VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

November 26, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 796
PO/DLB:baw
Docket Nos: 50-338
50-339

Dear Mr. Denton:

Subject: North Anna Unit 1 Cooldown
Incident of September 25, 1979

In response to your letter of September 28, 1979, the attachment provides additional information on the subject incident.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

cc: Mr. James P. O'Reilly

1398 300

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RESPONSE TO REQUEST FOR ADDITIONAL
INFORMATION RELATED TO THE
SEPTEMBER 25, 1979 EVENT

Enclosed are the responses to the information requested in your September 28, 1979 letter.

Should you have any questions or require additional information, please contact this office.

1398 301

Question 1. Perform an engineering analysis of the Sept. 25, 1979 event, identifying all of the significant initial conditions, and provide a comparison with events analyzed in Chapter 15 of the North Anna FSAR especially with regard to assumed initial conditions, boundary conditions and system failures.

Response: An engineering analysis was performed comparing the Sept. 25, 1979 event with the Chapter 15 events analyzed in the North Anna FSAR. This event most closely follows the accidental depressurization of the Main Steam System analysis - Section 15.2.13 of the North Anna FSAR. Accidental depressurization of the main steam system is classified in the FSAR as a Condition II fault, that is, a fault of moderate frequency. The specific challenge presented by such an event is whether the reactor would, because of the associated primary system cooldown, go critical and experience DNB and possible fuel damage. The case which bounds the consequences of such an event is the spurious opening of a steam dump valve. The nature of the September 25 event was much less severe than the transient shown in the FSAR, since the FSAR analysis includes highly conservative assumptions which were not present in the September 25 event. Significant differences include the following:

1. Initial TAVG = 570°F, vs. hot no-load condition. This provided additional stored energy, most of which was dissipated by the controlled steam dump operation which brought TAVG to the no-load value.

2. Initial power = 78%, vs. no-load in the FSAR. Thus decay heat was available to reduce the cooldown rate and heat the plant after cooldown was terminated.

3. Automatic feedline isolation, not assumed in the FSAR. This tended to make the rate of cooldown less rapid than in the FSAR.

4. Manual main steamline isolation, not assumed in the FSAR. This ended the plant cooldown.

5. Two high-head charging pumps on SI, vs. one (worst single failure assumption) in the FSAR. This, together with the termination of the cooldown and decay heat, induced a rapid repressurization, as opposed to the extended decrease in pressure shown in the FSAR.

In both cases safety injection was initiated when the cooldown had adequately reduced pressurizer pressure, and was sufficient to prevent the reactor from returning to a critical condition. Therefore, in both the FSAR case and the actual incident, DNB was precluded.

Figure 1 shows the transient response reviewed in the North Anna FSAR and also shows that the plant did not return critical. Therefore DNB did not occur. Figures 2 and 3 show the transient response utilizing the same computer codes and methodology as discussed in the FSAR and Reload Topical (WCAP-9272), however, the assumptions and sequence of events model the Sept. 25, 1979 event at North Anna 1. The transient results show that the pressure reduction, temperature cooldown and margin to criticality are all less limiting than the case analyzed in the FSAR. Thus the FSAR analysis remains bounding.

