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

OFFICE OF THE
CHAIRMAN

March 22, 1982

The Honorable Morris K. Udall, Chairman
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, DC 20515

Dear Mr. Chairman:

This is in response to the questions posed in your February 5, 1982 letter relative to the recent event at the R. E. Ginna Nuclear Power Plant. Our responses to your questions are enclosed.

As a consequence of this event, I, too, have questions on the incident and its generic implications and have, on January 29, 1982, requested the staff to establish a Task Force to review and evaluate the Ginna incident. An interim report from that effort is expected to be completed this month and may provide detailed answers to some of your questions. The remainder of your questions are addressed in the enclosure.

Sincerely,

Nunzio J. Palladino

Nunzio J. Palladino
Chairman

Enclosure:
Responses to Questions

cc: Rep. Manuel Lujan

QUESTION 1. What is the primary significance of the Ginna incident?

ANSWER

The primary significance of this event is that it apparently occurred without advance warning and challenged the ability of the plant and operators to respond in a safe manner. It also points out the inadequacies in the steam generator inspections; i.e., the licensees do not inspect the secondary sides of steam generators, with the exceptions of a few plants that have suffered extensive tube denting and support plate cracking. The safety objective in such an event is to prevent fuel damage and to allow only minimal releases of radioactive materials to the environment. The tube failure, whether it be the result of chemical or metallurgical reasons, or some type of mechanical unloading mechanism, has not yet been determined. The failure mode and the plant and operator responses will be addressed in the NRC Task Force 45-day report.

QUESTION 2. What was the leak rate through the break as a function of time?

