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SUBJECT: Forwards "Environ Qualification of Electrical Equipment," Revision 2,in response to NRC 800306 & 28 requests.

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LEON D. WHITE, JR.
VICE PRESIDENT

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April 25, 1980

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: Environmental Qualification of Electrical Equipment
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

This submittal is in response to NRC letters from Dennis L. Ziemann to L.D. White, Jr., RG&E, dated March 6 and March 28, 1980 requesting information concerning the environmental qualification of the Ginna Class IE electrical equipment. We have already responded to requests in these letters for the Ginna Emergency Procedures and Containment Environmental Conditions in letter of February 26, 1980 and April 10, 1980, respectively.

The enclosed submittal is revision 2 of the RG&E document "Environmental Qualification of Electrical Equipment," originally submitted to the NRC on February 24, 1978, and updated December 1, 1978. It addresses the new NRC "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors," and also incorporates information regarding modifications made at Ginna since the earlier submittals.

Very truly yours,

L D White, Jr.

L. D. White, Jr.

LDW:rb

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Subscribed and sworn to me
on this 25th day of April 1980.

Rose Marie Perrone

ROSE MARIE PERRONE
NOTARY PUBLIC, State of N. Y., Monroe County
My Commission Expires March 30, 1982.

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Environmental Qualification of Electrical Equipment

R.E. Ginna Nuclear Power Plant
Docket No. 50-244

February 24, 1978

Rev. 1, December 1, 1978

Rev. 2, April 25, 1980

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Environmental Qualification of Safety-Related Electrical Equipment

I. INTRODUCTION

The equipment and systems identified in this report are classified in accordance with IEEE 308-1974 as Class IE.

The purpose of this report is to establish the level of environmental qualification of the Class IE equipment required during and after a loss of coolant accident (LOCA), main steam line break (MSLB), main feedwater line break (MFLB) or high energy line break outside containment in a format which provides for convenient comparison with the limiting design environment in which it is required to perform a safety function. Table 3 summarizes this information.

II. IDENTIFICATION OF NECESSARY SAFETY RELATED EQUIPMENT

This section of the report identifies the necessary safety related equipment for each of the Design Basis Events (DBE) of concern and a brief description of why the equipment is needed. This identification includes all electrical equipment required by the Ginna emergency procedures for accomplishing the necessary safety functions. It must be recognized that not all electrical equipment referenced in the procedures is required to function (as opposed to being useful if available), and is therefore not required to be qualified.

A. Events Accompanying a Loss Of Coolant Accident

Analyses of the course and consequences of loss of coolant accidents have been submitted previously (LOCA 1-4). A discussion of equipment required to function to mitigate the consequences of a loss of coolant accident is presented in the FSAR Chapters 6, 7 and 14. Post-LOCA operator actions are included in the Ginna Emergency Procedures. Additional descriptive material is presented in this report to provide summary information as to the sequence of events and the equipment involved at each stage. Figure 1 illustrates the sequence of events following a loss of coolant accident. Table 1 provides a specific equipment list for each numbered block in Figure 1. Also provided in Table 1 is the safety function which is required and the period of time that operability must be ensured. It should be noted that Table 1 includes all redundant equipment,

not the minimum safeguards equipment assumed in the safety analysis. In the "required" column it should be noted that equipment listed as "signal initiation" is required to be operable only until its required safety function is performed. Equipment listed as "long term" is required to provide long term decay heat removal, post-accident monitoring and sampling, or achieving and maintaining a safe shutdown condition. Equipment listed as "short term" is required only for a short period of time (hours-days).

Table 3 provides environmental qualification of all the Ginna Class IE equipment, in the format requested for SEP by the NRC's September 6, 1978 letter.

1. The first event in the loss of coolant accident following the rupture is the detection of the rupture. Any 2/3 low pressurizer pressure or 2/3 high containment pressure will initiate "safety injection".

1a. Instrumentation is available to the operator to distinguish between a LOCA and the other accidents, such as a steam line break or feed line break. It is important to note that the automatic actions and immediate operator actions (first 10 minutes) are identical in the mitigation of these accidents.

2. Upon "safety injection" signal generation, safeguards sequencing is initiated (see FSAR Table 8.2-4). The diesel generators start and energize the safeguards busses assuming there is a loss of offsite power. With

the safeguards busses energized, either by off-site power or the diesels, the three safety injection pumps, the two residual heat removal pumps, two of the four service water pumps, the two auxiliary feedwater pumps, and the four containment fan coolers will be loaded sequentially onto the busses. The two containment spray pumps are automatically loaded onto the busses when the appropriate containment pressure setpoint is reached.

3. A break in the reactor coolant system piping actuates the passive accumulator injection system when the reactor coolant system pressure is reduced to 700 psig. The flow path of the borated water from each accumulator is through a series of check valves and a normally locked open (with AC control power removed) motor operated valve. The motor operated valves, MOV 841 and MOV 865, are not required to function to mitigate the consequences of the accident [Flood-1].

4. The main steam isolation valves 3516 and 3517 close upon receiving a high containment pressure signal and the main and bypass feedwater control valves 4269, 4270, 4271 and 4272 close upon receiving a safety injection signal. The SI signal also causes a trip of the main feedwater pumps (which in turn causes the closing of the feedwater discharge valves).

5. "Containment Isolation" and "Containment Ventilation Isolation" (referred to as simply "Containment Isolation") is initiated by the safety injection signal. Containment isolation is discussed in detail in Section 5.2 of the FSAR. Most of the containment isolation valves are air operated valves. All air operated containment isolation valves close with safety injection signal with the exception of valves 4561 and 4562 which open full to insure service water supply to the containment recirculation fans. The fail safe position of the valves is the desired safeguard position as described above.

Six motor operated valves (313, 813, 814, ATV-1, ATV-2, ATV-3) receive a containment isolation signal. All of these valves are located outside of containment and only valves 313, 813, and 814 are fed from the safeguards busses.

During normal operation ATV-1, ATV-2, and ATV-3 are closed with blank flanges installed on their respective penetrations inside containment. The use of the process lines associated with these valves occurs only during the containment building integrated leak rate tests.

Valve 313, the reactor coolant pumps seal water return line, and valves 813 and 814, reactor coolant support inlet and outlet lines, are closed by the containment isolation signal.

6. The S.I. signal trips the reactor and turbine. Other reactor trips are discussed in the FSAR, Section 7.

7. The reactor coolant pumps are tripped by manual operator action when low pressurizer pressure (1715 psig) is reached, and SI flow is initiated.

8. Selected valves throughout the plant provide flow paths for the required safeguards equipment with the advent of the S.I. signal.

During normal operation all required valves in the flow paths for high head safety injection are normally open with the exception of valves 826A and 826C, the discharge valves from the boric acid storage tank to the suction of the safety injection pumps.

Valves 826A, B, C and D receive the safety injection signal and valves 826A & C open providing borated water to the reactor coolant loop cold legs.

When the level in the boric acid storage tank decreases to the 10% level, suction for the high head safety injection pumps is automatically switched from the boric acid storage tanks to the refueling water storage tank by the automatic opening of valves 825A and B and closing of valves 826A, B, C and D.

During normal operation, all valves in the flow paths for low head safety injection are normally open except for MOV 852A and MOV 852B, the valves in the vessel

injection lines. These valves open upon receipt of a safety injection signal and remain open thereafter.

The containment spray pumps will automatically start and the discharge valves 860A, B, C and D automatically open, receiving power from the safeguards busses when containment pressure reaches 30 psig. If containment pressure does not reach 30 psig, the operator may manually start the spray pumps after all other safeguards are loaded on the safeguards busses. Automatic NaOH addition via opening of valves HCV 836A, B takes place two minutes after containment spray pump start unless defeated manually.

The containment spray pumps are normally aligned to the refueling water storage tank with all suction valves open.

SIS actuation will automatically align the two post accident charcoal filters to the containment recirculation system by opening inlet valves 5871 and 5872, and outlet valves 5873 and 5874. Loop entry dampers 5875 and 5876 will close.

9. The control room ventilation is automatically placed in the 100% recirculation mode (with about 25% flow through charcoal filters), when SI is initiated.

10. After the safety injection pumps are automatically switched from the boric acid storage tanks to the refueling water storage tanks, the operator resets safety

injection, starts the component cooling water pumps and aligns flow to the RHR heat exchangers, and initiates SW flow to the CCW heat exchangers. At the 31% RWST alarm, the operator shuts off one CS and one SI pump (if more than one are running. When the refueling water storage tank level is reduced to 10%, the plant operator stops the remaining residual heat removal, containment spray and high head safety injection pumps and establishes flow paths to the reactor vessel for both high and low head safety injection from containment sump B.

The normal (non-safety grade) auxiliary feedwater supply source is from the condensate storage tanks. If this supply is exhausted the operator opens motor operated valves 4027 and 4028 and manual operated valves 4344 and 4345 to provide service water to the suction of the auxiliary feedwater pumps. If the AFW system is not functioning properly, the operator can align the Stand-by AFW system to the SG's (using Service Water suction) from the control room.

