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SUBJECT: Forwards followup actions resulting from NRC review of TMI-2 accident,in response to 790913 request.

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LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
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October 17, 1979

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

This letter is in response to a letter dated September 13, 1979 from Darrell G. Eisenhut which was received on September 17, 1979. The letter requested that we review the requirements resulting from NRC Lessons Learned Task Force report and the requirements resulting from the NRC Staff's Emergency Preparedness Studies. The attachment to this letter provides our response to these requirements.

Very truly yours,

L D White, Jr.
L. D. White, Jr.

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Section 2.1.1 - EMERGENCY POWER SUPPLY REQUIREMENTS FOR THE PRESSURIZER HEATERS, POWER-OPERATED RELIEF VALVES AND BLOCK VALVES, AND PRESSURIZER LEVEL INDICATORS IN PWR'S

A. TASK FORCE POSITION ON PRESSURIZER HEATER POWER SUPPLY

POSITION 1.

The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

All of the pressurizer heaters and controls are presently fed from safeguard buses which are provided with backup power from the diesel generators.

The proportional pressurizer heater bank has a capacity of 400 kw and the heaters and their controls can be loaded onto train A, emergency bus 14. The backup heater bank also has a capacity of 400 kw and the heaters and their controls can be loaded onto train B, emergency bus 16.

Westinghouse has performed analyses for the Owners Group to determine the pressurizer heater requirement to maintain natural circulation with subcooled conditions in the reactor coolant system hot legs. The analyses show that, for a plant with a 1000 $ft^3$ pressurizer, 100 kw of heaters are required within 5 to 6 hours after loss of forced cooling. Westinghouse has recommended that the heaters be supplied within 1 hour. This analysis conservatively bounds the Ginna conditions since the Ginna pressurizer volume is 800 $ft^3$. Thus, we currently have the capability to supply sufficient pressurizer heaters from the emergency buses in a manner that will provide redundant power supply capability.

POSITION 2.

Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

The only time the pressurizer heaters will not have a power supply available is upon actuation of a safety injection signal or on loss of offsite power.

Procedures and necessary training will be provided to instruct the operators when and how to restart the heaters. Direction will also be given on how to trip out unnecessary heater capability in the event that the alternate power supply is being used. This will be completed by January 1, 1980.

POSITION 3.

The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions. (Category A: Implementation shall be completed prior to OL, or January 1, 1980.

RG&E Response

The proportional heaters can be loaded onto the A diesel generator from the control room at any level between 0 and 400 kw. Thus, 100 kw can be supplied from the diesel generator within 1 hour. The backup heater capacity of 400 kw can be loaded onto the B diesel generator from the control room. Alternatively, the backup heaters can be adjusted for any power level between 0 and 400 kw at the distribution panel 1B1 and 1B2 (train B) in the auxiliary building. Thus, sufficient backup heater capacity can be supplied from the diesel generator within 1 hour. The present plant design satisfies the NRC concern.

POSITION 4.

Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

RG&E Response

Pressurizer heater motive and control power interfaces with the emergency buses are accomplished through devices that have been qualified in accordance with safety-grade requirements as established in the Ginna FSAR.

B. TASK FORCE POSITION ON POWER SUPPLY FOR PRESSURIZER RELIEF AND BLOCK VALVES AND PRESSURIZER LEVEL INDICATORS
POSITION 1.

Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

The pressurizer power operated relief valves, valves 430 and 431C are normally air operated valves which fail closed with the loss of air. The air compressors, the source of supply air for these valves, are not supplied from emergency power even though the solenoid valves associated with the PORVs are. The solenoids for both valves supplied from train B emergency power supply. With a loss of offsite power or with a containment isolation signal, the valves close.

A low temperature overpressure protection system has been installed at Ginna which provides an alternate system for the operation of valves 430 and 431C. This system will enable the valves to operate independently of the air system. Nitrogen accumulators provide the motive source to the valves while the electrical power required for the solenoids for each valve is supplied from separate emergency supplies. The power for the solenoids controlling valve 430 is from train A and the power for the solenoids controlling valve 431C is from train B. A failure of one emergency electrical supply will not affect the valve operation of the valve associated with the redundant emergency supply. The low temperature over-pressure protection system can be controlled from the control room.

POSITION 2.

Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

Ginna has two pressurizer power-operated relief valves, valves 430 and 431C. The block valves, which are located upstream of the relief valves, are valves 516 and 515, respectively.

The motive power for the block valves is from the emergency buses. Valve 516 (in series with PORV 430) is supplied from train B and valve 515 (in series with PORV 431C) is supplied from train A.

The present loading of the PORVs and the block valves on the emergency buses satisfies the NRC concerns as identified in the September 13, 1979 letter and as clarified in the NRC regional meetings.

POSITION 3.

Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

Motive and control power connections to the emergency buses meet the safety grade requirement established in the Ginna FSAR.

POSITION 4.

The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

The pressurizer level indicating instrument channels are powered from vital instrument buses. These buses have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Section 2.1.2 - PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES

TASK FORCE POSITION

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves. (Program and Schedule: Implementation shall be completed by January 1, 1980; complete tests: implementation shall be completed by July 1981.)