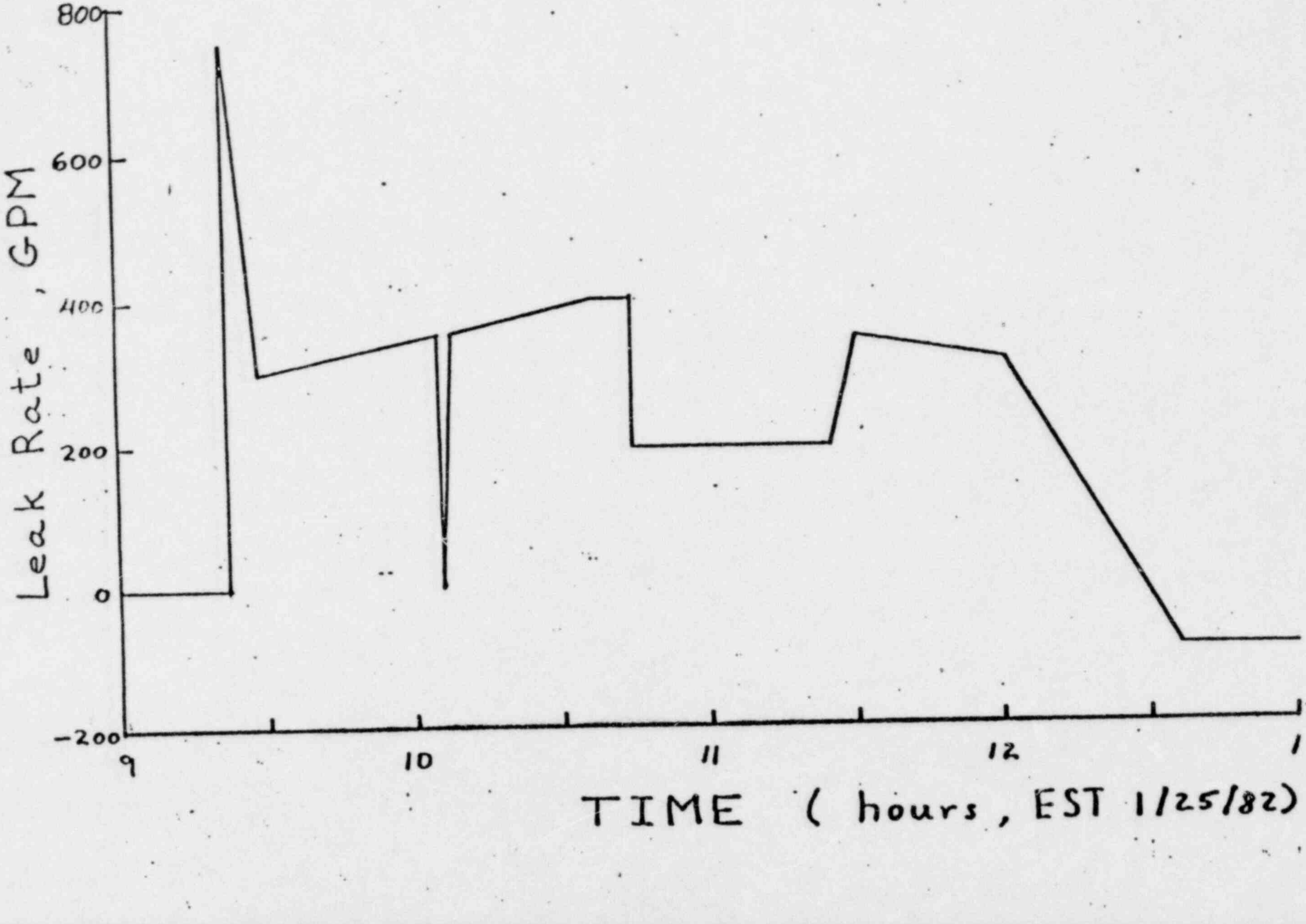
ANSWER

The attached graph (Figure 1) is our preliminary estimate of the leak rate as a function of time, calculated from information provided by the licensee.

FIGURE 1

G. Holahan 2/4/[illegible]

Preliminary Estimate of Leak Rate vs. Time



QUESTION 3. What triggered the steam tube rupture?

ANSWER

The licensee is continuing his inspection of the steam generator to determine the cause of the failure. This inspection will include removing sections of several tubes, including the ruptured tube, for laboratory examination.

QUESTION 4. Had there been indications of leaking steam generator tubes prior to the rupture on January 25?

ANSWER

A table of the history of steam generator tube inspection and plugging through May 1981, which includes leakage experience, is attached as Table 1. Although preliminary information from the licensee stated that the failed tube was not leaking immediately prior to the tube rupture, whether there was any indication of such leakage will be addressed in our 45-day report.

TABLE 1.

STEAM GENERATOR TUBE INSPECTION AND PLUGGING HISTORY

Date	No. Tubes Inspected		Primary to Secondary Leakage, gpm	Total Defects		Type of Degrad.	No. Defects Requiring Repair		No. Tubes Plugged/Sleeved/Pulled	
	A/B Hot	A/B Cold		A	B		A	B	A	B
In Factory			----	1	--	---	1	--	1/--/--	--/--/--
04/72	1050/----	----/----	----	0	0	---	0	0	0/--/--	0/--/--
03/74	3259/1098	516/ 516	----	19	0	a	19	0	19/--/ 2	0/--/--
11/74	1707/ 672	430/ 39	----	2	0	a	2	0	2/--/--	0/--/--
03/75	2174/1931	442/ 442	0.0050 A S/G	46	11	b/a	46	11	46/--/ 2	11/--/--
01/76	----/ 53	----/----	0.091 B S/G	0	2	a	0	2	0/--/--	2/--/--
02/76	3192/3247	3192/3247	----	39	2	a	39	2	39/--/--	2/--/--
04/76	100/1025	----/ 75	0.099 B S/G	0	15	b	0	15	0/--/--	15/--/--
04/77	2003/1525	268/ 268	----	13	1	a	13	1	13/--/--	1/--/--
07/77	----/ 300	----/----	0.012 B S/G	--	5	b	--	5	--/--/--	5/--/--
01/78	----/ 350	----/----	0.060 B S/G	--	8	b/a	--	8	--/--/--	8/--/--
04/78	2049/1714	325/ 375	----	1	15	b	1	15	1/--/--	14/--/ 1
02/79	2049/1714	325/ 375	----	--	6	b/a/c	--	6	--/--/--	6/--/--
12/79	----/1200	----/----	0.007 B S/G	--	13	c/a	--	13	--/--/--	13/--/--
04/80	3139/3182	325/ 375	----	1	31	d/c	1	13	1/--/--	28/--/ 3
11/80	3138/3151	325/ 375	----	--	3	c	--	2	--/--/--	0/ 5/--
05/81	3138/3141	325/ 400	----	--	15	c/a	--	6	--/--/--	1/16/ 3
				122 or 3.7%	127 or 3.9%		122 or 3.7%	99 or 3%	122/--/ 4 or 3.7%	106/21/7 or 3.5%