11. In the recirculation phase, the operator aligns the RHR pumps to containment sump B by opening valve 850A for pump A and valve 850B for pump B, and closing valve 704A, B, 856, and 896A or B. For low head recirculation, injection is through the vessel nozzles. For high head recirculation, the RHR pumps discharge to the safety injection pumps through alignment of valve 857A (for RHR pump B) and/or valves 857B and 857C (for RHR pump A).

Valves AOV 897, 898 are closed. The high head safety injection pumps then provide water to the cold leg injection points. This alignment also allows CS pump operation, if desired.

Long term recirculation to compensate for the possible effects of boron precipitation has been described in Ref [Flood-1] and includes the use of RHR pumped flow to the vessel nozzles and through a high head safety injection pump into either cold leg.

Post-accident reactor coolant and containment atmosphere sampling modifications are presently being undertaken, in accordance with the implementation schedule for the TMI Lessons Learned commitments. See [Ref TMI-1].

B. Events Accompanying a Main Steam Line Break or a Main Feed Line Break

The analyses of a main steam line break or a main feed line break and the consequences thereof have been discussed in Chapters 6 and 14 of the FSAR and in References [SLB/FLB 2-4]. The High Energy Line Break Analyses [HELB 1-7] provides additional analysis for steam line breaks outside of containment, as well as feedwater line breaks inside and outside containment.

Figure 2 illustrates the sequence of events required to mitigate the consequences of a main steam line break. The same initial sequence of events would occur for a feedwater line break. Since the same equipment is required to operate, and the same emergency procedure is

used to mitigate, a feedline break as a steam line break, but a steam line break is a more severe accident in terms of RCS cooldown (return to criticality) and mass and energy release to containment, the subsequent discussion will include the main steam line break only. Table 2 lists the required equipment for each numbered block in Figure 2.

1. A large main steam line break (greater than approximately one square foot) would first be detected by the low steam line pressure sensors. Low steam line pressure sensed by two out of the three steam line pressure transmitters initiates safety injection accompanied by reactor and turbine trip.

1a. Diagnostic instrumentation is available to the operator to distinguish among accidents, as described in the LOCA discussion.

2. Two out of three low pressurizer pressure signals would provide additional protection for a larger steam line break and also provides the initial safety injection signal for smaller breaks.

3. The Ginna design includes non-return check valves in each steam line just upstream of the main steam header in the intermediate building. Thus for any break upstream of the check valves, which includes all breaks inside containment, the check valves will preclude blowdown of the intact generator. Reactor trip will

result in closing the turbine stop valves. As redundant protection in the event of a steam line break upstream of the check valves, and for all breaks downstream of the check valves, the main steam line isolation valves are closed by several signals. These signals include 2/3 high containment pressure (20 psig); 1/2 high steam flow in either steam line plus 2/4 low Tave plus safety injection; and 1/2 high-high steam flow in either steam line plus safety injection. Additionally, high containment pressure (30 psig) will initiate Safety Injection.

4. The safety injection signal closes the main and bypass feedwater control valves, trips the feedwater pumps and closes their respective discharge valves.

5. The safety injection signal initiates containment isolation and containment ventilation isolation as described in the sequence of events in the loss of coolant accident.

6. The safeguards sequence as described in the loss of coolant accident is initiated by the safety injection signal. (For steam breaks outside containment, the spray pumps are not required.)

7. The safety injection signal trips the reactor and turbine. Other reactor trips are discussed in the FSAR, Section 7.

8. The reactor coolant pumps are tripped by manual operator action when low pressurizer pressure (1715 psig) is reached, and SI flow is initiated.

9. All valves associated with the safety injection systems are aligned and automatically function as described in the loss of coolant accident discussion. If high containment pressure of 30 psig is reached, the Containment Isolation and Containment Ventilation Isolation valves perform as described in the LOCA discussion.

10. When the boric acid storage tanks are drained to the 10% level and safety injection pump suction has automatically been aligned to the refueling water storage tank, the operator will reset safety injection and if reactor coolant pressure is above the shut-off head of the RHR pumps, will stop the RHR pumps and place them in the standby mode.

For a main steam line break inside containment the operator may start the containment spray pumps manually if containment pressure is below 30 psig.

A high steam line flow and/or low steam line pressure will indicate to the operator which steam generator has the steam line break. When this has been determined, the operator will terminate AFW flow to the faulted steam generator.

The inventory of the reactor coolant will be maintained by the "remote manual" operation of the high head safety injection pumps in combination with use of the charging pumps.

At least two hours after the start of the accident, supply water for the auxiliary feedwater pumps will be manually transferred from the condensate storage tanks to the service water system, by the method described in the LOCA discussion [See Ref. SLB/FLB-6]. If the auxiliary feedwater system is not operating properly, the operator can initiate operation of the Standby AFW system (using Service Water suction) from the control room

11. If conditions and equipment availability permit, the operator can begin a gradual cooldown and depressurization to cold shutdown conditions. However, the primary safety function is to maintain the RCS in a safe condition at all times, removing decay heat at a rate comparable to the generation rate. Maintenance of this safe shutdown condition is accomplished by a combination of steam dump (to the condenser or atmosphere) with primary and secondary inventory makeup, accomplished by use of the safety injection and/or the charging pumps, and the auxiliary feedwater system. It is expected that RCS temperature can be lowered to near 212°F by using the steam generators. The safe shutdown conditions can be maintained until a final cooldown and depressurization to ambient conditions can be effected.

C. High Energy Line Breaks Outside Containment

An analysis has been provided describing the effects of pipe breaks outside containment [HELB-1]. The report proposed a program of augmented inservice inspection of certain piping welds in order to preclude the necessity to address further full diameter high energy piping breaks. Credible breaks of main steam lines outside containment, that is, those not included in the inspection program, are bounded by a 6 inch main steam line branch connection in the Intermediate Building and a 12 inch main steam line branch connection in the Turbine Building. The accident environment created by these breaks, and other postulated breaks are provided in References [HELB 8-11]. The program has been accepted by the NRC [Ref. HELB 7,8]. Several modifications have been performed at the Ginna Nuclear Plant as a result of high energy line break analyses. Reference [HELB-1] discusses the various modifications, but of particular note is the Standby Auxiliary Feedwater system modification. A "remote-manual" controlled standby auxiliary feedwater system, identical to the auxiliary feedwater system in cooling capability, has been installed. The pumps are housed in a seismically designed structure (area 6 Figure 3) remote from the auxiliary feedwater and any high energy lines. Any portion of this system required to operate in an emergency is not subjected to an adverse environment. Ref. [HELB-8] includes the

NRC's Safety Evaluation Report concerning the RG&E modifications resultant from the review of Ref. [HELB-1]. It includes a discussion of the acceptability of the instrumentation relocation and cable re-routing performed to insure that sufficient equipment will be protected from the environmental effects of a HELB outside containment. This portion of [HELB-8] is attached to this report.

The failure of steam heating lines in the Auxiliary Building was identified and discussed in Ref [HELB-1]. It has been determined that steam heating lines also traverse the diesel generator rooms and the screen house in the vicinity of safety related equipment. Modifications are planned which will isolate the steam heating line to the affected area in the event of a failure and therefore preclude an adverse environment. Prior to its installation, regular inspections are being performed to reduce the likelihood of a failure creating an adverse environment. These inspections, performed during each plant operating shift, would detect any leakage. Plant procedures call for isolation of the affected piping promptly upon detection of the leakage. In addition, confidence of the low likelihood of an adverse environment in the vicinity of safety related equipment is provided by the fact that, in almost 11 years of plant operation, no adverse environment has been created by any mechanism, including failure of the steam heating line.

D. Flooding Outside Containment

The potential for and protection for submergence of equipment due to postulated failures in the circulating water system is discussed in References [Flood 6-10].

III. IDENTIFICATION OF THE LIMITING SERVICE ENVIRONMENTAL CONDITIONS FOR EQUIPMENT WHICH IS REQUIRED TO FUNCTION TO MITIGATE THE CONSEQUENCES OF EVENTS IDENTIFIED ABOVE

A. Inside Containment

Post accident containment environmental conditions are discussed in Appendix 6E of the Ginna FSAR. These conditions result from a loss of coolant accident. The temperature and pressure profiles are given in Figure 1 of Appendix 6E with peak values being 286°F and 60 psig respectively. The radiation profile is presented in Figures 4 and 5 of Appendix 6E and it is seen, for example, that the doses at 30 minutes and one year following a LOCA are 1.7×10^6 R and 1.6×10^8 R, respectively. Materials compatibility with post-accident chemical environment is discussed in detail in Appendix 6E. 100% humidity is assumed.

Design parameters for environmental conditions have been conservatively selected for Ginna. As seen in FSAR Figure 14.3.4-2, the calculated peak pressure is less than 53 psig while the design value is 60 psig. The duration of the peak, similarly, bounds the calculated values.

