Response to Long Term Commitments

Ginna Restart SER

Steam Generator Tube Rupture Incident

November 22, 1982

Item 1: Modify the 1C safety injection pump logic to establish a fixed loading sequence and to provide a lockout feature to prevent automatic transfer of a fault at the load to the redundant bus.

Response: A modification will be made to the 1C safety injection pump (SIP) logic. The modification will consist of replacing the instantaneous contacts used as interlocks with time delayed contacts. This will insure predictable loading of the 1C pump motor while not degrading overall reliability.

A control scheme will be developed that will detect the failure of the 1C SIP breaker to close onto bus 14, other than due to bus undervoltage, and allow it to swing over to the alternate bus, bus 16, after a total accumulated time delay of 37 seconds. On bus 14 undervoltage, the 1C SIP will be transferred to bus 16 within 2 seconds as is done at present. The control scheme will not allow the transfer to take place if an electrical fault occurs on the 1C motor. Thus, this transfer scheme will not subject both trains to a commom electrical fault. The modification satisfies all requirements of the plant safety analysis.

This modification will be implemented prior to startup from the 1983 refueling outage.

Item 2: Within six months, reanalyze the radiological consequences of a steam generator tube rupture. The analysis should include the effect of overfilling the steam generator or evidence should be provided that overfilling will not occur.

Response: A steam generator tube rupture analysis was performed using the assumptions required for a FSAR type analysis. The thermal hydraulic transient was run for two cases. One case assumed the primary to secondary leakage was terminated in 30 minutes; the other case assumed the leakage was terminated in 60 minutes. Two radiological evaluations were performed for each of these two cases. The radiological evaluations assumed a primary coolant activity of 1 microcurie dose equivalent I-131 with either a pre-accident iodine spike or an accident initiated iodine spike.

The two FSAR-type analysis cases include several additional conservatisms beyond those normally applied. First, a pre-existing primary to secondary leak rate of 1 gpm is assumed. Plant Technical Specifications limit the leak rate to 0.1 gpm per generator. (This assumption has a very small effect on the dose calculation). Second, in the 60 minute leak termination case, overfill is predicted to occur with resultant water relief from the faulted generator based on filling of the faulted steam generator as opposed to filling both the generator and the steam line. This results in earlier water relief and increased releases than would actually need to be assumed.

A third case was also analyzed. The mass releases calculated by LOFTRAN for the Ginna incident (see Attachment B) plus several best estimate assumptions were employed. Several conservatisms have been employed in this analysis, which differentiate it from a "best estimate" case. The pre-existing equilibrium primary coolant activity is assumed to be at the Standard Technical Specification limit, as opposed to the much lower best estimate value. A pre-existing primary to secondary leak of 1 gpm is assumed as opposed to 0.1 gpm per generator. The total break flow from the primary system is greater than that which would be derived from an evaluation of water drawn from the boric acid storage tank and refueling water storage tank during the Ginna incident. This would over estimate water relief from the faulted steam generator.

A detailed description of the analysis is presented in Attachment A and the thyroid dose results are tabulated below. Note that although Attachment A uses the terminology "site boundary", the meteorological parameters actually represent the more restrictive exclusion area boundary parameters.

As illustrated, the 60 minute termination case bounds the LOFTRAN best estimate case which in turn bounds the results if the Reference 1 mass releases were used. All doses are within 10CFR100 guidelines.

	Thyroid Dose (Rem)			
CASE	PRE-ACCIDENT		ACCIDENT INITIATED	
	EAB*	LPZ**	EAB*	LPZ**
30 min. leak termination	22.3	1.4	2.9	0.2
60 min. leak termination	273	17.1	91.5	5.7
LOFTRAN best estimate	8.5	1.5	2.9	1.4

* EAB = Exclusion Area Boundary
** LPZ = Low Population Zone

We believe that the evidence supports the fact that leak termination would occur well within 60 minutes following a steam generator tube rupture. In the Ginna incident, the criteria for SI termination, and therefore leak termination, were met within 47 minutes (0925 to 1012 AM). Due to the uncertainty resulting from upper head voiding, however, SI was not terminated until 1037 AM with primary and secondary pressures essentially equalized shortly thereafter. A number of changes since the January 25 incident give us