Type of Degrad.

a - Wastage
b - Cracking
c - ID Cracking
d - IGA
e - Pitting
f - Fatigue Cracking (B&W)
g - Erosion/Corrosion (B&W)

QUESTION 5. What was the cause of the PORV's apparent failure to close? Does the apparent failure of the PORV to close cause doubt about the adequacy of the industry's program to test such valves?

ANSWER

Ginna's power operated relief valve (PORV) uses pressurized "control" air to remotely operate the valve. Control air is routed through solenoid pilot valves which in turn pressurize one side of a flexible diaphram in the PORV's valve operator and simultaneously vent the other. The differential pressure across the diaphram causes it to flex and in turn moves the valve's stem and disc. If shut, the valve is held shut by an internal spring and air pressure. If open, the valve is held open by air pressure alone. Based on information provided at an NRC meeting with the licensee on February 10, 1982, the failure of the PORV to close resulted from a failure of a solenoid valve in the control system of the PORV. The failure was related to a modification of the solenoid valve that was made specifically for the Ginna PORV control system. The function of the failed solenoid valve is to open and relieve air pressure, thus permitting the PORV to close when signaled to do so. At Ginna, the failed solenoid valve is physically located within the pressurizer enclosure some distance from the PORV and is not considered to be a part of the PORV itself.

The PWR utilities, in response to one of the Commission-approved NRC Action Plan Items (i.e., Item II.D.1, NUREG-0660), funded the Electric Power Research Institute (EPRI) to conduct qualification testing of full-size prototypical PORVs and safety valves. The testing of the PORV's in the EPRI program was completed as of the end of August 1981. The PORVs were tested for open and closure capability for a variety of fluid conditions, proposed by the utilities and EPRI as generically representative of the types of fluids PORVs could be exposed to under transient or accident conditions.

The NRC staff reviewed and commented on the EPRI program as it was being formulated. During this review, and during the development of the Action Plan requirement, the problems of including PORV control systems in the program were specifically discussed. In addition to the enormous complexity involved in including as many control systems in the test program as there are PWR plants, it was also recognized that the PORV control system elements are not directly furnished by the valve manufacturer with the valve. For these reasons the PORV control system was not included in the generic PORV test program. However, the lessons learned from the malfunction of the air-operated control system for the PORV will be factored into a current evaluation which is assessing the need for improving air systems serving components and systems important to safety. In addition, the potential for accidents or transients being made more severe as a result of control system failures or malfunctions is being addressed in Unresolved Safety Issue A-47, "Safety Implications of Control Systems."

QUESTION 6. What would the course of the incident have been had the PORV block valve failed to close partially or fully following failure of the PORV to close fully?

ANSWER

Had the block valve failed to close after the PORV stuck open, the additional coolant loss from the primary system would have caused the primary system pressure to continue to decrease below approximately 900 psi. As the pressure in the reactor system decreased, the combined leak flow (through the valve and the rupture) would decrease and safety injection flow would increase until the flows were approximately equal. Analyses by Westinghouse in WCAP-9600 [1] indicate that the reator system pressure would stabilize at approximately 700 psi. The pressure would then remain relatively constant until the operator took action to depressurize the plant with the intact steam generator. If the block valve were only partially closed, the combined leak flow and safety injection flow would equalize at a pressure between 700 psi and the 1300 psi which was reached at Ginna after the block valve was fully closed. Additional leakage out of the reactor system through the broken tube in the isolated steam generator would not occur for the case of the block valve stuck fully open since the primary system pressure would be less than the affected steam generator pressure. If the block valve were only partially closed, the reactor system might be pressurized so that the leakage would be less than that which occurred with the block valve fully closed.