1398 307

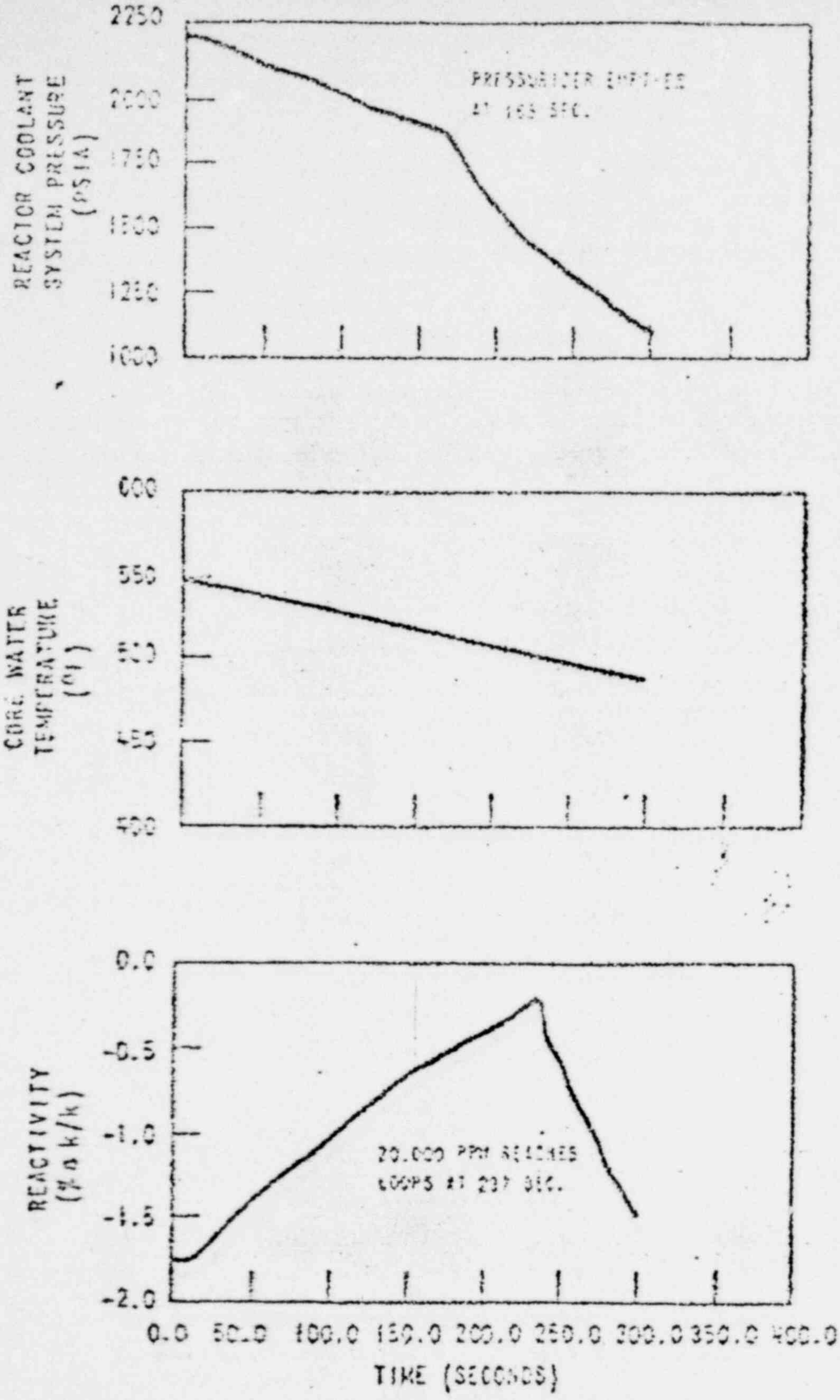


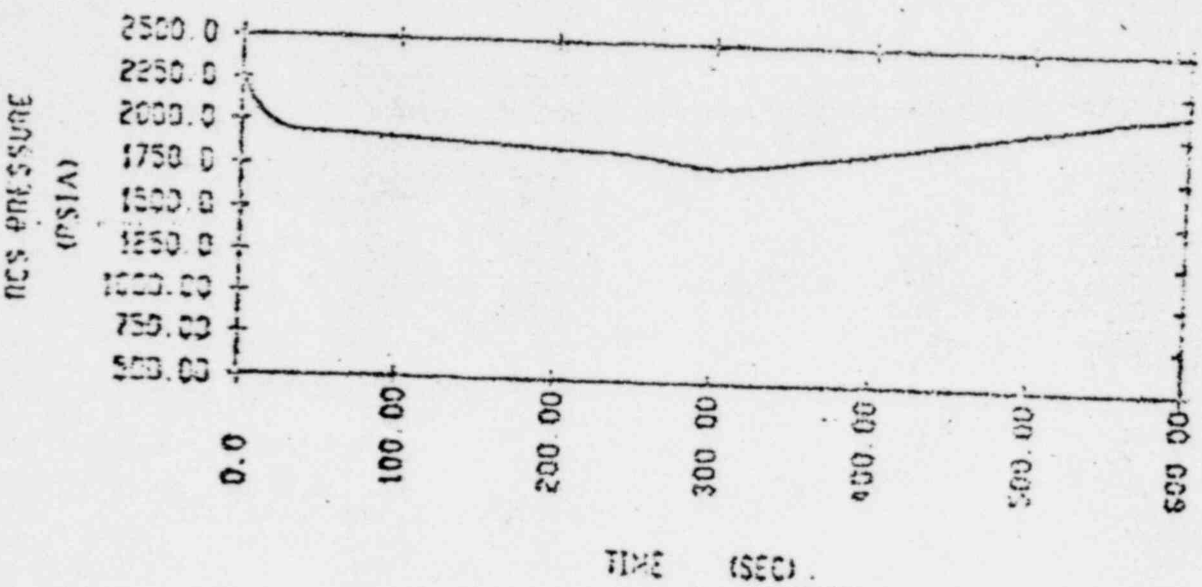
Figure 15.2-43 Transient Response for a Steam Line Break Equivalent to 202 lbs sec at 1020 PSIA With Offsite Power Available