confidence that leak termination would occur much sooner should a tube rupture reoccur. First, the specific cause of operation uncertainty, the upper head void, has been explicitly addressed in the Ginna procedure. SI termination is permitted whether or not an upper head void exists. Thus, the operators would not have hesitated to terminate SI. Extensive operator training has occurred regarding tube ruptures. The operators are aware that, when the reactor coolant pumps have been tripped, upper head voiding is to be expected. The broader understanding of plant response to a tube rupture has been aided by reviews of other contingency procedures, for example, tube rupture with failed open secondary side valve (see the response to item 12). A second factor, which will aid in many, but not all, steam generator tube rupture incidents, is the reduction in the pressure used for reactor coolant pump trip. Following installation of a environmentally qualified wide range pressurizer transmitter, the trip criteria was reduced from 1715 psig to 1270 psig. A third factor which will reduce the time to SI termination for a range of small tube ruptures, although not a factor in the January 25 incident, is applicable to those cases where the RCPs are tripped and primary system depressurization must occur via pressurizer PORV operation. During the January 25 incident, the operators had to reset the safety injection signal, then reset the containment isolation signal in order to supply instrument air to containment and to the pressurizer PORV. Plant design prohibited SI signal reset until the suction for the safety injection pumps had switched from the boric acid storage tanks to the refueling water storage tank. Thus, SI reset and subsequent PORV operation could be delayed. As described in the response to item 6, this restriction will be eliminated by a plant modification which will be accomplished during the Spring, 1983 refueling outage. SI termination would not have occurred earlier without this restriction in the January 25 incident, however, the modification will eliminate potential delay in a range of tube failures with a smaller leak rate. Further work in the area of reactor coolant pump trip is continuing (see item 8) which will permit RCP operation on a broader range of tube rupture sizes, including a rupture as occurred on January 25 and a full diameter tube failure. However, the RCPs would still not be available following a tube rupture concurrent with a loss of offsite power.

In conclusion, leak termination would occur within 60 minutes of a tube rupture. This is based on improved procedural guidance and improved operator training. Further confidence that this time will not be exceeded is based on the modifications which will facilitate earlier SI reset. Improved RCP trip criteria will address some but not all tube rupture incidents.

Based on the results of the radiological consequence evaluation plus the evidence supporting 60 minute leak termination, a proposed Technical Specification will be prepared and submitted for a primary coolant iodine concentration limit equal to that suggested in the NRC's Standard Technical Specifications of 1 μ Ci/gm.

Item 3: Revise the setpoints and surveillance requirements for the effluent monitoring system.

Response: The revised setpoints and surveillance requirements for the effluent monitoring system are presented below:

Main Steam Radiation Monitoring System

The basis of the high alarm setpoint has been added to Procedure P-9, Precautions, Limitations, and Setpoints, Radiation Monitoring System. The setpoint of 0.1 mR/hour is just above ambient radiation levels to give maximum sensitivity while preventing false alarms.

The status of the Main Steam Radiation Monitors is continuously determined by control terminals located in the control room and the Technical Support Center (TSC). Any change in status is recorded, the appropriate status light is lighted and an audible alarm sounds. The monitors are tested quarterly for response to a radioactive source in accordance with Periodic Test Procedure PT-17.3. Activation of recorders for radiation level, and safety valve and atomspheric steam dump position are verified on receipt of a high alarm signal caused by the portable source.

The monitors cover the range of 10^{-1} to 10^{3} μCi/cc as required by Reg. Guide 1.97, Rev. 2, December 1980. These setpoints are consistent with those identified in a telephone conversation with the staff on 5/6/82.

Monitor output can be programmed to print the last 24 hourly averages once a day and the last 24 ten minute averages every 4 hours. In the demand mode the last 24 daily averages, the last 24 hourly averages, the last 23 ten minute averages, and the current value can be printed.

Air Ejector Exhaust Monitoring System

The high range monitor (R15A) can be programmed to print the values described above. The same readouts are also available on demand as described above. Any change in status by one of the channels also causes the reading to be printed. If the status is a high alarm condition, readings are automatically printed every ten minutes while the condition exists.

The alert and high alarms for each channel are set to correspond to a percentage of the release rate limit and have been added to Procedure P-9. Setpoints are arranged so that an

alarm is received by a higher range channel before the lower range channel goes offscale. In addition, the lowest range channel of R-15A alarms before monitor R-15 goes offscale.

The status of all channels of the high range monitor R-15A are continuously determined by control terminals located in the control room and the TSC. Any change in status is recorded, an audible alarm sounds, and the appropriate status light is lighted. The low range monitor R-15 is tested for operability monthly using Periodic Test Procedure, PT-17.2. Monthly testing requirements for R-15A have been added to PT-17.2.

