have not been considered. These include:

a. What are your conclusions on the consequences of failure with regard to safety.

ANSWER

Based upon the analyses discussed in the referenced July 9, 1970 letter, the conclusion of the study is that no credible incident would result in circumstances where cooling of the core could not be accomplished. In the case of some reactors built before Dresden-2, a case was postulated where the shroud support ring was furnace sensitized stainless steel material. Under postulated conditions, failure of the shroud support ring could result in a loss of core cooling. In the Dresden-2 vessel, the shroud support ring is not sensitized stainless steel and, in fact, as stated in the July 9, 1970 letter, the only components in the reactor vessel which are sensitized stainless steel are brackets. The failure of these brackets could not lead to a loss of core cooling, and, therefore, failure of such brackets does not result in any compromise with regard to plant safety.

QUESTION

12b. What would be the consequences of failure of any of the specified furnace sensitized stainless steel brackets, i.e., the steam drier guide and support brackets, feedwater sprayer brackets, core spray line brackets, shroud heat guide brackets and the jet pump riser support pad?

ANSWER

The results of failure of the brackets and supports are discussed in Section II of the July 9, 1970 letter. Note the low stresses to which most brackets are subjected, which emphasizes their minor role as support and maintenance aids. Failure of any vessel internal bracket would not decrease core cooling, would not cause unsafe misalignment, and would not impede or prevent the proper function of the reactor safety systems.

QUESTION

12c. Since it appears that failure of the indicated furnace sensitized stainless steel brackets could lead to undesirable consequences, what courses of action are being considered to assure that the occurrence of failures would be highly unlikely? Your plans and programs in this regard should be discussed in detail.

ANSWER

Answers a and b above stressed the fact that failure of any FSSS brackets could not lead to undesirable consequences.

However, during the current maintenance outage and because the head was removed, it was decided that a maintenance inspection of the reactor internals was prudent. The inspection scope was not to be total, but it was to be sampling in nature. Nothing below core plate level was to be inspected unless fuel inspection results indicated that fuel problems had originated there.

Because of the nature of the outage, the critical path was determined by the fuel and LPRM change-out, and maintenance and inspection in the drywell. Internals inspection time was limited to available periods when the operations were either non-interfering or not in process, approximately 12 hours during reactor well fill (6/27) and 4:00-8:00 each following morning except Monday.

All visible horizontal surfaces were inspected using 7X binoculars and a 15-60X viewing scope. A few vertical and inaccessible features were viewed using a high resolution TV camera system which was shared with the Reactor Fuels and Reprocessing Department (RF&RD) personnel, on site for fuel inspection.

SUMMARY

In short, nothing of significant concern was found during this inspection.

The condition of possible feedwater sparger bracket pin rotation is not a problem when the likelihood of rotation and the consequences of this eventuality are considered.

All visible surfaces were coated with a black-gray oxide with no unexpected evidence of distress or abrasion which is witnessed by conspicuous bright scratches (spots on areas); lifting eyes showed these conspicuous signs, for example.

The use of the viewing scope for internals inspection was excellent for elevations down to the shroud top, when sighting from the refueling bridge. Below shroud level, its value is marginal due to thermal distortion of features. Also, only top surfaces can be seen.

OBSERVATIONS

A. Steam Dryer Assembly

The steam dryer assembly was examined both semi-dry and underwater with binoculars and lights. All steam carryover probes and instrument lines were intact with no evidence of distress. The dryer housing showed no evidence of distress or plate deformation due to the reported water hammer during the turbine trip incident. A premature report, that loose tubing was laying at the bottom of a discharge plenum, proved to be a vendor inspector's chalk mark. In fact, closer inspection revealed the number "5" chalked at the bottom of another outlet plenum. On the dryer top, a few words and numbers had been incompletely scrubbed off.

B. Shroud Head-Separator Assembly

The instrumentation lines and supports, separator braces, and the remainder of the assembly, in general, showed no signs of distress. A few bits of debris were scattered around on horizontal surfaces. This debris is believed to have originated in the storage pool floor and was washed up and onto these surfaces by personnel hosing the area to prevent air-borne contamination.

The shroud head bolt tees were noticed, in the majority of cases, to have bright areas consistently on the right hand side (tee aligned with a radial line in the unlatched position) when looking at the center of the shroud head assembly. These bright areas are indicative of contact areas or relative motion during disassembly with the shroud lugs; the rest of the tee surfaces were tinted with oxide as described.