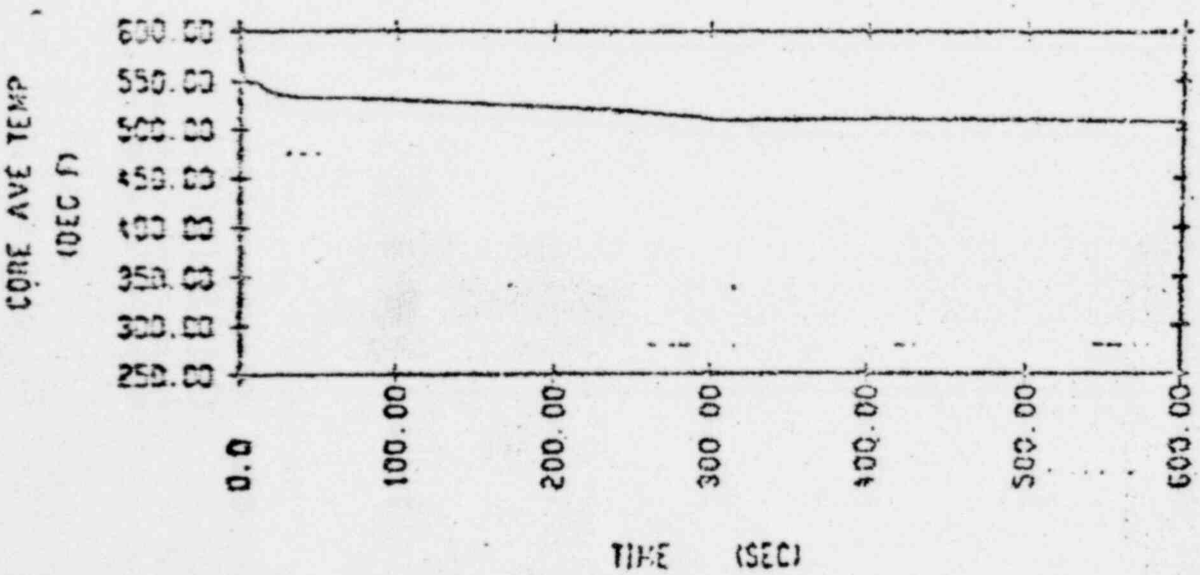
AMDB 57

POOR ORIGINAL

1398 303

TRANSIENT RESPONSE TO STUCK OPEN
STEAM DUMP VALVE

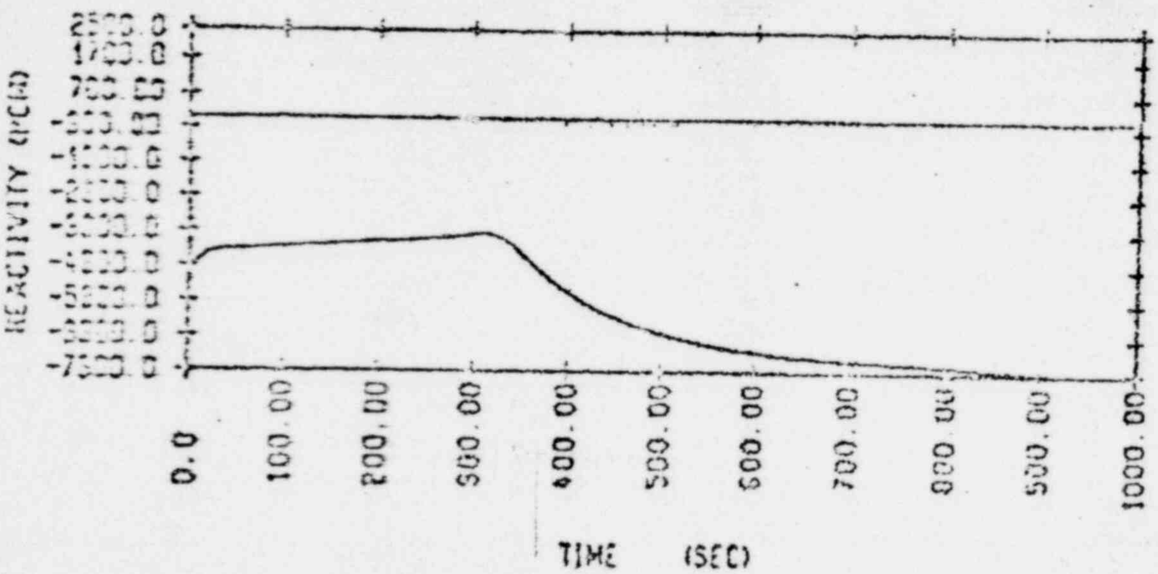




1398 304

MINIMUM APPROACH TO CRITICALITY
FOLLOWING A STUCK OPEN STEAM DUMP VALVE

POOR ORIGINAL



1398 305

Question 2. Identify all plant procedures (number, title, date) used at various phases of the event. Provide a summary describing the extent to which these procedures were used and why this use was considered appropriate.

Response

1) EP-1: Reactor Trip (Jan. 22, 1979) - completed entire procedure and is normally used following reactor trips.

2) EP-3: Main Steam Line Rupture (Sept. 13, 1979) - the procedure was only partially completed and was performed because the stuck open steam dump valve simulated a main steam line rupture.

3) AP-41: Safety Injection (Sept. 13, 1979) - the entire procedure was completed and performed because the depressurization of the RCS by the open steam dump valve resulted in the actuation of the ECCS.

4) AP-44: Loss of RCS Pressure (Sept. 13, 1979) - completed procedure as a result of the RCS depressurization following the failure of the steam dump valve to close.

1398 306

3. Describe specifically, the criteria that you used to determine that natural circulation was achieved following RCP trip. Identify the instruments relied upon and their readings. Provide the schedule by which formal written natural circulation cooldown procedures will be available to plant operators.

Response: Following the manual trip of the Reactor Coolant Pumps at 0616, the operators ensured that natural circulation was achieved by monitoring the following plant parameters:

A) Constant ΔT across the core of approximately 30°F as indicated by the Wide Range Cold Leg Temp (480°F) and the Wide Range Hot Leg Temp. (510°F)

B) Constant core temperatures as indicated by the Hot and Cold Leg Wide Range Temp. instrumentation.

C) Steam Generator Narrow Range Level indication returned and increased which ensured that the S/G tube bundle was covered.

D) Increasing Pressurizer Level and Pressure was observed.