Procedures or calculative methods to be used for converting the monitor readouts to release rate are contained in Procedure S-14.4 and PC-23.5.

For the purpose of offsite dose assessment the value of having instantaneous release rates and instantaneous meterological parameters is not apparent since these must be averaged over time to obtain the estimated dose. Therefore, a strip chart recorder in addition to the 10 minute average readouts in the computer is not necessary to indicate release rate from the air ejector exhaust.

Item 4: Prior to December 1, 1982, develop procedures for snow sampling.

Response: Procedure SC-452 "Sampling Snow, Grass, Soil and Vegetation" was developed and implemented on July 16, 1982. Input was obtained from both the NRC and NY State Environmental Groups during the procedure development. The procedure has also been recently provided to these groups for information and any additional comment.

Item 5: Within six months, consider procedure changes to reduce or prevent ventilation intake of contaminated air during unplanned releases.

Response: A step has been added to E-1.4 "Steam Generator Tube Rupture" and O-6.10 "Operation With a Steam Generator Tube Leak Indication," outlining actions which should be considered to reduce the intake of radiological contaminants by building supply air units. If air intakes are in line with the path of discharge of contaminated air or if building air monitoring indicates this is occuring, the supply air handling units should be removed from service. The areas most likely to be affected would be the Auxiliary and Intermediate Buildings through the Auxiliary Building supply air handling unit, the Turbine Building through the wall supply fans and the Service Building through the service building air handling units.

Item 6: Within six months, review the requirement for a safety injection signal to be present for automatic transfer of safety injection pump suction from the boric acid storage tanks to the refueling water storage tank.

Response: The requirement for a safety injection signal to be present for the automatic transfer to take place has been reviewed. The results of the review indicate that it is acceptable to remove this dependency. A modification will be made to the automatic switchover logic that will cause the switchover to occur on boric acid storage tank level only. The presence of an SI signal will not be required for the automatic switchover to occur. The modification will be implemented prior to startup from the 1983 refueling outage.

Item 7: Within six months perform a detailed thermal-hydraulic analysis of system behavior during the incident to verify phenomena, including void formation.

Response: Westinghouse has analyzed the Ginna tube rupture incident using their LOFTRAN code. A description of the analysis and results are provided in Attachment B. Reasonable agreement between LOFTRAN results and plant data have been obtained. The extent of voiding in the primary system is also evaluated.

Item 8: Within six months, study the RCP trip criteria with the purpose of finding a method to keep the RCPs running during a steam generator tube rupture.

Response: Westinghouse has studied the RCP trip criteria using their LOFTRAN code. A description of their analysis and results are provided in Attachment C.

The analysis evaluated a number of potential RCP trip criteria with the objective of finding a criteria which would not require RCP trip during a tube rupture equivalent to a double ended tube failure (which bounds the Ginna incident). A criterion which relies on both primary and secondary system pressures was found which satisfies the above objective. Prior to implementation of the new criterion, additional Westinghouse Owner's Group (WOG) evaluation is required. The criterion must be verified under simulator conditions to assure that operators can respond in sufficient time should the accident be a small break LOCA versus a SGTR. The WOG will study the RCP trip issue. Guidelines for RCP trip will be included in Revision 1 of the Emergency Response Guidelines scheduled for release in April 1983 with backup documentation scheduled for release in June 1983. The Revision 1 guidelines for RCP trip should be implemented 3 months after receipt.

Item 9: Within six months, study the RCP restart criteria to ensure that proper criteria are employed.

Response: The RCP restart criteria has been studied and is presented in Attachment D. The RCP restart criteria presented is sufficient to ensure that an indicated pressurizer level and reactor coolant subcooling will be maintained during a steam generator tube rupture. This criterion has been included into Ginna procedures.

Item 10: Within six months, review plant procedures to provide any additional guidance required for operator actions to be taken in response to real or suspected reactor vessel upper head voiding.

Response: Additional guidance beyond that present in the Ginna procedures on January 25 regarding real or suspected reactor vessel upper head voiding has been found necessary in two areas, safety injection termination and reactor coolant pump restart. Additional guidance has been added to the S/G Tube Rupture and Loss of Secondary Coolant procedures to permit SI termination with a upper RV head void as long as natural circulation and other SI termination criteria are met.