Another example of the conservatism employed is seen in the accident radiation environment used for design purposes. As noted in Ref (WCAP 7744), a release of 100% of the noble gases, 50% of the halogens, and 1% of all remaining fission products is assumed. In addition,

no credit is taken for removal of radioactivity from the containment atmosphere by sprays, filters and fission product plateout. Finally, the specific activity in containment was roughly doubled by assuming a containment free volume associated with an ice condenser containment. Thus the radiation environment clearly overstates that which would be present even in a minimum safeguards case.

Submergence of valves has previously been discussed in Reference [Flood-4] and it has been shown that operation following submergence is not required. Submergence of instrumentation has been discussed in Ref [Flood-5] and in further discussed in Sections IV.G and IV.H of this report. It is shown that operation following submergences is not required. Therefore, no qualification for submerged service is required.

The peak pressure following a MSLB is given in Section 14.2.5 of the FSAR as 52 psig, assuming no credit for containment pressure reducing equipment. Recent analyses for other facilities indicate that the containment vapor temperature following a MSLB in containment may briefly exceed those derived for a LOCA. These higher temperatures should not be limiting, however, for qualification of equipment required following a MSLB, because of the nature of the transient, that is, the fact that the high temperature transient is very brief and there is superheated steam as opposed to saturated

steam; the location of equipment relative to the steam lines; and equipment configuration, that is, thermal lag. For these reasons, the humidity and steam environment following a LOCA remains limiting. This is consistent with the NRC's position 4.2 of the "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors." Radiation levels in containment following a MSLB are not limiting since fuel failures are not projected to result from a MSLB. Chemical environment and submergence are bounded by the LOCA conditions.

B. Auxiliary Building

Post accident environmental conditions in the Auxiliary Building are limited to normal ambient levels. (see also Section J below for a discussion of loss of air-conditioning"). In this regard, measures designed to prevent flooding of the residual heat removal pumps are addressed in Section 9.3.3 of the Ginna FSAR.

Radiation environment is presently under review as a result of the TMI Lessons Learned Commitments [see Ref. TMI-1].

C. Intermediate Building

Implementation of an augmented inservice inspection program for high energy piping outside containment has reduced the probability of pipe breaks in these systems to acceptable low levels [Ref. HELB-7, 8]. A six inch

main steam line branch connection is the Intermediate Building DBE. The limiting pressure is established in Ref. HELB-1 as being a pressure of 0.80 psig. Assuming saturation conditions, one obtains a limiting temperature of approximately 215°F. A 100% humidity steam-air mixture is assumed. Radiation, and chemical spray, the same as for the Auxiliary Building, do not require qualification. The effects of submergence need not be considered, as described in References [HELB-1] and [HELB-4].

Radiation environment is presently under review as a result of the TMI Lessons Learned Commitments [see Ref. TMI-1].

D. Cable Tunnel

Since the cable tunnel is open to the Intermediate Building, the limiting environmental conditions for the cable tunnel are identical to the Intermediate Building conditions.

E. Control Building

The limiting environment of the Control Building is normal ambient conditions. (See also Section J below for a discussion of "loss of air-conditioning") Protection against events which could occur outside the Control Building and affect the Control Building environment (see Ref. HELB-1) are identified and discussed in References HELB-1, HELB-6, HELB-7, FLOOD-1, and FLOOD-5.

F. Diesel Generator Rooms

The limiting environment in the diesel generator rooms is normal ambient conditions. Protection against failure of steam heating lines in the rooms is described in Section II C above. Protection against events outside the rooms is described in References HELB-1, HELB-6, HELB-7, FLOOD-1 and Flood-5.

G. Turbine Building

A pressure of 1.14 psig on the mezzanine and basement floors and 0.7 psig on the operating floor, with saturated steam conditions, is the limiting environment of the Turbine Building.

H. Auxiliary Building Annex

This structure, which houses the Standby Auxiliary Feedwater System, is described in References HELB-1 and HELB-6. The limiting environment in this structure is normal ambient conditions. The cooling system for this building is redundant and seismically qualified.

I. Screen House

The limiting environment in the Screen House is normal ambient conditions. Protection against flooding is described in References FLOOD-1 and FLOOD-5. (Also see Section J below for a discusssion of "loss of air-conditioning")

J. Loss of Air-Conditioning

The Ginna plant does not include a redundant safety related air-conditioning system for any area except the containment and the newly-installed Auxiliary Building annex. This was discussed during the NRC's SEP Site Hazards site vsit of September, 1978. In the event of a loss of air-conditioning, most areas of the plant would not be subject to significant temperature buildup, since these areas (auxiliary building, intermediate building, and screenhouse), are large volume buildings. The diesel generator rooms can be brought to essentially ambient temperature by opening the access doors at the north end of each room. This provides an unlimited supply of outside air. These spaces do not require continuous manning during an emergency and therefore the potential for radiological exposure of personnel in this mode of operation is very small.

The control room air handling unit is powered from a single Class 1E motor control center (MCC 1K). If there is a failure of this train (MCC 1C which is fed by the 1A diesel) during the post accident period, it is possible to crosstie to the 1B diesel. The operator, after assuring that any faults are cleared, closes the bus tie between busses 14 and 16 to energize the inoperative Control Room air handling unit from the 1B diesel, while making sure that the operational diesel does not become overloaded.

IV. EQUIPMENT QUALIFICATION INFORMATION

Table 3 summarizes the qualification of electrical equipment. Information contained in this section augments the qualification information of Table 3.

A. Auxiliary Feedwater Pumps

The auxiliary feedwater pumps are located in the basement of the intermediate building (area 2). The environment qualification of these pumps is standard industrial level. The consequences of an unlikely failure of one or both of these pumps is acceptable as this system has a redundant system (the Standby AFW System) which is not exposed to an adverse environment (Section III C).

B. Valves 878A, B, C and D

Valves 878 A, B, C and D are located in the containment basement and are Limitorque valves with Peerless motors. During normal operation these valves are positioned in their safeguards position with AC power removed so as to preclude failure in the event of an accident. Therefore, exposure to an adverse environment is of no consequence to valve performance.

C. Main Steam Isolation Valves

The main steam isolation valves are located in the intermediate building (area 3).

As discussed in Section II.B.3 above, the steam line non-return check valves provide protection redundant to

the main steam isolation valves for breaks upstream of the check valves. Due to equipment configuration, the check valves provide protection for all branch line breaks in the Intermediate Building and for all postulated crack breaks in the Intermediate Building except for cracks in the few feet of pipe between the check valves and the Intermediate Building and Turbine Building wall. Based on vendor data, it is expected that the valves will perform their function up to a temperature of 250°F, since the solenoid valves are enclosed in a NEMA-2 drip-proof enclosure.

D. Main Feedwater and Feedwater Bypass Isolation Valves

The main feedwater and feedwater bypass isolation valves are located in the turbine building at the intermediate floor level. These valves close upon receiving the safety injection signal for a main steam line break or loss of coolant accident and fail closed with a loss of air or power.

E. Containment Fan Cooler Dampers

The containment fan cooler dampers are located in the basement of the containment building and will be subjected to the environment of the main steam line break (inside containment) and the loss of coolant accidents. These valves do fail in the "fail-safe" accident position [FSAR - Section 6.3].

F. Barton 332 Transmitter

The Barton 332 transmitters are used to detect steam flow for each steam generator and generate a portion of the signal for steam line isolation. These transmitters are located on the operating floor of the containment and would be exposed to the adverse environment produced by a main steam line break (inside containment).

The Barton 332 transmitters are housed in a Nema 4 enclosure for protection against water. For temperature qualification an ovendry bake test at 320°F for 66 hours has been successfully performed.

These transmitters do not need to function for any steam line break where they would be exposed to an elevated environment. For breaks inside containment, the non-return check valves which are provided in each steam line will assure that the intact steam generator is properly isolated. A signal to the main steam isolation valves is also provided by high containment pressure. For a break outside containment, the nonreturn check valves will provide isolation if the break is upstream of the header. The steam flow transmitters, which will be unaffected by the break, will provide for main steam isolation valve closure.

G. Foxboro 611 Transmitters

Pressurizer pressure measurements are determined by the Foxboro 611 GM-DSI transmitter. As discussed previously, two out of three low pressurizer pressure signals initiate safety injection for the DBE's of concern.

These Foxboro 611 transmitters have successfully undergone pressure, temperature, and humidity tests as described in WCAP 7410-L and WCAP 7354-L for the LOCA DBE. Radiation exposure testing and knowledge of material properties indicates that the transmitter may experience failure at exposures of greater than 3×10^4 rads. LOCA analyses, References [LOCA 2-4] demonstrate that fuel failures in a LOCA do not occur until well after the pressurizer pressure has decreased below the SI initiation set point. No fuel failures are predicted in a MSLB. Thus, it can be concluded that the safety injection initiation function of the Foxboro 611 transmitters will not be affected by irradiation.

For post-accident monitoring, these transmitters would be used for operator indication in the event of a small LOCA or a secondary side break. If unavailable due to radiation or flooding failure, however, backup instrumentation is available (SI flow). Furthermore, accident mitigation emergency procedures dictate that, with no indication of stable or increasing pressurizer pressure

available, SI flow would be maintained to the reactor to ensure continued core cooling. This Foxboro 611 transmitter is also used to measure main steam pressure. These transmitters are located in the intermediate building, and are thus not subject to a post-accident containment environment. For breaks in the intermediate building, other instrumentation located inside containment, such as steam generator level, could be used to perform the required accident functions.