The effect on core coolant inventory of a combined PORV leak and steam generator tube leak would be similar to a postulated break in the reactor coolant hot leg with an equivalent break size of about 2 1/2 square inches. The consequences of this event on core cooling would be bounded by the spectrum of small break analyses performed for Ginna. [2] These analyses demonstrate that the core is adequately protected by the emergency core cooling system in the event of a small break LOCA.

The staff preliminarily concludes, based on the discussions above, that the effect of the block valve failing to close or leaking during the event at Ginna would have been a decrease in coolant loss through the steam generator tube and an increase in coolant loss through the PORV. Since coolant loss through the PORV is confined within the containment building and coolant loss through the broken tube may be released through the secondary system safety valves, off-site doses would probably have been lessened had the block valve stuck open at Ginna. Small break LOCA analyses for Ginna indicate that the core would be adequately cooled had the block valve failed to close. However, the dual break situation would have been more complex for the operators to diagnose and would have introduced the added difficulty of more water and radioactivity being released inside containment.

1/ Report on Small Break Accidents for Westinghouse NSS System, WCAP-9600, Westinghouse Electric Corporation, June 1979.

2/ Letter from LeBoeuf, Lamb, Leiby & MacRae, Attorneys for Rochester Gas and Electric Corporation, to L. Muntzing, U. S. AEC, transmitting small break LOCA analyses for Ginna, September 6, 1974

QUESTION 7. Did the procedure for responding to a steam generator tube rupture contain instructions for actions to be taken in response to development of a steam bubble in the reactor pressure vessel?

ANSWER

Based on preliminary information from the licensee, we understand that the Westinghouse Guidelines were used (Revision 1, April 1980) for developing plant-specific procedures and did not contain specific instruction for responding to a steam bubble in the reactor pressure vessel head area; therefore, they were not included in the Ginna procedures. However, based on special training and their knowledge of the TMI event, the operators were able to recognize the existence of the steam bubble through observation of the rapid increase in pressurizer level and reactor vessel head temperatures in conjunction with reactor coolant system pressure which indicated saturated steam conditions existed in the head area. Furthermore, readings from the core exit and vessel upper head thermocouples in conjunction with the primary system pressure confirmed that the steam bubble was confined to the head area.

A full review of the Ginna procedures is being conducted, and the results will be included in the 45-day report.

QUESTION 8. Was there a need during the incident to take actions not specified in the plant's written operati-. and emergency procedures? Were the emergency procedures in pla-e at Ginna consistent with Westinghouse guidelines as discussed in the January 28, memorandum from Mr. Speis to Dr. Mattson?

ANSWER

Plant operator response to the event, including the use of procedures, is being reviewed and the results will be included in the 45-day report. The emergency procedures in place at Ginna were based on the Westinghouse Guidelines Revision I dated April 1980.

The discussion of the event in the subject January 28, 1982 Speis memorandum concerned proposed Westinghouse guidelines dated September 1981 which are currently under review by the NRC staff. Further discussion is provided in enclosure 8-1 (SECY 82-58) dated February 10, 1982).

QUESTION 9. Had a water level measuring device been available, would it have assisted the operators in determining the size of the steam bubble in the pressure vessel and otherwise in bringing the plant to a stable condition?

ANSWER

There are several types of water level indication systems being considered by industry and the NRC staff with respect to assisting the operator in making determinations of inadequate core cooling. Some of these systems include level indication in the reactor vessel head region. Had such a measuring device been installed, it likely would have been an aid to the operator. The operators, however, did use the available instrumentation (pressurizer level, reactor coolant system pressure, and vessel upper head thermocouples) in making determinations of the existence of the steam bubble in the reactor vessel head. Furthermore, the core exit thermocouple readings in conjunction with the reactor coolant pressure confirmed that the steam bubble was confined to the reactor vessel head area and took actions accordingly.