Guidance has also been added to the "E" series procedures (major accident procedures) concerning upper RV head void collapse during RCP start. The procedures permit RCP start with an upper head void as long as adequate pressurizer level and RCS subcooling are present.

Item 11: Within six months, provide procedures for cooldown following a steam generator tube rupture.

Response: The procedures for cooldown following a steam generator tube rupture have been prepared based on Westinghouse Owners Group guidance and have been implemented.

Item 12: Within six months, provide procedures to cover a steam generator tube rupture with a failed open steam generator safety valve.

Response: The steam generator tube rupture procedure has been broadened to include various size steam breaks, including a break equivalent to a failed open safety valve, coincident with a steam generator tube rupture on the same steam generator. This procedure has been implemented.

Item 13: Within six months, review the time response of simulators used for operator training of steam generator tube ruptures and implement any actions necessary to identify differences between the simulator and Ginna.

Response: The operators of the Zion and SNUPPS simulators were contacted and asked, and have agreed, to emphasize during operator training the difference between the simulated tube rupture response and the Ginna response. The simulators were tested to determine how to run the tube rupture to

closely approximate the Ginna response. A close approximation of the Ginna response can be obtained by making adjustments to the standard simulator lineup. The difference between the simulated and Ginna response is being emphasized during operator training. These simulators are the only simulators being used for Ginna licensed personnel training. Should other simulators be used in the future, similar arrangements will be made.

Item 14: Confirm by test that the modified letdown system isolation functions properly and submit, within six months, a detailed design description.

Response: This modification was performed in order to provide automatic closure of LCV 427 (letdown line stop valve) and valves V200A, V200B and V202 (letdown orifice valves) upon initiation of containment isolation.

During normal plant operation, reactor coolant letdown flow is drawn from Loop "B" reactor coolant pump suction line through LCV-427 (Figure 14-1). LCV-427 is located in containment inside the missile barrier and fails open upon loss of control power or instrument air. The letdown flow then passes through the shell side of the regenerative heat exchanger where its temperature is reduced by the charging stream flowing through the tube side. The three letdown orifices downstream of the regenerative heat exchangers limit the flow of the letdown stream during normal operation and reduce the pressure to a value compatible with the downstream piping design. Letdown flow is controlled by remote manual operation of valves AOV-200A, AOV-200B, and AOV-202. These valves will automatically close when LCV-427 closes to prevent the formation of a steam bubble in the shell side of the regenerative heat exchanger. When one or more orifice valves are open, the letdown flow passes out of containment through containment isolation valve AOV-371 to the non-regenerative heat exchanger in the CVCS system.

A 600 psig relief valve (V203) located downstream of the letdown orifice valves provides overpressure protection of the downstream piping. The relief valve discharges to the pressurizer relief tank (PRT).

Before this modification, if a containment isolation signal were generated during normal letdown operations, AOV-371 would automatically close while LCV-427 and one or more orifice valves would remain open, allowing pressure to build to the relief valve setpoint. To prevent this from occurring, a containment isolation signal was input to LCV-427. When LCV-427 receives a close signal, the orifice valves will also close, providing redundant isolation of the letdown stream from the relief valve.

To reopen these valves the containment isolation signal must be reset at the system level followed by reset of the individual reset buttons on the CI Matrix panel.

This modification required the installation of two cables running from the containment isolation relay racks in the Relay Room to the main control board, where the appropriate terminations were made to the LCV-427 control circuit (Figure 14-2). Since LCV-427 is within the missile barrier it is subject to pipe whip from a high energy line break (HELB).

A HELB near LCV-427 can be postulated to short out the air solenoid on the valve, resulting in the loss of LCV-427 control and the auxiliary relay (20X-427) which isolates the orifice valves. To prevent the loss of 20X-427 due to HELB, fuses were installed in the main control board to isolate the LCV-427 solenoid if it becomes shorted. This ensures that the orifice valves will close.

This modification was installed by procedure SM-3257.1, Rev. 0, "Letdown Isolation Modification" which verified proper operation of the modification.

Item 15: Confirm by test that the wide range pressurizer pressure transmitter functions properly and submit, within six months, a detailed design description.