RG&E Response

RG&E is a member of an Owners group formed by utilities owning and operating Westinghouse reactors. The Westinghouse Owners Group is working in conjunction with other PWR Owners and the Electric Power Research Institute (EPRI) to develop a program for qualification of relief and safety valves under expected valve operating conditions including solid water and two-phase flow condition. Thus, we will follow the schedule developed and carried out by EPRI.

Section 2.1.3.a - DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRS AND BWRS

TASK FORCE POSITION

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe. (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

The reactor coolant boundary valves, including the relief and safety valves, are listed in Table I. As indicated, the position of all valves except the pressurizer safety valves, is monitored directly by limit switches indicating valve stem position, open or closed. The flow of steam (coolant) from the pressurizer safety valves is monitored directly by thermocouples in the valve discharge piping from each valve. All indication is available in the control room. The flow of steam (coolant) from the PORVs is also monitored directly by thermocouples in the valve discharge piping from each valve. This indication is available in the intermediate building. Temperature indication on the PORV discharge header is available in the control room.

Safety and relief valve indications will be monitored and alarmed by the plant process computer on or before the required implementation date, January 1, 1980. Qualification of the components in these circuits, including the limit switches, to perform their indicating function during events which might be associated with a misaligned or failed valve, is being reviewed and qualification commensurate with function will be provided on or before the required implementation date, January 1, 1980, dependent on the availability of the containment for installation.

TABLE I

Reactor Coolant System
Boundary Valve List

| Valve No. | Type | Function | Position Status Indication |
|---|---|---|---|
| V310 | AOV | Excess Letdown Isolation (A Loop) | Limit Switches O&C |
| LCV427 | AOV (F.O.) | Excess Letdown Isolation (B Loop) | Limit Switches O&C |
| PCV430 | AOV (F.C.) | Pressurizer Power Operated Relief | Limit Switches O&C, Temperature |
| PCV431C | AOV (F.C.) | Pressurizer Power Operated Relief | Limit Switches O&C, Temperature |
| PCV434 | Safety | Pressurizer Safety | Temperature |
| PCV435 | Safety | Pressurizer Safety | Temperature |
| V515 | MOV | Pressurizer Relief Block | Limit Switches O&C |
| V516 | MOV | Pressurizer Relief Block | Limit Switches O&C |
| V521 | AOV (F.O.) | Leakoff Drain Valve | Limit Switches O&C |
| V700 | MOV | RHR | Limit Switches O&C |
| V701 | MOV | RHR | Limit Switches O&C |
| V720 | MOV | RHR to RCS | Limit Switches O&C |
| V721 | MOV | RHR to RCS | Limit Switches O&C |

Section 2.1.3.b - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING IN PWR'S

TASK FORCE POSITION

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

(Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

The Westinghouse Owners Group, of which RG&E is a member, is performing calculations associated with the definition and recognition of inadequate core cooling conditions in accordance with Item 2.1.9. A description of the program and relations to this item is found in our response to Item 2.1.9.

Ginna Emergency Procedure E-1.5, Void Formation in the Reactor Coolant System, is an existing procedure developed to aid the operator in dealing with the detection of inadequate core cooling and gives the necessary actions required to deal with this emergency. Currently available instrumentation used in this procedure to recognize inadequate core cooling include core exit thermocouple, RCS pressure, and RCS flow. RCP current indication is available directly on ammeters at the respective RCP breakers and indirectly in the control room via respective bus ammeters.

Information presently available to operating personnel from the plant computer includes:

1. Correlation between RCS temperature and saturation pressure in a special alarm message consisting of RCS pressure, 50°F subcooled pressure, saturation pressure and reactor coolant loop temperature. The first alarm/return message occurs at 50°F subcooled pressure.

2. Core exit thermocouple instantaneous values.

3. Core exit thermocouple history maps with a variable collection and storage frequency.

Operating procedures will indicate that this information is to be used in conjunction with other monitored variables to assure proper subcooling.

Subcooling Meter Installation

The hot leg temperature in each loop will be directly monitored.

A programmable controller will provide $T_{SAT}$ as a function of primary pressure, based on a polynomial approximation of the saturation curve. Pressure signals will be provided by existing pressure transmitters which provide signals to the Safeguards Actuation System.

An error signal indicating the subcooling margin, $T_{SAT}-T_H$, will be generated and displayed in the main control room. An integrated bistable unit will provide an alarm when the subcooling margins are less than 50°F.

The primary signals for this system will be derived from qualified, redundant safety grade circuits. Separation and isolation will be provided to ensure that safety system independence is not degraded.

The installation of this system is scheduled to be complete by January 1, 1980.

POSITION

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

RG&E Response

A description of the program and commitment to meet this item can be found in our response to Item 2.1.9.

Section 2.1.4 - CONTAINMENT ISOLATION PROVISIONS FOR PWRS AND BWRS

TASK FORCE POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and nonessential systems, shall identify each system determined to be essential, shall identify each system determined to be nonessential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.

3. All nonessential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

RG&E Response

1. The existing containment isolation and containment ventilation isolation functions occur on diverse signals in accordance with SRP 6.2.4.