H. Foxboro 613 Transmitters

The Foxboro 613M-MDL modified transmitters are used to measure pressurizer level. These transmitters have been tested for LOCA environmental qualification conditions, as described in WCAP 7410-L and WCAP 7354-L.

Tests by Foxboro at 318°F and 90 psig demonstrate that the transmitter will perform its intended safety function. One test by Westinghouse indicates the transmitter had a 7.8% high error after 10 seconds.

A second test showed the instrument to perform satisfactorily with errors throughout the test at less than +4% and -5%.

Radiation capabilities of the 613 transmitters are expected to be similar to those at the Foxboro 611 transmitters; however, as with the Foxboro 611, qualification is not required. Pressurizer level is no longer used to generate a SI signal (This change is described in

Amendment No. 27 to the Ginna POL, June 15, 1979). Its primary purpose is to provide post-accident information to the operator. However, in the event of a LOCA or loss of secondary coolant, when an adverse radiation or flooding environment could damage this instrumentation, emergency procedures provide for the continued addition of safety injection flow if there is no indication of returning pressurizer level. This ensures continued adequate core cooling.

The Foxboro 613 HM-HSI transmitter is used to measure steam generator level. Although this transmitter is not qualified to withstand the post-accident containment environment, instrumentation to determine heat removal via the steam generators is available. The main steam pressure transmitters, located outside containment, gives indication of the steam generator status. Auxiliary feedwater flow instrumentation, also outside the containment, provides indication of sufficient flow to each steam generator.

I. Instrumentation Terminal Blocks

Within the containment building, the only safety related instrumentation circuits of concern that utilize terminal blocks are those associated with the pressurizer level and pressure transmitters. The transmitters and associated terminal blocks are enclosed in instrument cabinets located at the basement level of the containment.

The terminal blocks are Westinghouse 542247 and have the required documentation in support of environmental qualification for use in a DBE environment (see letter from L. D. White, Jr. to B. Grier, USNRC, dated February 10, 1978).

Inside containment, all other safety related instruments of concern have terminations within the transmitter enclosure and therefore, will not be exposed to a DBE environment.

J. Cable

The safety related power, control and instrumentation cable in the containment have been successfully tested for the DBE's. The qualification tests and results are described in WCAP 7410-L, Volume I.

All safety related power cable outside of containment is Kerite HT-FR, the same power cable qualified for, and used, in containment.

Control cable used outside containment was supplied by General Cable Company, rated 600 volt, PVC insulated, glass braid covering each conductor, 3 mil nylon jacket with overall black PVC jacket [HELB-1, Section 5].

The PVC jacket may soften around 212°F, however the glass braid over the insulation will retain the conductor insulation and prevent separation from the conductor.

The highest credible temperature that this type cable could be exposed to would be a 220°F temperature associated with a DBE in the turbine building. Thus, a failure of this type control cable is unlikely.

Instrumentation cable outside containment for all low level analog signals was supplied by Rome Cable Company, with "Synthenol" insulation covered with glass braid with an overall synthenol jacket. Test reports verify that this cable was heat aged at a temperature of 248°F for 168 hours and the jacket heat aged at 212°F for 120 hours. This cable is not required for safe shutdown of the plant during a high energy line break in the Intermediate Building. The safety related instrumentation cable used inside containment was supplied by Coleman Cable and Wire Company. The insulation is silicone rubber with glass jacketing. Silicone rubber insulated cable was qualified for LOCA conditions as described in WCAP 7410-L.

K. Reactor Coolant System Temperature Detectors

The reactor coolant system temperature detectors (RTD) are not required for a loss of coolant accident. In a steam line break accident, low Tave plus high steam flow plus a safety injection signal will close the main steam line isolation valves. Also, high-high steam flow will perform this function. As described in Section IV.C above, for a break upstream of the non-

return check valves, which includes all breaks inside containment, closure of the main steam isolation valves is not required.

For breaks downstream of the check valves, closure of the main steam isolation valves is desirable, however, in this case the RTDs are not subjected to an adverse environment. Therefore, the RTDs do not require environmental qualification to provide their required safety function.

The RTDs are also used as inputs to the subcooling meter. Additional backup to this function is available by use of the in-core thermocouples. The subcooling meter instrumentation design and qualification parameters are presently under review for installation within the TMI Short Term Lessons Learned implementation schedule [Ref. TMI-1].

L. Safety Related Cable Splices Subject to LOCA and MSLB Effects

Cable for the safety related pressurizer instrumentation, the core deluge valves, MOV 852 A and B, and the 480V power cable for the fan coolers utilizes Raychem Thermofit, WCSF-N, heat shrink sleeves on the splices subject to LOCA and MSLB effects. These sleeves have been qualified in tests which exceed the worst case Ginna accident environments. Refer to Franklin Institute Research Laboratories Test Report F-C4033-3. This report has been submitted to the NRC by Raychem Corporation.

In order to provide further assurance that the splice sleeves used for fan cooler 480V power cables are capable of operating in the accident environment, LOCA/ MSLB tests were performed on a mock-up of the actual splices in the Ginna containment. This test also included in line splice samples to further verify compatibility of the Raychem splice sleeve materials with the existing cable materials at Ginna. These tests are documented in Franklin Research Center Final Report F-C5074 dated April 1979.

M. Aging of Equipment Prior to Qualification Testing

Electrical equipment in general consists of components and materials with widely diverse physical properties. When establishing the design lifetime for such equipment, a program for accelerated aging is appropriate only when there is sufficient empirical data or a well understood aging mechanism upon which to base a quantitative estimate. Such bases exist for thermally accelerated aging of motor and cable insulation. When electronic components are involved, such as in transmitters, no such bases exist and accelerated aging is not appropriate. This type of equipment normally exhibits a failure rate of the "bath-tub curve" type. That is a period of relatively high failure rate early in life, "infant mortality", followed by a long period of low, constant, random failure rate, which finally terminates at what might be called the "end

of design life", characterized by a very rapidly increasing failure rate.

The objective of aging is to put samples in a condition equivalent to the end-of-life condition. When the equipment or component is of the first type described above, this objective is met by a period of thermal aging, which, based on the data available, can be shown to be equivalent to the design lifetime under normal operation conditions. Qualification tests for motor and cable insulation used in the Ginna Nuclear Plant were conducted using this approach as described in references.

For equipment of the second type there is a large amount of evidence to indicate that after an initial "burn in", sufficient to eliminate components which will be subject to "infant mortality", the failure mechanisms are truly random. That is, the probability of failure, per unit time, is low and constant, and most importantly there is no particular predominant failure mechanism.

Under these conditions there is no coupling between the design basis event for which the equipment is to be qualified and the age of the equipment. For such equipment, the "end-of-life condition" is identical with the condition immediately after "burn in". At the Ginna Nuclear Plant the period of operation of such

equipment has been long enough to assure that "burn in" has been accomplished. Periodic testing and maintenance assures that any equipment degradation is detected and thoroughly investigated for corrective action.

All failures of safety related equipment at Ginna Station are reported, documented, and reviewed for appropriate action (LER, Engineering review, etc.). Component failures are documented and logged under the NPRD data collection program in accordance with its standard format under the direction of the Ginna Technical Assistant, Operation Assessment Engineer.

Since the safety related systems and components generally exhibit very low failure rates, the most sensitve measure of degradation is direct observation by test and maintenance personnel, rather than quantitative analysis of failure rates. Accordingly, the plant procedures which describe the job functions of th Electrical and I & C foremen, and the Tests and Results Department personnel, specifically call attention to the necessity to evaluate any unusual <u>changes</u> in equipment performance or failure rate.

V. CONCLUSIONS

It has been determined that the facility will adequately respond to the design basis events as presented in this report and is acceptable for continued operation.