C. In Reactor Inspection

The viewing scope, mounted on the refueling grapple platform, was used to survey the entire above core region, including visible (from above) portions of feedwater spargers, brackets and pins, vibration instrumentation, brackets, support and lines, guide rods and brackets, core spray lines and brackets, top guide, fuel, shroud head sealing surface, shroud head lugs, jet pumps, riser braces, and jet pump vibration instrumentation. No signs of distress were seen. The following conditions were observed:

1. In passing over the southeast (150°) feedwater sparger, a couple of bright spots were observed on the top surface on either side of a wedge block. Initially, they looked like angular chips of

metal, but these bright spots could be simulated by dragging one of the heavy drop lights across the feedwater sparger surface. In addition, later inspection in that vicinity with the high resolution TV camera showed that the bright spots were light reflections from the surface. Television scan of only this feedwater sparger showed the outlet holes and installation weld pull beads in considerable detail. As noted, everything appeared normal.

2. It was noted that one of the southwest (240°) feedwater sparger bracket pins was rotated ~45° clockwise from the "normal" position. It is not known whether this pin was installed this way or had rotated during service. In either case, pin lifting eye dimensions will permit rotation. However, consideration of geometry all but precludes the possibility of rotation. In the unlikely event that rotation occurs, wear is not considered to be significant or detrimental to the safe operation of the reactor.

3. Another location where bright spots were noted is on top of the northern (20°) shroud head guide rod bracket. These spots were brighter and sharper in contrast and appearance than those noted on the feedwater sparger; the appearance was more characteristic of chips of metal. The shroud head guide rod bracket upper surface is shielded somewhat by the guide rod top and the vicinity of the feedwater sparger brackets: precluding that these spots were scratches or marks from drop lights, etc. The largest of the approximately 5 spots noted was 1/8-1/4 inch wide, as estimated by comparative thickness to the shroud head guide rod bracket. The smaller spots were estimated to be 1/32-1/16 inches wide. It is hypothesized that these spots were scrapings from the guide rod left by the shroud head guide as it passed by during shroud head removal or stray debris washed into the reactor during reactor well filling.

4. An attempt was made to look with binoculars and the viewing scope, down into an empty fuel position while lowering a light as far as the fuel support casting. No details could be seen because of thermal distortion of the features.

5. A rather extensive attempt was made to lower the high resolution TV camera into the annulus between the shroud and RPV wall (at ~ 155° azimuth) to get a close-up look at the riser braces, the jet pump vibration instrumentation and jet pumps in general. The entire top portion of the riser, the jet pump mixer "ram's horn", the hold-down beams and studs, and part of the instrumentation lines could be seen in good detail.

   The features of the riser braces and the nearby instrumentation were visible, but just out of focus. There was no visible evidence of distress.

QUESTION

12d. We will need your evaluation of the consequences of failure of the indicated furnace sensitized stainless steel components in conjunction with an assumed loss of coolant accident sequence of either a recirculation line or main steam line rupture.

ANSWER

Section II of the July 9, 1970 letter analyzes the failure of the FSSS components during normal operation. The failure of each of the components for the LOCA case is the same as during normal operation except for the core spray line brackets and the jet pump riser support pads. Failure of a FSSS core spray line bracket would not affect operation of the core spray because three brackets would still be present, two of which are non FSSS. As provided in Section III, the stresses during core spray operation on these brackets are negligible. For the jet pump riser support pads, the failure of one of these during a LOCA could eventually result in fatigue failure of recirculation inlet thermal sleeve after the core has been reflooded. LPCI flow could be degraded a small amount, but both core spray flow (two 100% systems) and the remaining amount of LPCI flow would be sufficient to keep the core cool. Thus the failure of a FSSS component within the reactor vessel will not appreciably add to the consequences of a LOCA.

QUESTION

12e. Describe the results of your evaluation of breaks in the region between the reactor pressure vessel and the sacrificial shield in terms of pressure and jet impingment loads that could cause failure of the shield structures or cause portions of the shield plugs to become missiles that would effect engineered safety features necessary to mitigate the consequences of such an event; i.e. ECCS and containment structure.