2,3. The subject of containment isolation of nonessential lines was discussed in our responses to IE Bulletin 79-06A dated April 28, 1979 and June 22, 1979. Specifically, in response to item 4 of IE Bulletin 79-06A, we provided a listing of all valves which are isolated on containment isolation or containment ventilation isolation. Further, we stated that remaining lines are required for safety functions or for reactor coolant pump operability. We concluded that no changes were required with respect to containment isolation. Thus, this item is complete.

4. The effect of resetting containment isolation and containment ventilation isolation was discussed in detail in our responses to item 9 of IE Bulletin 79-06A dated April 28, 1979 and June 22, 1979 and in our letters to the NRC dated January 2, 1979, February 16, 1979 and March 30, 1979. As identified in those letters, there are certain valves which could reopen upon reset of the containment isolation or containment ventilation isolation if their controllers were set in the open position.

The reopening of valves is currently precluded by several means. First, the operator is directed to place all valve position controllers in the closed position so that no valve will open on initiation of the reset. The reset of containment ventilation isolation can be actuated only through use of a key switch. The key is under the control of the shift foreman. Therefore, no single operator error can result in improper use of this reset function. The reset for containment isolation, at present, is a reset button. This reset button will be replaced with a key switch by November 1, 1979. To further reduce the likelihood of inadvertent reopening of valves, a modification is currently under design to eliminate the possibility of valves opening on reset and is scheduled for implementation during the March 1980 refueling outage. This schedule is acceptable based on the current administrative controls as described above and in the letters referenced above.

Section 2.1.5.a - DEDICATED PENETRATIONS FOR EXTERNAL RECOMBINER OR POST-ACCIDENT EXTERNAL PURGE SYSTEM

TASK FORCE POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system. (Description and implementation schedule is Category A: Implementation shall be completed by January 1, 1980. Category B: Implementation shall be completed by January 1, 1981).

RG&E Response

Ginna Station has two hydrogen recombiners which are located inside containment. Therefore, dedicated penetrations are not required.

Section 2.1.5.c - <u>CAPABILITY TO INSTALL HYDROGEN RECOMBINERS AT EACH LIGHT WATER NUCLEAR POWER PLANT</u>

<u>TASK FORCE POSITION</u>

The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2. (Category A: Implementation shall be completed by January 1, 1980.)

<u>RG&E Response</u>

Procedures are presently available which govern the use of the hydrogen recombiners. In view of the fact that our recombiners are inside containment, no further review is required to consider shielding requirements and personnel exposure limitations resulting from the recombiners. Access to the control panel will be considered in response to Section 2.1.6.b.

Section 2.1.6.a - INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS (ENGINEERED SAFETY SYSTEMS & AUXILIARY SYSTEMS)

TASK FORCE POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as practical levels.

This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as practical levels. This program shall include periodic intergrated leak tests at a frequency not to exceed refueling cycle intervals.

(Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

1. All practical leak reduction measures have been normal practice for the systems that carry radioactive fluids outside of containment.

2. An adequate preventive maintenance program is presently being implemented for pumps, compressors, and other equipment on these systems.

3. A preventive maintenance program for valves on these systems will be developed by January 1, 1980.

The following systems have been identified as systems which process primary coolant, and could contain high level radioactive materials.

1. Residual Heat Removal (RHR)
2. Containment Spray (CS) (portion of the system only)
3. Safety Injection (SI) (portion of the system only)
4. Chemical and Volume Control System (CVCS)
5. Primary Sampling
6. Waste Gas
7. Liquid Waste

Tests will be performed on the RHR, CS, and SI systems with the system in operation to the extent practical to determine applicable potential leakage paths. Visual inspections will be made and actual leakage rates will be measured and reported by January 1, 1980. Appropriate measures will be taken up to reduce undesirable leakage paths. Further inspection will be included in surveillance schedules.

A hydrostatic pressure test was conducted on the RHR system (including the SI recirculation system) during our 1979 annual refueling shutdown. A visual inspection of the system was made to locate leakage paths. Applicable maintenance was performed at that time to eliminate all visible system leakage.

Portions of the CVCS system which are in service during normal operation (makeup and letdown) are monitored as part of a daily leakage surveillance program to control the reactor coolant system water inventory. The average total leakage rate for the reactor coolant system has been less than 0.40 GPM of which approximately 0.14 GPM is identifiable and associated with the Reactor Coolant Drain Tank pumps and the charging pumps. A visual inspection will be made on those portions of the CVCS system which are located outside of containment. The leakage from applicable flow paths will be noted and actual rates measured and reported by January 1, 1980.

Visual leakage inspection was performed on September 13, 1979 of all high energy lines outside containment and no leakage was observed.

A visual inspection will be made to determine any liquid leakage from the sampling system and radioactive liquid waste system. If any leakage is detected, leakage rates will be determined and reported by January 1, 1980.

Excessive gaseous leakage from the primary sampling system, and gas waste systems would be indicated by abnormal airborne radioactivity levels in the spaces occupied by the systems. Stack monitors and constant air monitors located throughout the plant continuously monitor airborne activity. If increased levels are observed, leakage paths are located and appropriate measures are employed to reduce noted leakage.