At approximately 0629, RCP B was restarted which ended the period at natural circulation.

Formal written natural circulation cooldown procedures will be available to plant operators on or about December 3, 1979 which is prior to the restart of the plant.

1398 307

Question 4 Provide a description of the reactor coolant pump seal performance during the event.

Response

Prior to the 9/25/79 event, the C RCP at North Anna Unit 1 operated with some variation in leakrate; in general, the trend was low #1 seal leakage and high #2 seal leakage. This leakage variation was reviewed by Westinghouse and it was suspected that the #2 seal was not sealing properly. Continued operation was recommended unless further deterioration was indicated. Although a leakage variation existed, the C RCP seals continued to function throughout the 9/25/79 event.

Subsequently the C RCP was disassembled and the seals were inspected. The inspection revealed the following:

1) #1 seal assembly - in good condition and was reused.
2) #2 seal insert - some wear and was replaced.
3) #2 seal ring - operating surface worn away. Replaced
4) #2 seal runner - operating surface worn away for 45^{o} on it's circumference. Replaced
5) #3 seal ring - operating surface worn on a taper measuring 0 to 0.034 inch. Replaced
6) #3 seal runner - visual appearance was good, but was replaced (for the reasons discussed below).

The wear of the #2 and #3 seals was abnormally high. It is suspected that degradation of the aluminum oxide surfaces (of the #2 and #3 seal runners) may have occurred due to high ph levels during hot functional testing; Vepco was advised of this concern via a Westinghouse NSD Technical Bulletin. As a result of this high wear, the #2 seal was not sealing properly allowing more leakage from the #1 seal to pass through the #2 seal and resulting in the indicated low #1 seal leakage and the high #2 seal leakage. The increased #2 seal leakage was accompanied by increased #3 seal leakage. The hi hi air particulate alarm in the containment attributed to leakage by the #3 seal to the containment atmosphere. This leakage was observed by operators to be present only on RCP "C" the day following the event.