LOSS OF COOLANT ACCIDENT

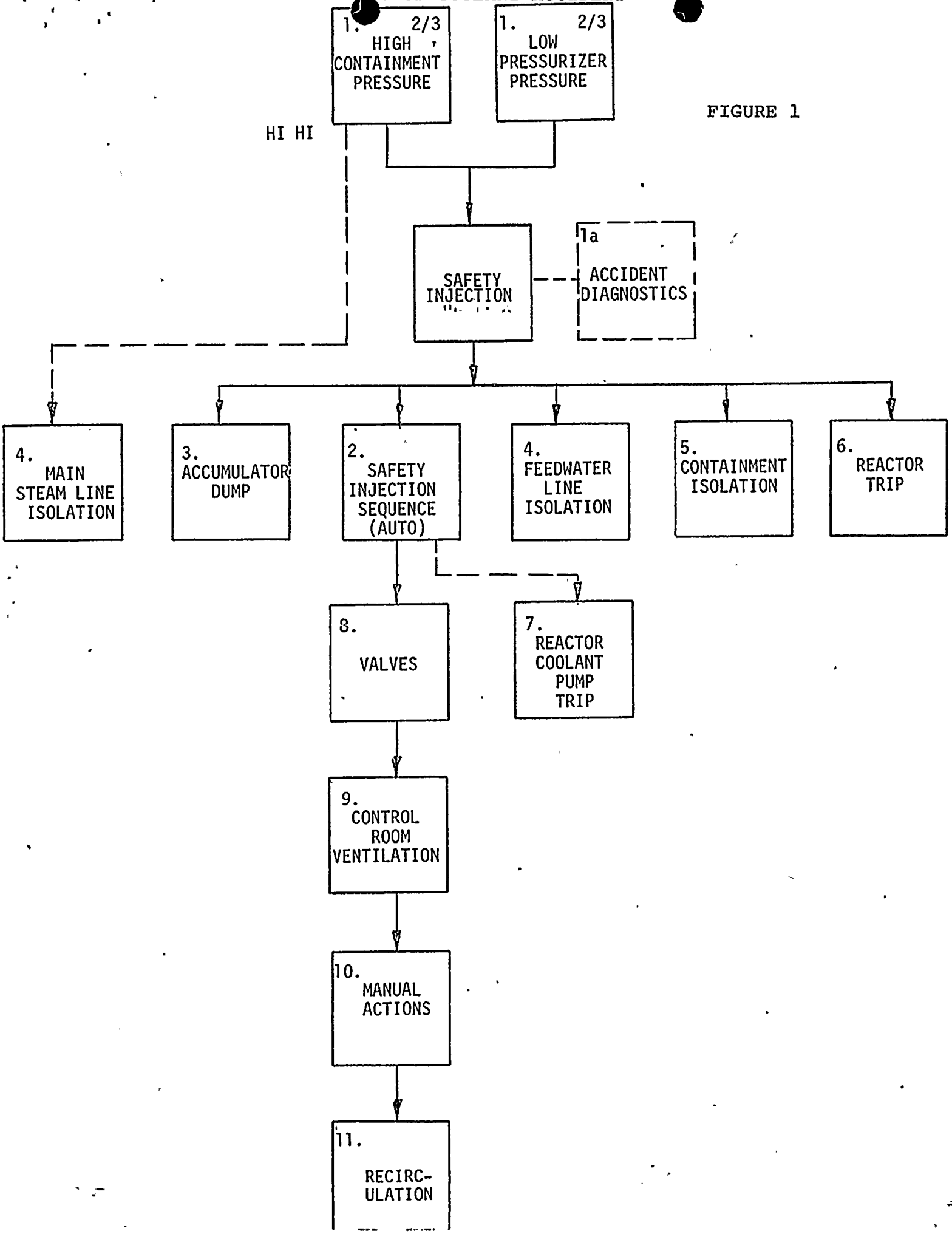

FIGURE 1

TABLE 1

LOSS OF COOLANT ACCIDENT

BLOCK NO./EQUIPMENT	SAFETY FUNCTION	REQUIRED OPERATION TIME
1. High Containment Pressure Low Pressurizer Pressure		
PT 945, 946, 947 PT 948, 949, 950	Provide signals for Containment Spray, Safety Injection, Containment Isolation, and Main Steam Line Isolation	Signal Initiation
	Accident Diagnostics	Short Term
PT 429, 430, 431, 449	Provide Reactor trip and Safety Injection signals	Short Term Signal Initiation
	Accident Diagonstics	
1a. Steam Line Pressure PT 468, 469, 482 PT 478, 479, 483	Accident Diagnostics	Short term
Containment Radiation [Being provided per TMI STLL]	Accident Diagnostics	Short term
Containment sump level LT 942, LT 943	Accident Diagnostics	Short term
2. Safety Injection Sequence (Auto)		
1A, 1B Diesel Generator and Auxiliaries	Power supply to safeguards bussess during loss of outside AC Power	Long term
480 Volt Safeguards busses 14, 16, 17, 18	Provide the distribution of power to safeguards equipment	Long term
1A, 1B, 1C Safety Injection Pumps	High head injection of borated water to Reactor Coolant System	Long term
1A, 1B Containment Spray Pumps (only on hi-hi Cont. pressure)	Containment Pressure, Temperature, and Iodine control	Long term
1A, 1B Residual Heat Removal Pumps	Low head injection of borated water to Reactor Vessel	Long term
1A, 1B, 1C, 1D Service Water Pumps	Cooling water to RHR and CCW Heat Exchangers	Long term
1A, 1B, 1C, 1D Containment Recirc. Units	Containment Pressure, Temperature, and Iodine control	Long term

TABLE 1 I[illegible]S OF COOLANT ACCIDENT

BLOCK NO./EQUIPMENT	SAFETY FUNCTION	REQUIRED OPERATION TIME
1A, 1B Motor Driven Aux. Feedwater Pumps	Cooling water to Steam Generators	Long term
480 Volt Safeguards MCC's 1C, 1D	Provide the distribution of power to safeguards equipment	Long term
3. Accumulator Dump		
MOV 841 (N.O.)* MOV 865 (N.O.)	Provide path to Reactor Vessel from Accumulators for injection of borated water	Not required to function
4. Main Steam Line Isolation Feedwater Line Isolation		
AOV 3516 AOV 3517	Isolate 1A, 1B Steam Generators	5 Seconds after signal
AOV 4269 AOV 4270 AOV 4271 AOV 4272	Isolate Main Feedwater System	5 Seconds after signal
5. Containment Isolation	See Text, Section II.A.5	
6. Reactor Trip		
Reactor trip breakers	Provide means to trip the reactor	Required for Reactor Trip
Reactor protection and instrumentation cabinets	Provide the instrumentation and protection circuits for the control and tripping of the Reactor	Required for Reactor Trip
7. RCP Trip RCP Trip Breakers	Provide means to trip RCP's	Short term
8. Valves		
MOV 825 A,B MOV 826 A,B,C,D (B&D N.O.)	Provide path to SI Pumps for borated water to high head safety injection	10% BAST Level or~1/2 hour
AOV 836 A,B	Provide controlled addition of NaOH to Containment Spray for Iodine control	Short term
MOV 852 A,B	Provide path to Reactor Vessel of borated water for low head safety injection	SI initiation
MOV 860 A,B,C,D	Provide path to Containment Spray headers for CS Pumps	Long term

*N.O. = Normally Open

TABLE 1

ITEMS OF COOLANT ACCIDENT

BLOCK NO./EQUIPMENT	SAFETY FUNCTION	REQUIRED OPERATION TIME
BAST Level LT 102, 106, 171, 172 BAST to RWST	Indicate BAST Level for automatic transfer of SI Pump suction from	10% BAST Level or~1/2 hour
MOV 878 B,D (N.O.)	Provide path to cold legs of RCS from high head safety injection	not required to function
MOV 4007, 4008 1A, 1B Steam Generators	Provide path for Aux. Feedwater to	Short term
AOV 5871, 5872, 5873 AOV 5874, 5875, 5876	Provide path for cleaning of cont. atmosphere by fan coolers	signal initiation
9. Control Room Ventilation Dampers Fans	Provide cleaning of Control Room atmosphere	Short term
10. Manual		
Safety Injection Reset Button	Reset Safety Injection signal after automatic S.I. Sequencing is complete	less than 24 hours
1A, 1B Component Cooling Water Pumps	Cooling water for safeguards equipment	Long term
1A, 1B Containment Spray Pumps (if Cont. Pressure <30 psig)	Containment Pressure, Temperature and Iodine control	Long term
RWST Level LT 920, LIC 921	Indicate RWST Level for operator transfer from S.I. phase to Recirculation phase	less than 24 hours
MOV 4027, 4028	Provide Service Water to Motor Driven Aux. Feedwater Pumps suction	within~2 hours
MOV 4734, 4735, 4615, 4616 MOV 738 A,B	Direct SW Flow to CCW HX's Direct CCW Flow to RHR HX's	less than 24 hours less than 24 hours
Standby AFW Pumps	AFW Flow to SG's if normal AFW System inoperable	Long term
MOV 9629 A,B	Provide SW to suction of standby AFW Pumps	Long term
MOV 9710 A,B; 9703 A,B; 9704 A,B	Standby AFW Discharge Valves to provide flow to SG's	Long term

BLOCK NO./EQUIPMENT	SAFETY FUNCTION	REQUIRED OPERATION TIME
11. Recirculation		
MOV 850 A,B outside cont. MOV 851 A,B (N.O.) inside cont.	Provide path to RHR suction from B sump for low head safety injection	long term
MOV 856 (N.O.)	RWST isolation valve to RHR pumps suction, must close after RWST is drained	required to function to switch to recirc phase
MOV 896 A,B 9 (N.O.)	RWST isolation valve, must close after RWST is drained	required to function to switch to recirc phase
MOV 857 A,B,C	Provide path to suction of SI and CS Pumps from RHR pumps discharge	required to function to switch to recirc phase
AOV 897, 898	Isolate high head recirculation flow to RWST during sump recirculation	Short term
MOV 704 A,B	Close during switch to sump recirculation	less than 24 hours