ANSWER

The concern is the effects on the biological shield wall due to breaks in the annulus between the vessel and the wall. The wall consists of a 24 foot diameter circular cylinder attached to the vessel support pedestal and extending upward approximately 45 feet. This cylinder forms the outer shell of the annulus. The inner shell is the vessel wall and support skirt. The pedestal forms the base of the annulus with the top open to the drywell. The shield wall is 27 inches thick and consists of 27 inch vertical WF beam columns, tied together by horizontal WF beams and 1/4 inch plates. These plates are welded to the column flanges, both inside and outside; thereby forming a double walled shell. The shell is filled with concrete thus providing the shielding capability. The pipes leaving the vessel at elevations below the top of the shield wall penetrate the wall. The penetrations in the vicinity of the core utilize removable shield plugs which fit around the penetrating pipe. The plugs are provided in order to allow access to the pipe welds for purposes of in-service inspections. The Dresden plugs are two 9-inch thick steel plates attached to the shield wall by two vertical hinges, each 1-7/16 inch diameter and both halves locked in place by a 1-7/8 inch diameter locking pin. The remaining plug shielding is made of Permali blocks that are stacked into the shield wall penetration inside the steel plates.

This described configuration was conservatively analyzed to determine the capability of the shield wall to withstand pressures generated in the annulus. The criteria utilized to estimate beginning of wall failure is that only the two 1/4" plates, acting as a thin cylindrical shell resist the pressure forces with no credit for WF beam or concrete strength; that failure commences when shear stress at the fillet weld joining the plates to the column flanges reaches 90% shear yield; that shear yield is only 1.5 times code allowable which results in a failure commencement at approximately 18 ksi whereas shear ultimate is in the order of 40 ksi; and that the pressure differential is a constant load whereas such a case would continually decrease the differential as the drywell pressurized. For these assumptions the beginning of wall failure occurs when the pressure in the annulus reaches 41 psi. This pressure is a differential pressure from the annulus across the shield wall to the drywell space. The inner annulus wall is the vessel and its skirt. The skirt will withstand an external differential pressure of 150 psi before the onset of skirt buckling could commence. Hence the shield wall is the more critical component.

Answer, Question 12e (continued)

Estimates of pressures that could be generated in the annulus have also been made. The differential pressure is primarily a function of the break area and the annulus vent area. These parameters have been investigated parametrically to determine differential pressure as a function of the break to vent area ratio. The analyses assumed that 100% of the energy released through a given break size entered the annulus. Also, only the annulus top gap opening was considered as contributing to the vent area whereas a significant vent area is actually available near each pipe penetration through the shield wall. Results of this parametric study show that a break area of 4 square feet would result in 41 psi differential pressure for the Dresden vent area of 56.5 square feet. The largest pipe, the 28" recirculation pipe, has a cross sectional area of 3.65 square feet. Such size is equivalent to a 36 psi annulus pressure but no pipe of this size is considered as being a credible break size within the annulus. The minimum wall thickness for the various piping systems occurs at the safe-end joint to the piping. All other sections from this joint back to the vessel have thicker wall sections and therefore lower stresses. The largest line which has the safe end located in the annulus is the 4 inch jet pump instrument line nozzle. For all larger lines the double-ended line break results in the flow being directed into the drywell volume and not into the annulus. The 4 inch line that could credibly pressurize the annulus results in a pressure differential across the wall of 1 psi which is considerably less than the capability of the shield wall.

The effects of such an occurrence in terms of missile generation were also considered. For Dresden-2 the 1 psi will not generate a missile because the plugs are hinged and locked as described. Forces are not high enough to fail these attachments. Also, the effects of the shear drag forces created by the jet which results from an open ended break of larger lines within the shield wall penetration was considered. The 12" jet pump inlet is typical of these larger lines where the analyses show that depending upon such factors as roughness coefficient, degree of two phase flow and break size that a shear drag force of 1 to 8 kips is created on the plug which tends to move the plugs out of the shield wall. The 8 kips load plus an additional 12K load which is within the capability of the locking bars and hinges so that no missiles are generated in this manner.