QUESTION 10. What consideration has been given the potential for radioactivity escaping PWRs via a path including breaks in steam generator tubes and a stuck open safety valve?

ANSWER

Steam generator tube rupture accidents are one of the class of design basis accidents considered by applicants and staff in each review of PWR license applications. The staff's Standard Review Plan, NUREG-0800, describes the criteria and procedures used at Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)".

The analysis focuses on the potential release of radioactive noble gases and radioiodine both pre-existing in the reactor primary and secondary coolant, and generated concurrently with the accident. The former case uses the maximum activity levels permitted by the plant's proposed Technical Specifications. The latter case postulates activity released from the fuel as a result of the accident, including the potential for fuel failures.

The steam generator tube failure is assumed to be a double ended rupture of a single tube for purposes of calculating the rate of transfer of primary coolant to the secondary side of the affected steam generator. Flashing of the primary coolant is assumed to occur in this process with subsequent atomization and transfer of activity to the steam phase. Radioactivity entering the steam generator from the primary system is assumed to leave the steam generator, become airborne immediately, and be transported directly to the atmosphere via leakage paths not mechanistically specified. Such leakage could be through a stuck open safety valve, an open atmospheric dump valve, or through the condenser vent system. For FSAR safety analyses, such releases are assumed to occur during the first 30 minutes of the event, after which credit for operator action is allowed to terminate releases.

Exclusion area boundary and low population zone boundary doses are calculated and compared with the thyroid and whole body dose guideline values cited in 10 CFR Part 100. Conservative values of site specific atmospheric dispersion characteristics are used in these calculations.

The system response to the event postulated in this question is not covered by the Standard Review Plan. However, it is being considered in the preparation of new emergency procedure guidelines per TMI Action Plan item I.C.1.

QUESTION 11. Is it generally agreed that if a leak had developed in both steam generators, the operators would have been able to institute the "feed and bleed" process described in Mr. Speis' January 28 memorandum?

ANSWER

Had a leak developed in the second ("A") steam generator at Ginna, the need to institute the "feed and bleed" process to assure continued core cooling would have depended upon the leak size and total leak rate of primary coolant out of the primary system. It is uncertain whether the operators would have been able to institute "feed and bleed" for reasons described below.

The primary concern associated with two leaking generators is that in order to use the steam generators to cool down the primary system to the residual heat removal (RHR) system entry level, the primary system pressure would have to remain slightly higher than the pressure in both faulted generator secondaries during cooldown. This would result in continued leakage of primary coolant to the secondary system. Primary coolant would have to be replaced by the high pressure injection (HPI) system which pumps water from the refueling water storage tank (RWST) into the primary system. Thus, there is an amount of leakage that eventually affects the ability to cool the plant to RHR entry conditions prior to depleting the RWST. This behavior is different than other small loss-of-coolant accidents in the primary system. In those accidents leaking water will accumulate in the containment sumps. Once the RWST level drops to a preset value, the pump suction is switched from the RWST to the sump and sump water is recirculated through the core. Decay heat is ultimately removed by the containment heat removal system.

For larger tube leaks in both steam generators, which might deplete the RWST inventory prior to RHR entry conditions being reached, the operators would be expected to open all PORVs, essentually the same effect as causing a small break LOCA inside containment. This would rapidly depressurize the primary system (as well as remove decay heat) to below the faulted steam generator secondary pressures. In parallel with this action the operators would isolate both steam generators. Primary coolant makeup would be accomplished with the HPI pumps.

At Ginna, a two-loop 1300 MWth Plant, there are two PORVs manufactured by Copes-Vulcan with a relief capacity of 179,000 lb/hr. steam. Although neither the staff nor the licensee has performed any detailed calculations, scoping estimates indicate that the Ginna plant can remove decay heat by the "feed and bleed" process. It should be pointed out that in order to establish "feed and bleed," the operator must first establish PORV operability. In the case of Ginna, this involves reestablishing the air supply to the PORV which was initially isolated on low pressure safety injection actuation.