At the present time we do not have a satisfactory method for determining the leakage rate for the waste gas system. We will continue to investigate possible methods of meeting the NRC concerns. We will report to the NRC by January 1, 1980 on our resolution of this issue and how we have addressed the NRC concerns.

As part of the In-Service Inspection Program requirements, containment isolation valve leak rate tests are performed during each refueling shutdown to determine possible leakage. These tests were conducted during our 1979 shutdown and results will be reported by January 1, 1980.

As required by Technical Specifications and Appendix J to 10 CFR Part 50, leak rate tests are performed annually on all containment penetrations. These tests were conducted during our 1979 shutdown and results will be reported by January 1, 1980.

Section 2.1.6.b - DESIGN REVIEW OF PLANT SHIELDING AND ENVIRON-QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS

TASK FORCE POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility. (Design review, Category A: completion by January 1, 1980; plant modification, Category B: Implementation shall be completed by January 1, 1981.)

RG&E Response

We will perform the design review as requested by the NRC. We expect to implement plant shielding modifications and procedure changes which may be required by January 1, 1981 unless major modifications which are affected by Systematic Evaluation Program (SEP) topies are identified. There may be a selected number of modifications which we will recommend including in the integrated assessment of the SEP. This would be in those cases where the modification could potentially interact with reviews being conducted under SEP such as topics III-4.A, III-4.B, III-4.C, III-4.D (missiles), III-5.B (pipe break outside containment), III-6 (seismic), and VI-8 (control room habitability). Recommendations for inclusion in SEP will be reached on a case-by-case basis and will be presented to the NRC for concurrence.

Section 2.1.7.a - AUTOMATIC INITIATION OF THE AUXILIARY FEEDWATER SYSTEM

TASK FORCE POSITION

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term: (Category A: Implementation shall be completed by January 1, 1980.)

POSITION 1.

The design shall provide for the automatic initiation of the auxiliary feedwater system.

RG&E Response

The following conditions will initiate motor driven auxiliary feedwater pump start:

1. 2/3 low level in either Steam Generator
2. Opening of both Main Feedwater Pump Circuit Breakers
3. Any Safety Injection Signal
4. Manual Start

The following conditions will initiate turbine driven auxiliary feedwater pump start:

1. Coincidence of 2/3 low level in both Steam Generators
2. Loss of Voltage on both 4160 volt buses (Station Blackout)
3. Manual Start

When the motor driven auxiliary feedwater pumps start, their associated discharge valves receive a signal to open fully and then throttle back to approximately 230 gpm flow. The turbine driven auxiliary feedwater pump steam supply valves and pump discharge valves receive their signal to open on pump start.

The remaining system valves are normally open and require no actions for system operation.

POSITION 2.

The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

RG&E Response

The auxiliary feedwater initiation circuitry is part of the Engineered Safety Features (ESF) system, and is capable of accommodating any single failure of an active component.

POSITION 3.

Testability of the initiating signals and circuits shall be a feature of the design.

RG&E Response

The auxiliary feedwater initiation signals and circuitry are testable. Such testability is included in the surveillance test procedures for the plant and is delineated in the Plant Technical Specifications.

POSITION 4.

The initiating signals and circuits shall be powered from the emergency buses.

RG&E Response

The initiating signals and circuits are powered from the emergency buses.

POSITION 5.

Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system functions.

RG&E Response

Manual initiation for each train exists in the control room. The manual initiation system is installed in the same manner as the automatic initiating system. No single failure in the manual initiation portion of the circuit can result in the loss of auxiliary feedwater system function.

POSITION 6.

The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

RG&E Response

The a-c motor-driven pumps and all required valves in the system are automatically transferred to, and sequentially loaded on, the emergency buses on loss of offsite power. The sequence is shown in FSAR Table 8.2-4.

POSITION 7.

The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

RG&E Response

All automatic initiating signals and circuits are installed in accordance with regulatory requirements and are safety grade and redundant. No single failure in the automatic portion of the system will result in the loss of the capability to manually initiate the AFWS from the control room.

POSITION 8.

In the Long Term, the automatic initiation signals and circuits shall be upgraded in accordance with safety grade requirement.

RG&E Response

As described above in the preceding responses, the automatic initiating signals presently meet all safety grade requirements. No upgrading is required.

Additional Information

The Ginna auxiliary feedwater system consists of two 100% motor driven auxiliary feedwater pumps and one 200% turbine driven auxiliary feedwater pump. This system is discussed above. Further, two additional 100% motor-driven auxiliary feedwater pumps are available in a standby auxiliary feedwater system and may be manually loaded onto the diesels if for any reason the automatically initiated auxiliary feedwater system is not available. The complete system has been extensively discussed with the NRC Bulletins and Orders Task Force and, in addition, was the subject of License Amendment No. 29 issued by the NRC on August 24, 1979. These systems fully satisfy this NRC position.

Section 2.1.7.b - AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS FOR PWRS

TASK FORCE POSITION

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

(Category A: Implementation of control grade equipment shall be completed by January 1, 1980; Category B: Implementation of safety grade equipment shall be completed by January 1, 1981.)

RG&E Response

The existing flow indication for each steam generator complies with the requirements for "control grade" systems.