There are innumerable other assumptions such as pipe break shape, location, size, etc. that can be postulated that could conceivably result in missile generation. Even if missiles are generated, it then becomes a matter of another multitude of possible assumptions as to the effects of the missiles. For example, it depends upon the friction coefficient and the radial pressure distribution forcing the plugs against the wall in determining the amount of energy absorbed in freeing the missile. Also, the break size determines the time factor that is to be used in evaluating the vessel source pressure. These variables enter into the missile effects, and then the angle of contact of the missile with the containment wall, for instance, and the shape of missile, and the size of the missile impact area are all variables determining effects. Assuming that all these variables occur in a worst case condition is considered incredible. However, for various reasonable combinations of circumstances, such a break would not result in circumstances where cooling of the core could not be accomplished nor where radioactive releases would exceed the AEC guidelines of 10CFR100.

QUESTION

13. Discuss the results of your recent non-destructive testing of furnace sensitized stainless steel nozzle safe-ends.

ANSWER

A complete in-service inspection of the Dresden-2 furnace sensitized stainless steel nozzle safe ends has been made in response to the desires of the ACRS and AEC. This inspection included a visual examination, a liquid penetrant examination, and an ultrasonic examination. The extent of this inspection exceeds the requirements of ASME Boiler and Pressure Vessel Code.

The following safe ends were examined:

- 2 - Recirculation outlet
- 10 - Recirculation inlet
- 1 - HPCI steam supply
- 1 - Isolation condenser steam supply
- 1 - CRD header
- 4 - Instrument
- 2 - Core spray
- 1 - Liquid control inlet
- 2 - Jet pump instrument
- 1 - Head vent
- 2 - Head instrument

This list represents all the furnace sensitized safe-ends on Dresden-2. No indications were found as a result of the visual examinations.

A very thorough liquid penetrant examination was made under the direction of personnel who had been involved in similar examinations at Nine Mile Point. Script type liquid penetrant indications were found on: two recirculation outlets, six recirculation inlets, the HPCI steam supply, the isolation condenser steam supply, one instrument, one core spray, and the two jet pump instruments. A linear type liquid penetrant indication was detected on the CRD header weld. This indication was not judged to be acceptable by the ASME Boiler and Pressure Vessel Code. The indication on the CRD header safe end was located in the pipe to safe end weld.

All script indications plus the CRD header safe end weld indications have been removed by removing metal with a burring tool in approximately 0.010 inch steps, until a "PT white condition" was observed. The depths of the script indications varied from .020 inches to .090 inches. The minimum wall thickness requirements were maintained after metal removal.

The CRD header is currently undergoing weld repair after metal removal and a PT white condition was achieved at a depth of .149 inches.

All safe ends were then cleaned with alcohol and treated with a 1/2% TSP solution. An ultrasonic examination of the pipe to safe end and the safe end to nozzle welds, plus the safe end itself was performed. No significant

change from the data obtained during the ultrasonic baseline inspection conducted prior to the unit going into service was observed. All indications detected were less than that obtained from a 3% of the wall thickness notch. All safe ends were then cleaned with demineralized water and recoated with the TSP solution. A flap wheel grind followed by a final liquid penetrant examination and a treatment with the TSP solution completed the inspection. Re-installation of the insulation is currently in process.

QUESTION

14. Discuss your plans and programs with regard to the performance of an independent stress analysis of the "as built" piping systems and observation of piping system during plant heat up prior to power operation of Unit 2.

ANSWER

while Building!

An independent stress analyses of the "as built" piping systems has been made. The flexibility analyses were done by Sargent and Lundy (except the recirculation piping). As the analyses were completed the computer output and isometrics were sent to GE San Jose for review. These reviews were dual purpose reviews; to assure the results were based on proper input assumptions, and to assure the results were in agreement with other analyses of similar systems. Operating pressure stresses were similarly accomplished. In addition, the flexibility results were provided to the hanger contractor, Bergen-Patterson, who determined the weight loadings and hanger requirements. These analyses were sent to Sargent and Lundy for review and approval. Likewise, seismic analyses were performed for the various systems in accordance with the methods described and discussed with the AEC seismic consultants. The Main Steam, Feedwater and Recirculation Seismic Dynamic Analyses were done by outside seismic consultants and reviewed by GE. Results of these various loading analyses were combined in accordance with B.31.1 piping code by Sargent and Lundy or GE.