ANSWER 11 (CONTINUED)

At Ginna, there are procedures in place which instruct the operator on how to reset the safety injection signal in order to enable reestablishing the air supply necessary for PORV operability. This procedure was, in fact, used in reestablishing instrument air which allowed the initial operation of the PORV at Ginna during the tube rupture event.

Additionally, there is a backup nitrogen system which is manually controlled from the control room which can be used to actuate the PORVs in the absence of normal instrument air.

It is noted that failures in both steam generators are not required in the design basis for PWRs. Furthermore, existing emergency procedures, such as those at Ginna at the time of the tube rupture accident, do not provide the operators with explicit guidance on how to cooldown the plant with ruptures in multiple steam generators. However, as a result of the TMI accident, the staff's TMI Action Plan item I.C.1 requires the industry to upgrade emergency operating guidelines and procedures to cover multiple failure events. One of the specific events cited in NUREG-0737 is tube failures in multiple steam generators. Significant resources to the upgrading of guidelines and procedures have been allocated by both the industry and the staff. We anticipate approving the new emergency procedure guidelines by the end of FY 82. If this goal is met, upgraded procedures should be implemented at all operating plants by FY 83.

QUESTION 12. How many steam generator tube ruptures per year of the Ginna magnitude or greater do you expect?

ANSWER

There have been four steam generator tube failures of this type (greater than 50 gpm) at pressurized water reactors in the U. S. to date. The facility, date of the event and estimated leakage rate is as follows:

Plant	Date	Gallons/Minute (Maximum)
Point Beach Unit 1	02/26/75	125
Surry Unit 2	07/15/76	80
Prairie Island Unit 1	10/02/79	390
Ginna	01/28/82	700

The above data indicates that for all 48 PWRs licensed to operate in the U. S. (as of February 1), about one tube failure has been occurring every two years since 1975. The leakage rate from the Ginna failure is approximately the maximum possible for a single tube failure; therefore, leakage much in excess of this amount is not expected.

The technical resolution of Unresolved Safety Issue A-3,4,5, "Steam Generator Tube Failure," is in its final stages of development and includes consideration of recommendations for improvements in inservice inspection, steam generator secondary water chemistry monitoring and turbine condenser inspection. These improvements when completed should lessen the overall problem of tube corrosion. However, these changes, when implemented, are not expected to eliminate totally the possibility of future tube failures.

QUESTION 13. What is the likelihood of several steam tube ruptures occurring at one time? What is the maximum number of simultaneous or near simultaneous steam generator tube ruptures that are considered design basis accidents following which the reactor can be brought to a safe shutdown condition by following plant operating and emergency procedures?

ANSWER

Experience to date indicates that multiple tube failures is a low probability event.

As was discussed in our response to question 10, the steam generator tube rupture that is postulated to establish the design basis for the plant is the equivalent of a double-ended rupture of a single tube. For design base purposes, this is considered to encompass a spectrum of smaller leaks in either single or multiple tubes.

It is our belief that plants can most likely accommodate a larger number of tube failures, 1/ without exceeding the capacity of the ECC systems and without leading to core damage. Consequential radiological releases would also be calculated to increase. However, the radiological source would still be due to the induced primary coolant activity and not from fission products released due to gross fuel failures resulting from the event.

In addition to the tube rupture used for establishing the plant design basis, emergency operator guidelines and procedures presently being upgraded as a result of the TMI-2 accident will address methods for managing ruptures in multiple tubes and multiple generators. (See response to Question 11 last paragraph).

1/ We interpret the second part of the question to mean tube ruptures alone, not to be concurrent with or as a consequence of design base accidents (either primary system loss of coolant accident or main steam line break). The tolerable number of tube ruptures concurrent with or as a consequence of design basis accidents is rather small, dependent on the plant thermal hydraulic design and the design basis accident in question. However, we expect the tolerable number of tube ruptures would most probably be much larger for more likely accidents or transients.