Qualified flow indication for each steam generator will be installed. The power for each system will be supplied by the train that supplies the associated auxiliary feedwater pump. Installation of this equipment is scheduled prior to January 1, 1981.

Section 2.1.8.a - IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

TASK FORCE POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being performed promptly, i.e., the boron analysis within an hour and the chloride sample analysis within a shift. (Design review and procedures, Category A: Implementation shall be completed by January 1, 1980; Modifications, Category B: Implementation shall be completed by January 1, 1981.)

RG&E Response

A review of the design of the sampling systems for containment atmosphere and primary coolant and the operations involved in collecting samples will be conducted prior to January 1, 1980. This review will assess the capability for obtaining the samples and transporting them to the laboratory under post-accident conditions, without incurring radiation exposure to any individual in excess of 3 Rem whole body

dose or 18 3/4 Rem extremity dose. Additional shielding or design changes will be proposed if found necessary with modifications completed during 1980 with the possible exception of selected major modifications that may be affected by SEP review. These may be incorporated into the Systematic Evaluation Program (see response to Section 2.1.6.b).

The capability for performing radiological spectrum analysis will be reviewed in relation to quantifying the degree of core damage by January 1, 1980. The review will consider the effects of direct radiation sources and possible airborne contamination in the existing count room. If the review indicates that the required analysis cannot be performed, alternate locations or equipment will be established by January 1, 1981.

Procedures for making a boron and chloride analysis on highly radioactive samples will be reviewed and revised insofar as possible given existing equipment configuration by January 1, 1980. We are currently using a remote controlled burette for boron analysis and selective electrode for chloride, both of which aid in maintaining doses in the laboratory as low as reasonably achievable. If the review indicates that the required analyses cannot be performed, alternate locations or equipment will be established by January 1, 1981.

Section 2.1.8.b - INCREASED RANGE OF RADIATION MONITORS

TASK FORCE POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

POSITION 1.

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

a. Noble gas effluent monitors with an upper range capacity of $10^5$ uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of $10^5$ uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

RG&E Response

The current noble gas effluent monitors in use on the plant vent at Ginna Station have a range from $10^{-6}$ uCi/cc to $10^{-3}$ uCi/cc. We are currently investigating the availability of the suggested level of monitoring. The appropriate equipment, if available, will be installed by January 1, 1981 with back up power source for continuous display and recording.

Procedures for estimating the rate of noble gas releases based upon portable instrumentation will be developed and approved prior to January 1, 1980 for use if currently in place equipment is indicating above scale releases.

POSITION 2

Since iodine gaseous effluent monitors for the accident conditions are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by on-site laboratory analysis.

RG&E Response

RG&E currently has the capability to monitor iodine gaseous releases. This equipment is in place and used for routine release reports of iodine isotopes.

POSITION 3.

In-containment radiation level monitors with a maximum range of $10^8$ rad/hr [total or $10^7$ rad/hr photon] shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

RG&E Response

We currently have installed in containment a radiation monitor with readout in the control room which has a range up to $10^4$ rads/hr.

Two redundant and independent containment radiation monitors with range up to $10^7$ rad/hr (photon only) will be installed. A continuous display of each channel will be provided and one channel will be recorded. This system will be designed in accordance with Regulatory Guide 1.97.

The system will be designed and procured to support installation by January 1, 1981, however, actual installation will be dependent on availability of the plant containment for installation.

Section 2.1.8.c - IN PLANT IODINE INSTRUMENTATION

TASK FORCE POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determing the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

RG&E Response

We now have portable instrumentation located in various areas throughout the plant which monitors airborne iodine concentrations in work areas. The isotopic analysis is made by GeLi detector and used to determine MPC hours of exposure. Chemistry and Health Physics technicians are trained in the use of this equipment and use it in routine work. Thus, no further action is required.

Section 2.1.9.a - ANALYSIS OF DESIGN AND OFF-NORMAL TRANSIENTS AND ACCIDENTS

TASK FORCE POSITION

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long-term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of

the operators to perform required control manipulation shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long-term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to intiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of cure uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

RG&E Response

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analysis of transient and accident scenarios including operator actions not previously analysed are being performed on a generic basis by the Westinghouse Owners Group, of which RG&E is a member. The small break analysis has been completed and reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners Group on June 29, 1979. Incorporated in that report were guidelines that were developed as a result of the small break analyses. These guidelines have been reviewed with the B&O Task Force and will be presented to the Owners Group utility representatives at a seminar to be held on October 16-19, 1979. Following this seminar, we will develop plant specific procedures and train personnel on the new procedures. It is intended that the revised procedures and training will be in place by January 1, 1980, in accordance with the requirement in Enclosure 6 to Mr. Eisenhut's letter of September 13.

The work required to address the other two areas -- inadequate core cooling and other transient and accident scenarios --is being performed in conjunction with the Bulletins and Orders Task Force, including establishment of information requirements to meet the duties specified in Enclosure 6. Analyses related to the definition of inadequate core cooling and guidelines for recognizing the symptoms of inadequate core cooling based on existing plant instrumentation and recovery from such a condition will be provided by the Owners Group by October 31. Further work to better define the approaches to inadequate core cooling and recovery operations may be required and will be performed later. It is intended that the guidelines provided by October 31, 1979, will be incorporated into plant procedures and training accomplished by the required date of January 1, 1980.