MAIN STEAM OR FEED LINE BREAK

FIGURE 2

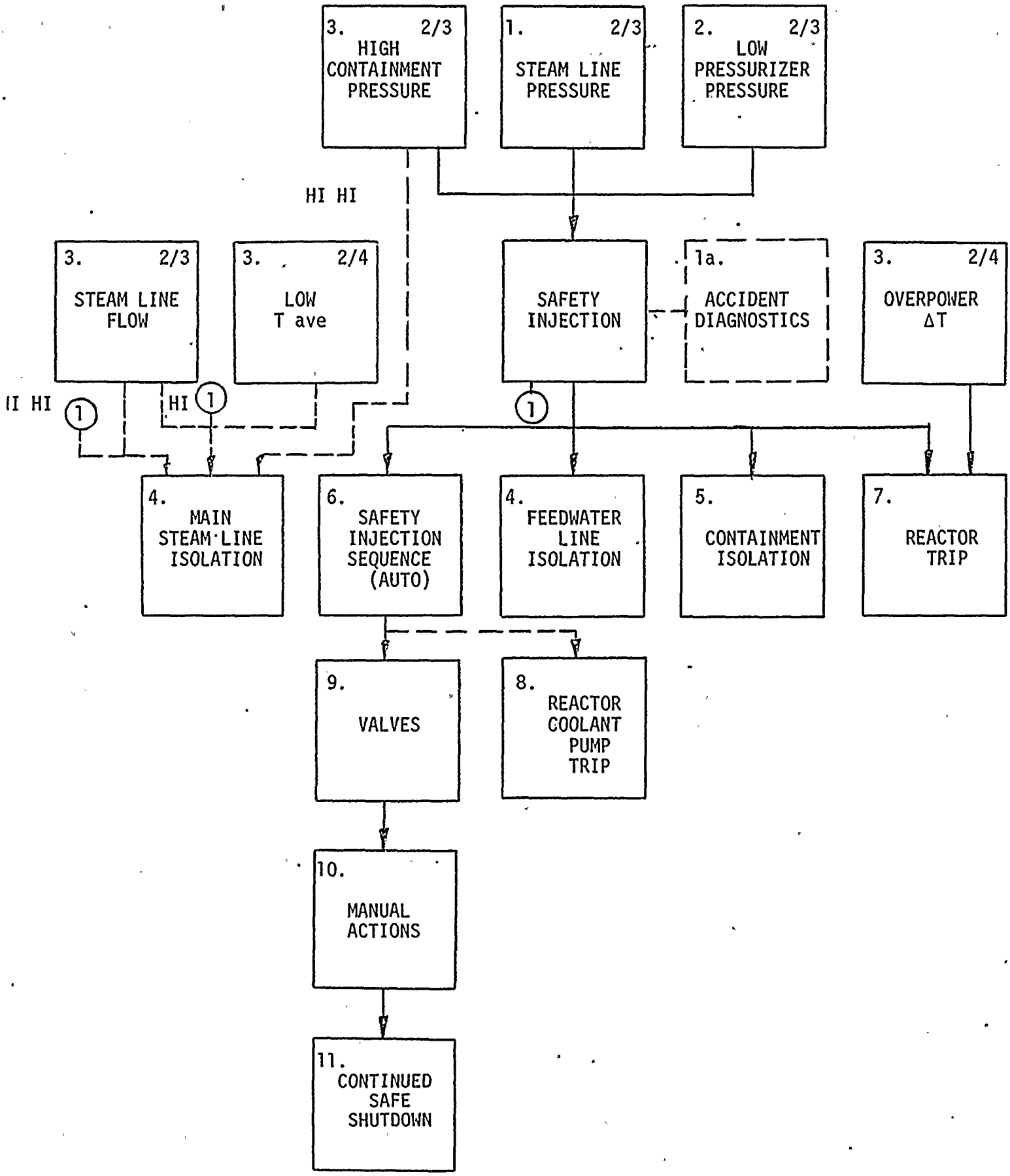

TABLE 2

MAIN STEAM LINE BREAK

BLOCK NO./EQUIPMENT	SAFETY FUNCTION/BREAK LOCATION INSIDE CV	SAFETY FUNCTION/BREAK LOCATION OUTSIDE CV	REQUIRED OPERATION TIME
1. Steam Line Pressure PT 468, 469, 482 PT 478, 479, 483	Provide signal for SI on low steam line pressure	same	signal initiatic
1a. Steam Line Pressure (see 1 above)	Accident Diagnostics	same	short term
Containment Radiation	Accident Diagnostics	NA	short term
Containment Sump Level	Accident Diagnostics	NA	short term
High Containment Pressure (see 3 below)	Accident Diagnostics	NA	short term
2. Low Pressurizer Pressure			
PT 429, 430, 431, 449	Provide Reactor trip and Safety Injection signals	same	signal initiation
3. High Containment Pressure			
PT 945, 946, 947 PT 948, 949, 950	Provide signals for Containment Spray, Safety Injection, Containment Isolation, and Main Steam Line Isolation	NA	signal initiation
Steam Line Flow			
FT 464, 465 FT 474, 475	Provide signals for Reactor trip and Main Steam Line Isolation	same	signal initiation
Reactor Coolant Temperature			
Loop A Hot Leg TE 401A, 402A, 405A, 406A, 409A	Provide Low Tave & ΔT signals for Reactor trip, Safety Injection and Main Steam Line Isolation	same	signal initiation
Loop A Cold Leg TE 401B, 402B, 405B, 406B, 409B			
Loop B Hot Leg TE 403A, 404A, 407A, 408A, 410A			
Loop B Cold Leg TE 403B, 404B, 407B, 408B, 410B			

TABLE 2

MAIN STEAM LINE BREAK

BLOCK NO./EQUIPMENT	SAFETY FUNCTION/BREAK LOCATION INSIDE CV	OUTSIDE CV	REQUIRED OPERATION TIME
4. Main Steam Line Isolation			
AOV 3516 AOV 3517	Isolate 1A, B Steam Generators	same	5 seconds after signal
Feedwater Line Isolation			
AOV 4269 AOV 4270 AOV 4271 AOV 4272	Isolate Main Feedwater system	same	5 seconds after signal
5. Containment Isolation	See Text, Section II.B.5	same	
6. Safety Injection Sequence (Auto)			
1A, 1B Diesel Generators and auxiliaries	Power supply to safeguards busses during loss of outside AC Power	same	Long term
480 Volt Safeguards busses 14, 16, 17, 18	Provide distribution of power to safeguards equipment	same	Long term
1A, 1B, 1C Safety Injection Pumps	High head injection of borated water to Reactor Coolant System	same	Long term
1A, B Containment Spray Pumps (only on hi-hi cont. Pressure)	Containment Pressure and Temperature control	N/A	Long term
1A, 1B, 1C, 1D Service Water Pumps	Cooling Water to CCW Heat Exchanger	same	Long term
1A, 1B, 1C, 1D Containment Recirc Units	Containment Pressure and Temperature control	N/A	Long term
1A, 1B Motor Driven Aux. Feedwater Pumps	Cooling water supply to Steam Generators	same	Long term

BLOCK NO./EQUIPMENT	SAFETY FUNCTION/BREAK LOCATION INSIDE CV	OUTSIDE CV	REQUIRED OPERATION TIME
480 Volt Safeguards MCC's 1C, 1D	Provide the distribution of power to safeguards equipment	same	Long term
7. Reactor Trip			
Reactor trip breakers	Provide means to trip the reactor	same	Required for Reactor Trip
Reactor Protection and Instrumentation Cabinets	Provide the instrumentation and protection circuits for the control and tripping of the reactor	same	Required for Reactor Trip
8. Reactor Coolant Pump Trip RCP Trip Breakers	Provide means to trip RCP's	same	Short term
9. Valves			
MOV 825A, B MOV 826A, B, C, D (B&D N.O.)	Provide path to SI Pumps for borated water to high head safety injection	same	10% BAST Level or~1/2 hour
AOV 836A, B	Provide NaOH to CS if needed		Short term
MOV 860A, B, C, D	Provide path to Containment Spray headers for CS Pumps	N/A	Long term
MOV 878, B, D (N.O.)	Provide path to cold legs of RCS from high head safety injection	same	not required to function
MOV 896, A, B (N.O.)	Provide path from RWST of borated water for SI and CS pumps suction	same	short-term (to close if need sump recirculation)
MOV 4007, 4008	Provide path for Aux. Feedwater to Steam Generators	same	Short term
AOV 5871, 5872, 5873 AOV 5874, 5875, 5876	Provide path for cleaning by fan coolers, cooling of cont. Atmosphere	N/A	signal initiation