As a result of the Nine Mile Point incident, a separate review of all systems connected to the vessel was performed jointly by Sargent and Lundy and GE. This review consisted of verifying that the inputs to the computer programs were correct and that the worst conditions were considered. Such factors as vessel movement, pipe temperatures, pipe size, pipe schedule and arrangement configurations were rechecked. In addition, a completely separate flexibility run of the core spray line was done by GE and the results compared very closely to the results obtained by Sargent and Lundy. Sargent and Lundy also ran the core spray line again after it was determined that the line configuration had changed slightly since the original analyses. It was this latter Sargent and Lundy run with which the GE run was compared. Further, the loadings resulting from weight, operating pressure, seismic and thermal factors were combined in accordance with B.31.1 code by both Sargent and Lundy and GE at the Safe-End locations. The results of these separate calculations were essentially identical. The GE calculations of the recirculation system piping was also rechecked by persons different than those who did the original calculations, and the results of the Dresden analyses, including inputs to the program and output forces, moments, deflections and stresses, were compared to similar analyses on other plants. The comparison showed excellent correlation.

The seismic input to the piping loadings were reviewed in detail during 1969 with AEC consultants, by GE consultants, by Sargent and Lundy and by GE.

The validity of the stress analyses have further been proven by actual physical inspection of the piping suspension systems settings in both the cold and hot positions. The inspection of the cold settings showed that the hangers were set properly in accordance with the design analyses. The hot inspection of the hanger settings verified that pipe movements were as predicated in the analyses. Therefore, the accuracy of the analyses have been verified.

? I didn't know hot checks were made after NMP!

A visual inspection of piping and hangers has been accomplished. The piping contractor, Bergen-Patterson, has reviewed settings of all hangers within the drywell. In addition, the AE, Sargent and Lundy, GE design and field personnel and CECO have made inspections to assure hanger settings are correct in both the cold and hot conditions and that there are no physical obstructions to the thermal movement of the piping systems.

QUESTION

15. Amendment 13/14 contained a discussion on the instrumentation systems that would be available to provide plant operators with necessary information regarding the environment within the primary containment following an accident or an incident. Describe your plans to assure that the necessary instrumentation will be installed and operable prior to resuming operation of Dresden-2.

ANSWER

The instrumentation discussed in Amendment 13/14 was installed and was available to monitor the June 5, 1970 incident. The drywell pressure was monitored to 5 psig and indicated and alarmed in the control room. The drywell temperature was monitored and could be read out in the reactor bldg. outside the drywell (temperature recording was available, but the recorder had run out of paper). Torus pressure was monitored by control room indicator over the range to 5 psig. Torus water level was monitored and indicated and alarmed in the control room. This instrumentation will be operable prior to resumption of operation of Dresden-2.

Since the June 5, 1970 incident, the primary containment monitoring instrumentation has been reviewed and revised to provide further information to the control room operator. The following represents the increased capability of the primary containment monitoring instrumentation.

Temperature and pressure monitors have been installed in the drywell which will monitor the complete range of accident conditions. The drywell will have temperature monitoring from 75°F to 350°F and pressure monitoring from 0 psig to 75 psig. All of these parameters will be indicated in the control room. These monitors will provide the operator with information regarding the environment within the primary containment during normal operation and following an accident or incident. (Emer. Power)!

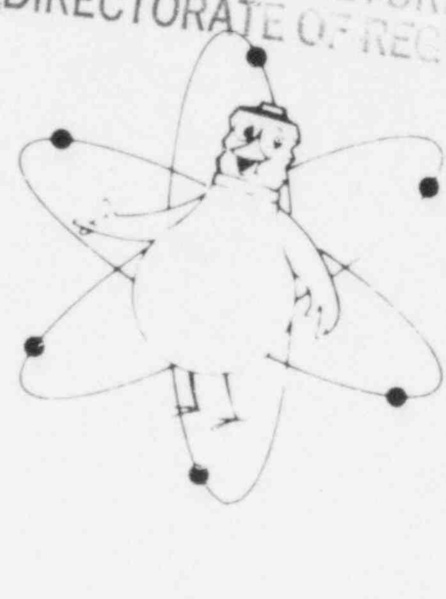
These monitors will be properational tested prior to startup of Dresden-2.

DRESDEN NUCLEAR POWER STATION
UNIT 2

Supplementary Information

to

Special Report
of
Incident of June 5, 1970



Commonwealth Edison
Company