5. Provide an analysis of the actual pressurizer level, including the minimum level reached.

Response: Prior to the reactor trip, the pressurizer level was being maintained at a constant level of 48%. Following the reactor trip and the RCS depressurization, the level began to drop rapidly.

At approximately 1 minute prior to safety injection actuation, the level indication went off scale low. Subsequent computer models and mass balance calculations showed that the pressurizer was emptied at about 20 seconds prior to SI.

After the initiation of SI, the level began rising rapidly until a maximum level of 73% was achieved and maintained.

The pressurizer level began to return to normal when SI was secured and normal letdown and charging were established.

6. Indicate how many times the PORV cycled and what indications were available to the operator. Explain the apparent 2nd cycling of the PORV approximately 25 mins. after the first interval.

Response: A PWR cooldown computer model developed by Vepco indicated that the PORV cycled approximately 14 times during the first interval and 3 to 4 times during the second interval. The operator estimated between 12 and 24 openings and noted that the valve was not opening fully. Indications to the operator that the PORV was cycling included increased relief line temperature; valve position indication lights on the control board; increased temperature, pressure, and level in the pressurizer relief tank; and pressure indication cycling about the setpoint of the PORV on the pressurizer pressure strip chart.

The second cycling of the PORV was a result of the pressurizer volume surge, which occurred as a result of letdown being stopped. A summary of the sequence of events provides insight into this case. At 0648, about 36 minutes after turbine trip, volume control tank high level and high pressure alarms were actuated. As a result, letdown flow was first reduced, and then temporarily stopped at 0659. Pressurizer water level and pressure increased and the PORV again was cycled from about 0701 until letdown was restored at 0705. Pressure continued to be sensitive to level changes until about 0820, when the pressurizer heaters finally brought the water in the pressurizer up to saturation temperature. After that time the pressure was held near normal while the pressurizer water level was gradually reduced to normal over the next hour.

The pressure increase and poor pressure control is attributed to lack of normal spray and pressurizer water subcooling. A similar pressure response would also occur if significant amounts of non-condensible gas were present in the pressurizer. Subsequent analysis of a pressurizer steam space sample, however, did not show unusual non-condensible gas concentrations. Therefore, this effect is attributed to lack of normal spray and water subcooling.

1398 310

Question 7 Quantify the mass loss through the PORV and explain how this was determined.

Response:

The mass loss through the PORV during the transient was determined to be less than 3500 lb. The method used to calculate this value was based on assumption that mass loss is directly proportional to the change in pressurizer water volume while the PORV was cycling. This assumption is valid since the pressurizer liquid volume was highly subcooled over the duration of the repressurization phase (~ 520°F @ 2335 psig). This change in volume can be inferred from the strip charts showing indicated pressurizer level and pressure and hot leg temperature. From this data actual pressurizer level changes can be derived and thus the change in pressurizer volume for both PORV openings can be calculated. For the first opening a change in volume of 431 Ft^3 occurred which translates into a total steam mass of ~3000 lbs. For the second series of PORV openings a volume change of 65 ft^3 occurred which translates into ~450 lb. of steam release. A diverse check of the calculation can be performed by using the temperature rise in the Pressurizer Relief Tank. The total steam release is assumed to be condensed in the PRT and result in a rise in PRT temperature. Readings taken at 6 and 10 A.M. on September 25 show a 35°F change in PRT temperature. Assuming a release of 3450 lb. of saturated steam at 2335 psig, the temperature rise would be ~51°F. This therefore shows that the calculation is conservative. The primary reason for the deviation is due to the assumption that all steam releases are condensed over this four hour period and that no heat losses are assumed from the PRT walls.

1398 311

Question 8 Considering the nature of how plant parameters (e.g., pressure, temperature, pressurizer level, etc.) varied and were displayed to the operator during the event, indicate how and when you would have decided to terminate the high pressure injection based on the HPI termination criteria recommended by Westinghouse.