QUESTION 14. Did WASH-1400 or more recent risk assessments determine the probability of occurrence of events in which one or more steam generator tube failure(s) are followed by various combinations of PORV, block valve and safety valve failures?

ANSWER

Steam generator tube rupture alone has been considered in PRA's as one of several types of small-break accidents to which pressurized water reactors may be subject. Multiple tube ruptures and ruptures in more than one steam generator have not been considered in PRA's nor have combinations of other component failures such as those identified in the question been considered. We are now taking a more careful look at these scenarios.

QUESTION 15. How long did it take to reach cold shutdown? Is this a period longer than desirable? What was the reasons for the period being longer than normal? What kinds of malfunctions during the extended cooldown period might have led to a significant release of radioactivity to the environment?

ANSWER

The plant was in cold shutdown the day following the vent (6:53 p.m.). The time from reactor trip to cold shutdown was 33 hours 25 minutes.

The period from reactor trip to cold shutdown was no longer than desirable. In fact, there was no urgent need to reach cold shutdown conditions, especially after the steam generator tube leak had been terminated (equalizing primary pressure with the faulted steam generator) and the plant was in a stable shutdown condition. This stable safe shutdown was reached about two and half hours after the reactor trip.

In general, it is expected that cooldown with a ruptured tube in one steam generator would be significantly slower than a normal cooldown. This slower cooldown is because the reactor coolant system pressure is to be equalized to the pressure in the ruptured steam generator to minimize or terminate reactor coolant leak flow through the rupture. Since the direct release of steam from the ruptured steam generator is to be minimized (the steam would contain radioactive products from the primary system), depressurizing the faulted steam generator must be by other less direct means. In Ginna, the steam generator with the ruptured tube was drained to the reactor coolant system through the ruptured tube. Additional cooling and depressurization was provided by cold auxiliary feedwater which replaced part of the drained water. The length of time for the cooldown was primarily governed by the management's desire to go slowly and cautiously. The time to reach cold shutdown was consistent with the plant's condition and, therefore, no longer than desirable.

If there had been no steam release from the ruptured steam generator in the early stage of the event, it is reasonable to expect the cooldown period would have been longer. For a large initial steam space in the ruptured steam generator, a limiting factor for steam generator draining is need to keep the steam generator tubes covered. Should the steam come in direct contact with the tubes, rapid condensation would occur resulting in a rapid depressurization of the ruptured steam generator secondary side and re-initiation of reactor coolant leakage back through the ruptured tube.

During most of the extended cooldown period at Ginna, the ruptured steam generator was isolated and its pressure was significantly lower than the safety valve set pressure. All other steam valves from the steam generator were secured. The reactor coolant system was controlled similar to a normal cooldown, except for measures (increased letdown, boration) to accommodate the leak flow to the primary system coming from the secondary side.

As indicated in the response to Question 10, potential releases of radioactivity to the environs during the short term or long term most directly relate to additional malfunctions in the faulted steam generator. Such leakage could be through a stuck open safety or relief valve flow path or through the condenser vent system. For Final Safety Analysis Report radiological safety analyses, such releases are assumed during the first 30 minutes of the event, after which credit for operator correction is allowed.

QUESTION 16. Did any part of the reactor pressure vessel cool at a rate in excess of that stipulated in the plant technical specifications?

ANSWER

Analysis of information to determine specific cooldown rates is being conducted and will provided in the 45-day interim report.

QUESTION 17. Was there a capability at Ginna to remotely vent the reactor pressure vessel high points? D.. s the Commission believe that conditions might develop in PWRs calling for the use of remotely controlled valves for the purpose of venting steam?

ANSWER

The physical capability existed at Ginna but was not declared operational since staff review is not complete. The reactor vessel head vent system, including associated hardware and control system, has been installed at Ginna. Before the staff authorizes use of the installed vents, we will review not only the design but also the associated procedures which are to specify when to vent and when not to vent. Procedural guidelines for venting is an integral part of our review of transients and accidents. We expect to complete procedural reviews in FY-1982, and finish designs for all PWRs in FY-1983.