In the course of performing these analyses and developing companion guidelines, it is possible that a need for additional instrumentation will be identified. Should this occur, we will notify you of our schedule for the procurement and installation of any instrumentation for which a need is identified.

The work related to other transients and accidents contained in Chapter 14 of the Ginna FSAR will be provided by the required date of January 1, 1980.

The Owners Group is also providing pretest prediction analysis of the LOFT L3-1 nuclear small break experiment. This analysis will be submitted by the Owners Group by the required date of November 15, 1979, in accordance with the schedule established by the B&O Task Force.

Section 2.1.9 - CONTAINMENT PRESSURE MONITOR

TASK FORCE POSITION

A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments. (Category B: Implementation completed by January 1, 1981.)

RG&E Response

There are six existing, safety grade, containment pressure channels. Three pressure transmitters have a calibrated range of 0-60 psig while the remaining three have a calibrated range of 0-90 psig. The design basis accident load on the Ginna concrete containment structure is 60 psig.

One independent containment pressure channel with calibrated range -5 to 180 psig designed in accordance with Regulatory Guide 1.97 will be installed by January 1, 1981. This channel will provide continuous indication in the control room.

## Section 2.1.9 - CONTAINMENT WATER LEVEL MONITOR

### TASK FORCE POSITION

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. Also for PWRs, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool. (Category B: Implementation completed by January 1, 1981.)

### RG&E Response

There are two redundant existing safety grade sump level indication channels. The existing sump level channels are in compliance with Task Force recommendations for narrow range indication. A wide range channel will be designed and procured to support installation by January 1, 1981, however, actual installation will be dependent on availability of the plant containment for installation.

Section 2.1.9 - INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT MONITORING OF HYDROGEN CONCENTRATION IN CONTAINMENT

TASK FORCE POSITION

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure. (Category B: Implementation completed by January 1, 1981.)

RG&E Response

Containment hydrogen indication in the main control room will be designed and procured to support installation in accordance with Task Force recommendations by January 1, 1981, however, actual installation will be dependent upon availability of the plant and the required equipment.

Section 2.1.9 - INSTALLATION OF REMOTELY OPERATED HIGH POINT VENTS IN THE REACTOR COOLANT SYSTEM

TASK FORCE POSITION

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses demonstrating that the direct venting of non-condensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentrations limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1) and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

RG&E Response

Since the issuance of the September 13, 1979 letter from the NRC which delineated the requirements listed above, the NRC has revised the requirements.

We intend to provide at least a single vent line on both the reactor vessel head and the pressurizer. The vents will meet the NRC staff positions discussed in a topical meeting in Bethesda, Maryland on October 11, 1979. The systems will meet seismic criteria and will be safety grade but will not be redundant. At least one vent valve, and possibly two in series, will be provided in each line. Each valve will have direct position indication in the control room. New vent paths which do not make use of existing equipment will be small in size so that failure of this line will not result in system losses in excess of the capacity of the charging pumps.

A design for the system vents will be provided by January 1, 1980. The design and procurement of required valves and components will be expedited to support installation of the systems by January 1, 1981, however, actual installation will be dependent upon the availability of the plant and the required components.

Procedures will be developed by January 1, 1981 which give guidance for use of the vents.

Section 2.2.1.a - <u>SHIFT SUPERVISOR'S RESPONSIBILITIES</u>

<u>TASK FORCE POSITION 1.</u>

The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the Shift Supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

<u>RG&E Response</u>

The Vice President, Electric and Steam Production will issue, at least annually, a management directive that emphasizes the primary management responsibility of the Ginna Station Shift Foreman for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

<u>TASK FORCE POSITION 2.</u>

Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the Shift Supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

a. The responsibility and authority of the Shift Supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the Shift Supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

b. The Shift Supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the Shift Supervisor shall be specified.

c. If the Shift Supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

RG&E Response

Plant procedures have been reviewed to ascertain the extent to which duties, responsibilities and authority of the Ginna Station Shift Foreman and Control Room Operators are defined. A definite line of command and clear delineation of command decision authority of the Shift Foreman is provided in procedure A-52.1, Shift Organization Relief and Turnover, in its section on Control Room Operations.

This is reinforced in the specific listing of duties in procedure A-201 Ginna Station Administrative and Engineering Staff Responsibilities. Regarding the additional particular emphasis required, the following information is provided:

a. Although the above procedures are concerned with overseeing facility operations and operator performance and overall direction of operations activities, a revision will be provided to explicitly address the need for a broad perspective and avoidance of total involvement in any single operation.

b. The above procedures require two licensed operators to be stationed in the Control Room at all times. They specify the normal control room crew to be the Head Control Operator and the Control Operator, with provisions for temporary relief by a licensed operator. They indicate that the Shift Foreman should use the Control Room as his base for directing operations. This will be augmented to specify that during accident conditions he shall remain in the Control Room until properly relieved. Those authorized to relieve the Shift Foreman will be specified.

c. Although the job description for Control Room Operators clearly indicate the Control Room command function of the Head Control Operator, subject to the direction of the Shift Foreman, procedure A-52.1 will be augmented by January 1, 1980 to indicate this as well.