BLOCK NO./EQUIPMENT	SAFETY FUNCTION/BREAK LOCATION INSIDE CV	OUTSIDE CV	REQUIRED OPERATION TIME
BAST Level LT 102, 106, 171, 172	Indicate BAST Level for automatic transfer of SI Pump suction from BAST to RWST	same	10% BAST Level or~1/2 hour
MOV 852A, B	Provide path for low had SI to Reactor Vessel	same	Signal Initiation
10. Manual			
SG Level Instrumentation LT 470, 471, 472, 473 LT 460, 461, 462, 463	Determine affected SG	same	short term
Safety Injection Reset Button	Reset SI signal after Automatic SI sequencing is complete	same	less than 24 hours
1A, 1B Component Cooling Water Pumps	Cooling Water for safeguards equipment	same	Long term
1A, 1B Containment Spray Pump (If cont. Pressure < 30 psig)	Containment Pressure and Temperature control	N/A	Long term
MOV 4027, 4028	Provide Service Water to Motor Driven Aux. Feedwater Pumps Suction	same	within ~2 hours
Charging pumps	Inventory control to RCS	same	Long term
Standby AFW pumps	Provide AFW flow to SG's if AFW system inoperable	same	Long term
MOV 9629A, B	Provide SW to suction of Standby AFW Pumps	same	Long term
MOV 9710A, B; 9703A, B; 9704A, B	Standby AFW discharge valves to provide AFW flow to SG's	same	Long term
MOV 4000A, B	AFW Cross-Connect Valves	same	Short term

Accident References

LOCA analysis [LOCA]

1. FSAR

2. "ECCS Analysis for the R. E. Ginna Reactor with ENC WREM-2 PWR Evaluation Model" dated December 1977 submitted with Application for Amendement to Operating License, on January 6, 1978.

3. ECCS Analysis submitted by letter dated April 7, 1977 from L. D. White, Jr., RG&E to A. Schwencer, Chief, Operating Reactors Branch #1, USNRC.

4. ECCS Analysis for the R. E. Ginna Reactor with ENC WREM-2 PWR Evaluation Model. Exxon Nuclear Co. Report XN-NF-77-58.

5. Ginna Emergency Procedures E1.1 and E1.2, submitted by letter dated February 26, 1980 from L. D. White, Jr. RG&E, to D. L. Ziemann, USNRC.

Steam Line Break and Feedwater Line Break [SLB/FLB]

1. FSAR

2. Steam line break analyses submitted with Application for Amendment to Operating License on September 22, 1975.

3. Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant, Exxon Report XN-NF-77-40 (11/77 and updated 12/15/78 and March, 1980.

4. Letter dated May 24, 1977 from K. W. Amish, RG&E to J. F. O'Leary, NRC.

5. Ginna Emergency Procedures E1.1 and E1.3, submitted by letter dated February 26, 1980 from L. D. White, Jr., RG&E to D. L. Ziemann, USNRC.

6. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, March 28, 1980.

High Energy Line Break [HELB]

1. "Effects of Postulated Pipe Breaks Outside the Containment Building", GAI Report No. 1815, submitted by letter dated November 1, 1973 from K. W. Amish, RG&E, to A, Giambuso, Deputy Director for Reactor Projects, USNRC.

2. Letter dated May 24, 1974 from K. W. Amish, RG&E, to J. F. O'Leary, Director, Directorate of Licensing, USNRC.

3. Letter dated September 4, 1974 for R. R. Koprowski, RG&E to Edson Case, Acting Director, Directorate of Licensing, USNRC.

4. Letter dated November 1, 1974 from K. W. Amish, RG&E, to Edson Case, Acting Director, Directorate of Licensing, USNRC.

5. Letter dated May 20, 1977 from L. D. White, Jr., RG&E, to A. Schwencer, Chief Operating Reactors Branch #1, USNRC.

6. Letter dated February 6, 1978 from L. D. White, Jr., RG&E, to A. Schwencer, Chief, Operating Reactors Branch #1, USNRC.

7. Amendment No. 7 to Provisional Operating License DPR-18, transmitted, by letter dated May 14, 1975 from Robert A. Purple, Chief, Operating Reactors Branch #1, USNRC, to L. D. White, Jr., RG&E.

8. Amendment No. 29 to Provisional Operating License DPR-18, transmitted by letter dated August 24, 1979 from Dennis L. Ziemann, Chief, ORB #2, to L. D. White, Jr., RG&E.

9. Letter, L. D. White, Jr., RG&E, to D. L. Ziemann, May 17, 1979.

10. Letter, L. D. White, Jr., RG&E, to D. L. Ziemann, USNRC, June 27, 1979.

11. Letter, L. D. White, Jr., RG&E, to D. L. Ziemann, USNRC July 6, 1979.

Effects of Flooding [Flood]

1. Letter dated May 13, 1975 from L. D. White, Jr., RG&E, to Benard C. Rusche, Director, Office of Nuclear Reactor Regulation, USNRC.

2. Letter dated May 20, 1975 from L. D. White, Jr., RG&E, to Robert A. Purple, Chief, Operating Reactors Branch #1, Division of Reactor Licensing.

3. Letter dated May 30, 1975 from L. D. White, Jr., RG&E, to Robert A. Purple.

4. Letter dated June 16, 1975 from L. D. White, Jr., RG&E, to Robert A. Purple.

5. Letter dated July 3, 1975 from Robert A. Purple to L. D. White, Jr., RG&E.

6. Letter dated August 8, 1972 from Donald J. Skovholt, Assistant Director for Operating Reactors, USAEC, to Edward J. Nelson, RG&E.

7. Letter dated October 3, 1972 from K. W. Amish, RG&E, to Donald J. Skovholt, Assistant Director for Operating Reactors, USAEC.

8. Letter dated May 31, 1973 from K. W. Amish, RG&E, to Donald J. Skovholt, Assistant Director for Operating Reactors, USAEC.

9. Application for Amendment to Operating License, submitted March 10, 1975.

10. Amendment No. 14 to Provisional Operating License DPR-18, transmitted by letter dated June 1, 1977 from A. Schwencer, Chief, Operating Reactors Branch #1, USNRC.

TMI Lessons Learned [TMI]

1. RG&E letters of October 17, November 19, and December 28, 1979, L. D. White, Jr., RG&E, to D. L. Ziemann, USNRC, "TMI Short Term Lessons Learned Requirements."

Table 3

Reactor: GINNA — SYSTEMATIC EVALUATION PROGRAM

EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document Reference	COMMENTS
1.								
SOLENOID VALVE	OUTSIDE		Temp (°F)	220	130	VENDOR DATA	23	DBE - MAIN SLB
ASCO /			Pr (psia)	17	ATM	EXPERIENCE		IN TURBINE BLDG
V-4269, V-4270			RH (%)	100	AMB	EXPERIENCE		FAIL SAFE
LB 8300 B 61 U			Chem.	NO	—	—		
(FW CONTROL VALVES)			Rad.	NO	—	—		
V-4271, V-4272			Sub.	NO	—	—		
LB 8300 B 64 RU								
(FW BYPASS VALVES)								
LCV-112B		SHORT	Temp (°F)	AMB	AMB	EXPERIENCE		NO ACCIDENT
LBX 831616			Pr (psia)	ATM	ATM	EXPERIENCE		ENVIRONMENT IN
(CHARGING FROM RWST)			RH (%)	AMB	AMB	EXPERIENCE		AUXILIARY BLDG
2.								
SOLENOID VALVE	OUTSIDE	SHORT	Temp (°F)	AMB	150°	VENDOR DATA	24	NO ACCIDENT
COPES-VULCAN /			Pr (psia)	ATM	ATM	EXPERIENCE		ENVIRONMENT IN
D-100-60			RH (%)	AMB	AMB	EXPERIENCE		AUXILIARY BLDG
AOV 836 A,B			Chem.	NO	—			
(NaOH to CS)			Rad.	NO	—			
			Sub.	NO	—			
3.								
SOLENOID VALVE	OUTSIDE		Temp (°F)	215	250	VENDOR DATA	25	ENCLOSED IN NEMA-2
LAWRENCE /			Pr (psia)	16	ATM.	EXPERIENCE		DRIP PROOF ENCLOSURE
:110114W - SUPPLY			RH (%)	100	AMB	EXPERIENCE		WHICH IS SUBJECTED TO
125434W - VENT			Chem.	NO	—			SALT WATER SPRAY
V-3516, V-3517			Rad.	NO	—			QUALIFICATION T
(MAIN STEAM ISOLATION)			Sub.	NO	—			
4.								
SOLENOID VALVE	INSIDE	SECONDS	Temp (°F)	286	200	VENDOR DATA	26	FAIL-SAFE. PERFORMS
VERSA / VSG			Pr (psia)	75	ATM			SAFETY FUNCTION
V-5871, V-5872, V-5873			RH (%)	100	AMB			WITHIN SECONDS OF
V-5874, V-5875, V-5876			Chem.	YES	YES		27	START OF DBE. NOT
			Rad.	1.7×10^6	NO			REQUIRED TO
			Sub.	NO	—			OPERATE WHEN
								ACCIDENT CONDITIONS
								ARE REACHED.