In PWRs with inverted U-tube steam generators (i.e., Westinghouse and Combustion Engineering reactors), high point vents are required to be located on the vessel head. This requirement was added for the purpose of providing a vent path for non-condensible gases that could accumulate in the primary system under degraded core cooling conditions. Although these vents could be used to vent steam which might accumulate in the vessel upper head after saturation conditions are reached in parts of the vessel, it is not expected they would be used for this purpose, nor is it recommended that they be used to vent steam. Steam in the upper head of Westinghouse and Combustion Engineering reactors does not pose a direct threat to continued core cooling. If the steam bubble were to expand to the hot leg outlets, it would most likely condense as it came into contact with subcooled water exiting the core. If, for any reason, the water exiting the core was saturated, the steam would enter the hot leg pipes and travel to the steam generators, where it would be condensed.

For events such as the one at Ginna, the method preferred for removing steam which accumulates in the upper head of the vessel is to restart a reactor coolant pump. The pump will force subcooled water into the upper head region and condense the steam bubble. The operators at Ginna demonstrated the capability to do this following the formation of a steam bubble in the upper head.

In PWRs with once-through steam generators (OTSGs) (i.e., B&W reactors), a steam bubble in the upper head of the vessel has the potential to temporarily interrupt natural circulation if it expands and is able to enter the hot leg outlets without condensing. Pursuant to item II.B.1 of the TMI Action Plan these plants will eventually have high point vents installed on the top of the hot leg inverted U-bends. In addition, some utilities with B&W reactors will install vents on the top of the vessel head.

Analyses by B&W have indicated that interruption of natural circulation is a temporary phenomenon. The analyses show that system repressurization following the interruption of natural circulation will ultimately produce thermal-hydraulic conditions in the primary system which restore natural circulation. The staff is still reviewing the capability of the B&W analysis methods to properly predict the relevant thermal-hydraulic phenomena.

ANSWER 17 (CONTINUED)

B&W has recently recommended use of the hot leg high point vents to vent steam which may accumulate during the recovery phase of a small break loss-of-coolant accident (SBLOCA). During the accident phase of a SBLOCA, B&W has recommended the "bumping" of the reactor coolant pumps to sweep any steam trapped in the hot leg high points into the steam generator.

The use of the high point vents to vent steam in B&W reactors, as well as the acceptability of the B&W calculational models to properly predict the thermal-hydraulic behavior of the primary sytem under two-phase conditions, is under active staff review. At this point in the review, it is our preliminary conclusion that the use of the vents in B&W reactors to remove steam which accumulates at primary system high points may be the preferred method of steam removal if a reactor coolant pump cannot be restarted and run continuously.

QUESTION 18. At any point during the Ginna event, did the steam generator containing the ruptured tube control the primary system pressure? Are operators at Ginna and other PWRs trained with respect to actions to be taken when a steam generator controls primary system pressure?

ANSWER

In order to prevent further contamination and to aid in cooling down the faulted steam generator, a feed and bleed operation was used. This operation consisted of providing feedwater to the faulted steam generator in order to maintain level within a desired band; a steam bubble in the steam generator was maintained during this period. As a result of primary system pressure control by the operators through the use of the normal charging/letdown systems and by controlling the cooldown rate through the "A" steam generator, primary system pressure was decreased in a controlled manner. Pressurizer level was maintained and primary system pressure was controlled by the pressurizer. However, during this period the plant was controlled in such a manner as to result in an inflow of water from the "B" steam generator to the primary system through the ruptured tube.

This area, specifically during the early part of the transient, is being reviewed further and the results will be included in the 45-day report.

The operators at PWRs are trained to maintain control over both primary and secondary system pressure following a steam generator tube rupture. The goal is to minimize flow between the two systems by maintaining the two systems within 50 psi of one another.