Response: As indicated by the strip chart recordings, wide range reactor coolant temperature T-Hot readings were always greater than 350°F throughout the event, as well as an indicated water level in the steam generators at a level to assure that the U-tubes were covered. Therefore, HPI termination would have relied on satisfying the pressurizer pressure and level criteria described in Westinghouse procedure E-2. An examination of the transient strip chart recordings and the post accident analysis logs indicates that the pressure criteria (>2000 psia) is satisfied after approximately 0618, or about 4 minutes after safety injection. However, the transient return of indicated pressurizer level during the event was such that the level criterion for HPI termination would not have been satisfied until 0628. At this time all of the Westinghouse requirements for HPI termination would have been satisfied, and the pressurizer PORV would have subsequently been allowed to close. This is to be compared with the actual pressurizer PORV initial closure time at approximately 0638. Therefore, the initial period during which discharge from the pressurizer PORV occurred would have been approximately 8 minutes, versus that actually recorded during the event of 18 minutes.

Since the North Anna event, Westinghouse has proposed revised HPI termination criteria for the E-2 Instruction which modifies the pressurizer level criterion to permit HPI termination in the absence of abnormal containment indications at a level of 20% span. Based on this criteria, HPI termination during the event could have been accomplished at approximately 0619, thus preventing opening of the pressurizer PORV.

Question 9 Indicate incore temperature readings taken during the event. Provide details as to magnitude, time and location.

Response: Vepco currently has no provision to automatically record incore temperatures following a trip and since the operators were responding to the plant's transient, no incore readings were taken during the natural circulation phase of the event.

A computer program which will record incore T/C temperatures after a plant trip is currently being developed.

Question 10 Provide the extent to which you consulted with Westinghouse during and immediately following this event.

Response: Westinghouse site Systems Engineer was notified of the transient when he reported for work on the morning of the event. The Systems Engineer was requested by VEPCO to contact Westinghouse to perform an evaluation of the reactor vessel cooldown.

1398 314

Question 11. Provide a detailed chronology of significant events during the period from the initiating event at approximately 0544 hours through return of activity in the auxiliary building to below MPC limits.

Response:

0544	LP HX 5B Drain Dump Valve begins cycling
0609	Turbine Trip on High 5B HX Drain Level
0609	Reactor Trip
0610	Aux. FW pumps start on S/G Lo-Lo Level
0610	Pressurizer Pressure 2000 psig
0611	Main FW pump C manually tripped
0611	Letdown Isolation on Low Pressurizer Level
0611	PZR pressure 1885 psig
0611	PZR level 7.4%
0612	Volume Control Tank Low Level Alarm
0613	PZR empties
0614	Safety Injection initiation on Low PZR Pressure
0615	FW Pump A tripped on SI signal
0616	Main steam trip valves manually closed
0616	All RCP's manually tripped
0617	PZR level returned to 9%
0618	PZR pressure 2160 psig
0619	Secured SI signal
0619	Charging pump B secured
0620	PZR PORV starts cycling
0621	FW pump C started
0625	Auxiliary FW secured
0627	Started establishing letdown
0629	Reactor Coolant Pump B started
0631	PZR level 62%
0633	S/G C narrow range level returns on scale
0635	Charging flow realigned & letdown initiated
0637	PZR PORV stops cycling
0646	VCT high pressure and level 69.9 psig and 85.7%
0647	VCT relief valve lifted
0652	First Hi Radiation level alarm - Ventilation Vents
0659	PZR pressure 2320 psig
0659	PORV begins cycling
0704	PORV stops cycling
0716	VCT pressure returns to normal
0730	Aux. Bldg. restricted to access
1000	Aux. Bldg. returned to normal access
1005	Plant Parameters at normal

1398 315

Question 12. Your alarm typewriter printout indicates that for at least 1 hour prior to the turbine trip, the containment sump level continually cycled to the alarm setpoint. Why?

Response: The cycling containment sump level is believed to be caused by some leaking check valves downstream of the containment sump pumps.

If these valves are not seating correctly, they will pass water through them and back into the sump. This will result in frequent high sump level alarms and near continuous pumping.

As a result of the SI signal, the containment sump discharge was isolated. During the event no apparent increase in containment sump level was observed which indicates that there was no leakage of coolant fluids inside the containment prior to or during the transient.

The check valves are currently being investigated and if needed will be repaired.

1398 316

Question 13. Identify the number of VEPCO personnel working in the control room during the first 30 minutes of the event.

Response: At the time of the reactor trip and during the first 30 minutes of the event, the following personnel were present in the control room: the shift supervisor (SRO license), 2 reactor operators (RO licenses), and an assistant control room operator trainee.

Approximately 15 minutes into the transient the Station Manager arrived in the control room.

1398 317

Question 14. State whether the site emergency plan was activated in any form.

Response: Under the guidelines of the Vepco Emergency Plan Implementing Procedure, it was decided not to activate the site emergency plan.