TASK FORCE POSITION 3.

Training programs for Shift Supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

RG&E Response

Training programs for Shift Supervisors will emphasize and reinforce the responsibility for safe operation and the management function that the Shift Supervisor is to provide for assuring safety.

TASK FORCE POSITION 4.

The administrative duties of the Shift Supervisor shall be reviewed by the senior officer of each utility responsiblie for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room. (Category A: Implementation completed by January 1, 1980.)

RG&E Response

Duties of the Ginna Station Shift Foreman have been reviewed by the Vice President, Electric and Steam Production in his review of each Position Analysis for the job classifications in his department. The administrative duties performed by the Shift Foreman will be reviewed and those not related to safe operation of the plant will be delegated to other personnel. This will be completed by January 1, 1980.

Section 2.2.1.b - <u>SHIFT TECHNICAL ADVISOR</u>

<u>TASK FORCE POSITION</u>

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

<u>RG&E Response</u>

RG&E believes that its present shift operation organization, augmented by an additional Licensed individual in the Control Room and further augmented by on-call Duty Engineers holding NRC Senior Reactor Operator Licenses and an expanded training program, represents a suitable alternative for accomplishing the objectives of the NRC Task Force position for a Shift Technical Advisor.

The Ginna Station Operation Department consists of five shifts with a minimum of five operators per shift, two of whom hold Senior Reactor Operator Licenses. The Shift Foremen have maintained their licenses since before January 1970 and have a broad knowledge of plant design and operation.

The Operation Department has a total of 164 Man-years of Licensed Operators experience. The present shift complements are normally comprised of a Shift Foreman (SRO), Head Control Room Operator (SRO), Control Room Operator (RO) and 3 Auxiliary operators (some with RO licenses). To meet the requirements of <u>Safety Monitoring Function</u> one additional licensed individual will be assigned to the control room when the reactor is critical and will be designated as Shift Technical Advisor. Thus the number of licensed individuals normally in the Control Room will be increased to 3. During the course of a reactor transient this additional licensed individual will be in the control room and will <u>only</u> be functioning in the Technical Advisory capacity to the Shift Foreman. The Shift Technical Advisor will administratively report to the Technical Assistant to the Plant Superintendent, Operations Assessment.

The concept of an on-call Duty Engineer has been in existence since our commercial operation in 1970. The individuals on-call are plant staff engineers and possess a NRC Senior Reactor Operators License. Recall of on-call personnel is effected through commercial telephone or through private paging devices.

Our Duty Engineers are assigned on a rotating basis 24 hours per day for a one week period and their responsibilities are defined by an Administrative procedure. This procedure outlines a weekly plant tour and a random backshift or weekend assessment of operations. The assessment of operations includes equipment status and plant conditions. In addition, any unusual events are immediately reported to the Duty Engineer for his evaluation.

At present we have seven Duty Engineers who have a broad technical knowledge in engineering, plant design, and operations comprised of forty four man-years of licensed operating experience. In addition, our Operations Engineer and Operations Supervisor reside within six miles of the plant and would be available on-site almost immediately to respond to an emergency situation.

To further facilitate the response of our duty engineer we intend to provide the duty engineer with a 4-wheel drive vehicle with a two-way radio to keep the duty engineer in close contact with the Control Room once he has been notified of an unusual condition while he is responding to the call and driving to the station.

We will initiate a training program by January 1, 1980 to augment the technical education of the assigned Shift Technical Advisors for the Safety Monitor Function in the Control Room. This program will be comprised of college level engineering courses. The program will include but not be limited to the following areas: mathematics, thermodynamics, heat transfer, fluid flow, corrosion of material, basic electrical engineering and reactor analyses. We have outlined the program and are presently discussing its implementation with various colleges and universities.

RG&E as a member of the Electric Power Research Institute (EPRI) will participate in the EPRI sponsored Institute of Nuclear Power Operations. Operators and supervisors will receive additional training appropriate to their responsibilities in the response of plant transients and accident when this program becomes applicable.

In addition to the Operation personnel, we have available a number of supervisory and technical support personnel who maintain Senior Reactor Operator licenses and possess a background in engineering. These individuals would be available to respond to emergency situations. The total

experience of the operations and support personnel includes 278 man-years of licensed operating experience.

The <u>Operating Experience Assessment Function</u> will be implemented by the designation of a Technical Assistant to the Plant Superintendent, Operations Assessment whose responsibility will be the engineering evaluation of plant operations at Ginna Station. He will be assisted by the shift technical advisors and will have access to the station's plant technical engineering support on an as-needed basis.

The Operations Assessment Assistant will report to the plant superintendent. He will make recommendations to the Superintendent in the areas of operations, operator training, and modifications affecting plant operations. He will meet with plant staff personnel on a periodic basis to discuss operating experiences.