Response: A Foxboro NE-11 series pressure transmitter (PT-420A) qualified to IEEE 323-1971 and IEEE 344-1971 was installed off the pressurizer during the spring 1981 outage under EWR 2604A to provide a wide range RCS pressure input (0-3000 psig) to the subcooling margin monitor. Transmitters of this series are currently undergoing qualification testing to IEEE 323-1974 and IEEE 344-1975. PT-420A shares a pressurizer tap with PT-430 and LT-427 and is powered from a class IE source.

In February 1982, an output from PT-420A was input to a spare pen on the pressurizer pressure recorder to provide uninterrupted RCS wide range pressure indication in the event of loss of offsite power. This modification required the installation of an instrumentation cable from the relay room to the main control board and utilized a spare signal output card in the relay room.

During the spring 1982 outage, the pressurizer level and pressure transmitters (including PT-420A) were moved from the containment basement to the intermediate level under EWR 3259 to preclude loss of these transmitters due to flooding. The above described modifications were unaffected by this relocation.

Proper functioning of the pressure transmitter was confirmed by procedures CP-420A, Rev. 0, "Calibration and/or Maintenance of Reactor Coolant System Pressure Transmitter PT-420A," CP-410A.3, Rev. 2, "Calibration and/or Maintenance of PT-420A Current To Voltage Converter" and CP-410A.7 Rev. 1, "Calibration and/or Maintenance of TY-410 A2, Voltage to Current Converter."

Item 16: Perform fiber optic inspections during the intermediate outage of the tubes in Row 45 of the B-steam generator hot leg.

Response: A fiber optics inspection of plugged tubes in Row 45 of the B-Steam Generator hot leg was performed during an intermediate outage which began on September 25, 1982. The purpose of this inspection was to examine the tubes in this area whose ends were previously plugged and whose location was in the vicinity of where two severed tubes had been found wedged between the tube bundle wrapper and these candidate tubes during the Spring 1982 outage.

The head of the fiberscope was attached to a peroscopic device that was inserted through the Number 4 wedge access port. This arrangement provided the capability to scan and perform an external inspection of these tubes from a height just above the top of the tube sheet to just below the first support plate. The output of the fiberscope was coupled to a video camera and recording system. No defects or abnormalities were visible on these tubes.

This examination fulfills the commitment.

Item 17: Submit, within six months, a detailed design description of the Loose Parts Monitoring System.

Response: A description of the Loose Parts Monitoring System is presented in Attachment E.

Item 18: Within six months, review and identify potential transients and accident scenarios that could provide relatively stagnant flow conditions in a coolant loop, and examine the effect of the operator taking actions which would draw the cold water into the vessel. For these scenrios, review and modify procedures and train operators as necessary to prevent or minimize the flow of cold water into the vessel.

Response: A discussion of loop stagnation is presented in Attachment F. The attachment presents a review of potential stagnation transients and scenarios. Since the major concern of loop stagnation is pressurized thermal shock (PTS), the two issues are closely related. Operators have been trained on PTS and flow stagnation (Item 19). The results of procedure review for PTS will be incorporated in revision 1 to the Emergency Response Guidelines which are scheduled for release in April 1983. The stagnation issue will be further studied

by the Westinghouse Owner's Group (WOG). RGE will implement the results of this study.

Item 19: In the long term, include an evaluation of the scenarios identified above in the overall resolution of the pressurized thermal shock issue. Modify equipment, procedures and operator training in accordance with the resolution of that issue.

Response: RGE is a member of the Westinghouse Owner's Group (WOG) which is actively involved in the PTS issue. RGE will implement the procedures recommended by the WOG. As the issue is resolved, all required actions will be performed.

Operator training on PTS issues was conducted by Westinghouse on October 11, 12, and 13, 1982. This training included the accidents in which thermal shock can occur, fracture mechanics, flow stagnation, and remedies for PTS.

Item 20: Within six months, determine the criteria which should be provided in the steam generator tube rupture procedures for deciding when to discontinue the use of the main condenser in favor of the atmospheric steam dump.

Response: It has been determined that steam dump to condenser should be utilized whenever possible during a steam generator tube rupture. The determination was based on minimizing releases and the best method to monitor releases. When steam is dumped to the condenser many contaminants remain in the condensate system and less contaminants are released through the air ejector than would be released through steam dump to atmosphere. It is also more straight forward to monitor releases through the air ejector than through the atmospheric steam dump. Therefore, the current tube rupture procedure E-1.4 directs operators to use steam dump to condenser as long as necessary permissives are met.

References

1. Incident Evaluation, Ginna Steam Generator Tube Failure Incident, dated April 12, 1982.