This decision was based on knowledge of the implementing procedures which deals with releases of radioactive material. The radioactive releases which occurred following the transient were well below the limits established by 10CFR 20.403 (a) and (b).

Also the shift supervisor decided that there was no physical danger to any of the plant workers during or following the transient.

Therefore, the decision was made not to implement the site emergency plan.

Question 15: Describe the extent to which the events which occurred at North Anna had been simulated at the Surry Simulator and demonstrated to North Anna operators, prior to Sept. 25, 1979. Discuss the training which North Anna operators have received at the Surry Simulator on tripping RCP's, natural circulation and Bulletin 79-06B HPI requirements.

Response: The combination of the events which occurred at North Anna during the Sept. 25, 1979 cooldown transient had not been simulated at Surry prior to the event. The North Anna cooldown transient was partially simulated at the Surry Simulator after the transient.

The problems encountered are as follows:

1. No capability to fail a steam dump open. (All three atmospheric dumps were used instead.)
2. Whenever the simulator pressurizer loses level indication, pressure drops to about 1000#, while the actual transient showed pressure staying above 1750#. (Simulated by not allowing pressurizer to empty.)
3. Pressure response at the simulator did not follow the transient except for the first two to three minutes. Once pressurizer level increased enough to start cooling the liquid temperature, the pressure indication on the simulator dropped instead of increasing to the PRV setpoint. Pressure stayed low until the pressurizer approached a solid water condition. The PZR appears to be modeled as a saturation vessel with respect to the liquid space temperature.
4. The pressurizer liquid and vapor space temperatures at the simulator did not show the large ΔT experienced during the actual transient.
5. The release to the Auxiliary Building was not simulated.

The cooldown and pressure drop portion of the transient was simulated. Natural circulation indications were approximately the same as experienced on the transient. Once the cause of the excessive cooldown was isolated, the simulator response deviated drastically from the pressure response actually experienced.

Due to the simulator response not being like the actual transient, there has been no specialized training, on that transient at the Simulator.

The transient has been reviewed with the operators currently in training for Unit 2 licenses. The remaining operators have reviewed the LER and will receive additional training during the month of December.

North Anna operators have received several training sessions devoted to Natural Circulation, tripping RCP's, and procedure changes, HHSI securing criteria, void and hydrogen formation. This training package has been supplied to Operator's Licensing Branch of the NRC as a pre-requisite for Unit 2 licenses. All licensed operators have successfully scored greater than 90% on tests related to this information. All training and testing has been reviewed by OLB and found to be more than satisfactory.

Question 16. Explain in detail the uncontrolled reactor coolant activity release directly to the auxiliary building.

a. Include why it happened, how it was detected, its release path(s), how long it continued, the amount, type and form (liquid, gaseous, particulate) of activity released, personnel exposures at the site and potential dose rate at the site boundary and beyond.
b. Given the inadvertent operator error, equipment failure, or combination thereof, involved on September 25, 1979, state whether an uncontrolled release would have been prevented had the piping to the process vent from the high level waste drain tanks been installed as called for in the plant design rather than in the as-found conditions.
c. If the answer to b above is in the negative, propose a design modification that will prevent a future uncontrolled release of activity outside containment.

Response: When normal letdown was established after Safety Injection, the operating charging pump was still aligned to the RWST instead of the VCT. This caused the level in the VCT to rise, at 81% level LCV-1115A diverted all letdown to the gas stripper. High level in the gas stripper system caused the inlet trip valve to the stripper to close. This isolated letdown and letdown pressure began increasing until the low pressure letdown line relief valve opened (200 psig setpoint). This diverted letdown to the VCT increasing the level and pressure until the VCT relief valve opened (75 psig setpoint) at 0647. This initially dumped H_2 and radioactive gases to the High Level Liquid Waste Tank (HLLWT). The overpressure condition was not immediately diagnosed and the VCT went water solid and began relieving water to the HLLWT. The HLLWT air sweep to process vent flange for Restricting Orifice LW-104 (ROLW104) was disconnected which allowed the gases to leak directly into the auxiliary building atmosphere. The gases were ultimately discharged to the environment via the plant charcoal and HEPA filters through the auxiliary building ventilation vent.

The releases through the auxiliary building ventilation vents provided the first control room indication of Hi Radiation Alarm at 0652. The VCT relief valve continued cycling until 0704, when the gas stripper was restored and letdown returned to normal.