Section 2.2.1.c - SHIFT AND RELIEF TURNOVER PROCEDURES

TASK FORCE POSITION

The licensee shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and off-going control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

    a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptance status shall be included on the checklist).

    c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

RG&E Response

Procedure A-52.1 Shift Organization Relief and Turnover, directs each Operator and the Shift Foreman in informing his relief of continuing procedures and unusual conditions. Presently, a checklist exists as an optional aid in providing this information, particularly during periods of diverse and intensified activity. This includes the use of jumper wires, the observations of malfunctions, equipment out of service, and temporary procedure changes in effect. This checklist will become mandatory and review by the oncoming and offgoing Control Room Operators and Shift Foremen will be attested to by their signatures.

a. The check list will be expanded to include a listing of critical plant parameters with allowable limits.

b. The check list will be expanded to incorporate checking the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents. This will be done by requiring completion of existing procedure O-6.2, Main Control Board System Status. This procedure is completed each shift and requires documentation of any system not in its prescribed configuration.

c. The check list will be expanded to incorporate identification of systems and components that are in a degraded mode of operation permitted by Technical Specifications. This will be done by referring to existing procedures A-52.4, Control of Limiting Conditions for Operating Equipment, and A-52.5, Control of Limiting Conditions for System Specifications. The length of time in a degraded mode will be identified.

The changes described above will be completed by January 1, 1980.

TASK FORCE POSITION

Checklist or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate operational transients (what to check and criteria for acceptable status will be included on the checklist.)

RG&E Response

Auxiliary Operators and Shift Health Physics Technicians maintain logs of the activities they perform for the purpose of informing their reliefs of activities performed or in progress and unusual conditions. A checklist will be provided to indicate equipment to be checked and criteria for acceptable status. This will be accomplished by January 1, 1980. No other personnel are on shift.

TASK FORCE POSITION

A system shall be established to evaluate the effectiveness of the shift and relief turnover procedures (for example, periodic independent verification of system alignments).

RG&E Response

A system will be established to evaluate the effectiveness of the turnover procedures. The system will involve periodic independent verification of system alignments. This will be completed by January 1, 1980.

Section 2.2.2.a - CONTROL ROOM ACCESS

TASK FORCE POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of any emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

   (Category A: Implementation shall be completed by January 1, 1980.)

RG&E Response

An Administrative Procedure will be prepared to formalize existing policies which allow the Shift Foreman to restrict access to the control room during both normal operation and emergencies. This will be implemented by January 1, 1980.

Procedures will be written or revised as necessary to establish authority and responsibilities in the control room in the event of an emergency, and provide for line of succession for the person in charge of the plant operations. The line of succession will be limited to those who possess a current NRC Senior Operator's License. Existing procedures also specify the locations to which all on-site personnel are to report in the event of an emergency and these will be changed as necessary to reflect the task force recommendations. The above will be implemented by January 1, 1980. Procedures to incorporate the changes necessitated by the establishment of the On-Site Technical and Operational Support Centers with respect to the Task Force recommendations will be developed and implemented upon establishment of these centers.

Section 2.2.2.b - ONSITE TECHNICAL SUPPORT CENTER

TASK FORCE POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay.

RG&E Response

An interim on-site technical support center will be established by January 1, 1980. A permanent on-site technical support center will be established by January 1, 1981.

## Section 2.2.2.c - ONSITE OPERATIONAL SUPPORT CENTER

### TASK FORCE POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

### RG&E Response

Prior to January 1, 1980 an onsite operational support center will be designated. This area will have communication from the control room, technical support center and emergency survey center.. The emergency plans will be revised to indicate those individuals who should report to this center.

## IMPROVING EMERGENCY PREPAREDNESS

### TASK FORCE POSITION

1. Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.

2. Assure that implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

3. Determine that an emergency operations center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant technical support center is underway.

4. Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

5. Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

6. Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As a preliminary planning basis, assuming that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

RG&E Response

Our current emergency plan and implementing procedures were written in 1974 to meet the then current guidelines of NRC, at which time it was considered a model plan. Prior to January 1, 1980 our emergency plan will be reviewed against Regulatory Guide 1.101 and upgraded as necessary.

The short term actions recommended by Lessons Learned Task Force will be implemented as stated in appropriate sections. Criteria to be used as action levels will be factored into emergency plans as the instruments become available for control room use in monitoring an accident.

After consultation and further guidance from the NRC an appropriate operations center for Federal, State & Local officials will be established. Necessary communications will be completed by August 1, 1980. Any modifications necessary to our Emergency Survey Center in conjunction with the Technical Support Center will be completed prior to January 1, 1981.

Offsite monitoring capability has been upgraded by the scheduled placement of thermoluminescent dosimeters (TLDs) at locations in the environs by the initial survey teams. This procedure change was approved July 1979. Further, additional TLDs have been placed on a routine basis in the environs. Further improvements wil be made in implementing procedures as necessary to meet the review by NRC Emergency Planning Review Team. These will be implemented by August 1, 1980.

The New York State Emergency Plan for Radiation Accidents has been reviewed by the NRC and has received NRC concurrence. If further changes in light of new criteria are found necessary by the NRC we will work with the State and local organizations to revise their plans.

Ginna Station's Radiation Emergency Plan has been tested annually since 1975. The communications with state and local officials has been tested as part of the drill. The drill for 1979 is scheduled for this month.

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