The following information provides the amount, type and times for auxiliary airborne activities.

TIME	ELEVATION	EXPLANATION
0700	274'	100.96* Times MPC. Principle Nuclides involved were Xe 133 & 135 with some Kr 85 and RB 88.
	259'	155.68* Times MPC. Principle Nuclides involved were Xe 133 & 135 with some Kr 85 and RB 88.

1398 321

TIME	ELEVATION	EXPLANATION
0800	274'	1.12* Times MPC. Principle Nuclides involved were Xe 133 & 135 with traces of Rb 88.
0900	259'	6.01* Times MPC. Principle Nuclides involved were Xe 133 & 135 with traces of Rb 88.
1000	259'	0.68* Times MPC. Principle Nuclides involved were Xe 133 & 135 with traces of Rb 88.
1030	259'	less than 0.1 times MPC. Principle Nuclides involved was Rb 88. All reads after 1030 were less than 0.1 times MPC.

*This value represents the total submersion hazard involved with the total of all isotopes.

Perimeter TLD's were pulled and evaluated. No radiation exposures above background were observed in the 14 TLD's in the downwind direction on the perimeter fence.

Total Noble gas releases from ventilation vents A and B and the process vent amounted to 4.7 E-02% of the release rate limit of noble gases.

Total personal exposures involved 5 plant workers and the maximum individual dose was 7 mrem.

After reviewing the plant design, a release of gases from the VCT to the auxiliary building would not have been prevented had the piping to the process vent from the HLLWT been connected.

Calculations show that the volume of gases in the VCT prior to the opening of RV-1257 and the rate of reactor coolant letdown into the VCT during the transient would cause a discharge of gases into the HLLWT at a significantly greater rate than the 6 SCFM air sweep of the process vent system. This condition would result in the gases migrating out the LLLWT overflow vent into the auxiliary building.

The sizing of both relief valves was verified to determine their acceptability for overpressurization protection. This analysis was done in accordance with ASME Code Case N-94, Determination of Capacities of Liquid Relief Volumes.

As discussed above, if RV-1257 opens, the initial surge of high pressure gas will overload the air sweep system and a radioactive release to the auxiliary building would be probable. Since the liquid waste tanks are designed to handle liquid, a design is being evaluated such that the relief path from the VCT be repiped such that liquid would be released instead of gas. By making this modification, the only gas accumulation would be from outgassing as the liquid pressure is reduced from 75 psig to atmospheric pressure. The expected outgassing rate should be within the capacity of the air sweep system.

This modification will not impair the overpressurization protection of the VCT because the valve has been designed for liquid service and will pass the required capacity of liquid.

Question 17: The FSAR indicates that a high level in the VCT will alarm in the control room and divert the letdown stream to the boron recovery system. Did any part of this alarm and diversions system activate prior to or during the time that the VCT relief valve was open? Would such activity affect the release in 16 above?

Response: Prior to the event, an RCS dilution of 60 gpm was being maintained to reduce the boron concentration in the RCS. VCT level control valve LCV-1115A was in the automatic mode of operation and partially diverting the letdown stream to the gas strippers.

When the reactor trip occurred at 0609, the VCT level started decreasing which resulted in LCV-1115A diverting all letdown flow to the VCT.

At 0627, operations began to re-establish normal letdown and the VCT level began to rise. By 0646, the VCT level had risen to 86% level at which the letdown flow was being fully diverted to the gas strippers (an 81% of span signal will divert letdown to the stripper).

When the letdown flow was diverted to the gas stripper the water level in the stripper began rising to the Hi-Hi Level setpoint of 88% of span. Normal action at this time would be an auto start of the stripper discharge pump (1-BR-P-10A) which would transfer the water to the boron recovery tanks. Since the stripper was in manual operation, the pump failed to start and resulted in the stripper inlet valve (TV-BR-111A) tripping closed. At this time letdown flow was blocked which resulted in a pressure increase to the relief valve setpoint limit of 200 psig. Full letdown was diverted to the VCT where pressure increased to the VCT relief valve setpoint of 75 psig. The relief valve opened which relieved water to the High Level Liquid Waste Tank (HLLWT).

Meanwhile, the divert valve (LCV-1115A) remained in the divert to stripper position during the entire time in which the VCT relief valve was opened.

With the present design, the activity released to the auxiliary building would have occurred as long as letdown was being maintained with the inlet valve to the gas stripper closed.

1398 323