Reactor: GINNA — SYSTEMATIC EVALUATION PROGRAM

EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document. Reference	COMMENTS
5.								
SOLENOID VALVE			Temp(°F)	AMB	130	VENDOR DATA	23	
ASCO/			Pr(psia)	ATM	ATM	EXPERIENCE		
HCV-624, HCV-625	OUTSIDE	LONG TERM	RH(%)	AMB	AMB	EXPERIENCE		N.O. NO NEED TO FUNCTION
(RHR DISCHARGE)			Chem.	NO	—			
AOV-897, AOV-898	OUTSIDE	<24 hours	Rad.	[TMI]	—			FAIL CLOSE ON LOSS
(SI RECIRCULATION)			Sub.	NO	—			OF AIR
6. SOLENOID VALVE			Temp(°F)	See Ref	See Ref	See Ref.	39,40,41,42	TEST PROGRAM
VERSA/			Pr(psia)	↓	↓	↓		PRESENTLY
VSG-3731	BOTH	SECONDS	RH(%)					UNDERWAY
(CONT. PURGE VALVES)			Chem.					
VSG-3421			Rad.					
(CONT. DEPRESSURIZATION)			Sub.					
7.								
CONTROL ROOM DAMPERS D-81 → D-87	OUTSIDE	SHORT TERM	Temp(°F)	AMB	AMB	EXPERIENCE		NORMAL ENVIRONMENT
			Pr(psia)	ATM	ATM	EXPERIENCE		
			RH(%)	AMB	AMB	EXPERIENCE		
			Chem.	NO	—	—		
			Rad.	NO	—	—		
			Sub.	NO	—	—		
8a.								
LIMITORQUE	INSIDE	—	Temp(°F)	286	320	TEST	18,19	VALVES ARE
SMB-2			Pr(psia)	75	105	TEST	18,19	LOCKED-OPEN
RELIANCE MOTOR			RH(%)	100	100	TEST	18,19	WITH POWER
MOV 841, 865			Chem.	YES	YES	EVALUATION	27	REMOVED. NO
(ACCUMULATOR DISCHARGE)			Rad.	1.6×10^8	2×10^8	TEST	18,19	NEED TO FUNCTION
			Sub.	YES	NO	—	37	
8b. LIMITORQUE	OUTSIDE							
SMB-00, PEERLESS			Temp(°F)	AMB	120		13	NOT EXPOSED TO
MOV 826 A,B,C,D		<30 min	Pr(psia)	ATM	ATM	EXPERIENCE		DBE ENVIRONMENT
(BAST TO SI PUMPS)			RH(%)	AMB	AMB	EXPERIENCE		
MOV 896 A,B		<24 hr	Chem.	NO	—	—		
(RWST TO SI PUMPS)			Rad.	[TMI]	—	—		
			Sub.	NO	—	—		

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document Reference	COMMENTS
8c								
LIMITORQUE	OUTSIDE	<30 min	Temp (°F)	AMB	320	TEST	18, 19	NO EXPOSED
SMB-00			Pr (psia)	ATM	105	TEST	18, 19	TO DBE
RELIANCE MOTOR			RH (%)	AMB	100	TEST	18, 19	ENVIRONMENT
MOV 825 A, B			Chem.	NO	—	—	—	
(RWST TO SI PUMPS)			Rad.	NO	—	—	—	
			Sub.	NO	—	—	—	
8d								
LIMITORQUE	OUTSIDE	SHORT	Temp (°F)	215	320	TEST	18, 19	DBE - TURBINE
SMB-00			Pr (psia)	16	105	TEST	18, 19	BUILDING STEAM
RELIANCE MOTOR			RH (%)	100	100	TEST	18, 19	LINE BREAK
MOV 4007, 4008			Chem.	NO	YES	EVALUATION	27	
(AFW DISCHARGE)			Rad.	NO	2×10^8	TEST	18, 19	
MOV 4027, 4028			Sub.	NO	—	—	—	
(AFW SUCTION)								
8e								
LIMITORQUE	OUTSIDE	LONG	Temp (°F)	AMB	320	TEST	18, 19	NOT EXPOSED TO
SMB-00			Pr (psia)	ATM	105	TEST	18, 19	DBE ENVIRONMENT
RELIANCE			RH (%)	AMB	100	TEST	18, 19	EXCEPT POST-
V-850A, B (SUMP VALVES)			Chem.	NO	YES	EVALUATION	27	LOCA SUMP WATER
MOV 856 (RWST TO RHR)			Rad.	[TMI]	2×10^8	TEST	18, 19	RECIRCULATION
V-857A, B, C (RHR TO SI)			Sub.	NO	—	—	—	
V-860A, B, C, D (CS VALVES)								
8f								
LIMITORQUE	INSIDE	LONG	Temp (°F)	286	120	VENDOR DATA	13	NOT REQUIRED TO
SMB-00			Pr (psia)	75	ATM	EXPERIENCE		FUNCTION FOR DBE.
			RH (%)	100	AMB	EXPERIENCE		VALVES ARE IN
MOV-851 A, B			Chem.	YES	NO	—		LOCKED-OPEN POSITION
			Rad.	1.7×10^6	NO	—		AS REQUIRED FOR SI
			Sub.	NO	NO	—		

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document Reference	COMMENTS
8g								
LIMITORQUE	INSIDE	—	Temp(°F)	286	120	VENDOR DATA	13	NOT REQUIRED TO
SMB-00			Pr(psia)	75	ATM			FUNCTION FOR DBE.
PEERLESS MOTOR			RH(%)	100	AMB			VALVES ARE LOCKED
MOV 878 B,D			Chem	YES	—			IN OPEN POSITION,
(SI TO COLD LEGS)			Rad.	1.6×10^8	—			AS NEEDED FOR SI.
			Sub.	YES	NO			
8h								
LIMITORQUE	INSIDE	SI SIGNAL	Temp(°F)	286	320	TEST	18,19	VALVE COMPLETES
SMB-1			Pr(psia)	75	105	TEST	18,19	SAFETY FUNCTION
RELIANCE MOTOR			RH(%)	100	100	TEST	18,19	(TO OPEN) EARLY
MOV 852 A,B			Chem.	YES	YES	EVALUATION	27	INTO ACCIDENT
(CORE DELUGE)			Rad.	1.6×10^8	2×10^8	TEST	18,19	
			Sub.	YES	NO	—	37	
8i								
LIMITORQUE	OUTSIDE	LONG TERM	Temp(°F)	120	120	VENDOR DATA	43	STANDBY AFW SYSTEM
SMB-00			Pr(psia)	ATM	ATM	EXPERIENCE		LOCATED IN CON-
RELIANCE MOTOR			RH(%)	AMB	AMB	EXPERIENCE		TROLLED ENVT. RAD.
MOV 4000 A,B			Chem.	NO	—	—		LEVELS FOR AUX.
(AFW CROSS-CONNECT)			Rad.	[TMI]	—	—		BLDG. AND INT. BLDG
MOV 9703 A,B; 9704A,B;			Sub.	NO	—	—		REVIEWED PER TMI
9710A, B								
(STANDBY AFW SYSTEM)								
9.								
MOTOR, PUMP	OUTSIDE	LONG TERM	Temp(°F)	120	120	EXPERIENCE	43	STANDBY AFW
GENERAL ELECTRIC			Pr(psia)	ATM	ATM	EXPERIENCE		PUMPS LOCATED IN
			RH(%)	80	80	EXPERIENCE		AUX BLDG ANNEX
			Chem.	NO	—	—		WHICH HAS CON-
			Rad.	NO.	—	—		TROLLED ENVT.
			Sub.	NO	—	—		

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document Reference	COMMENTS
10. MOTOR, PUMP			TEMP	AMB	104°F	SPEC	15,16	NOT EXPOSED TO DBE
WESTINGHOUSE	OUTSIDE	LONG	PR. (PSIA)	ATM	ATM			
444 TS TBDP			RH	AMB	AMB			
445 TS TBDP			CHEM.	No	—	—	—	
(CONTAINMENT SPRAY,			RAD.	No	—	—	—	
RHR, COMPONENT			SUB.	No	—	—	—	
COOLING)								
11. MOTOR PUMP	OUTSIDE	LONG	TEMP	220°F	104°F	SPEC	8,16	HAVE INSTALLED TOTALLY
WESTINGHOUSE			PR. (PSIA)	17	ATM	EXPERIENCE		REDUNDANT SYSTEM NOT
505 US ABDP			RH	100%	AMB	EXPERIENCE		EXPOSED TO DBE
			CHEM.	NO	—	—	—	
AFNP			RAD.	No	—	—	—	
			SUB	No				
12. MOTOR PUMP	OUTSIDE	LONG	TEMP	AMB	104°F	SPEC	15,16	NOT EXPOSED TO DBE
WESTINGHOUSE			PR. (PSIA)	ATM	ATM	EXPERIENCE		
509 US AFDP			RH	AMB	AMB	EXPERIENCE		
509 UPH ABDP			CHEM.	No	—	—	—	
SAFETY INJECTION			RAD	NO	—	—	—	
SERVICE WATER			SUB	NO	—	—	—	
13a. PENETRATION, ELEC	INSIDE	LONG	TEMP	286°F	320°F	TEST	1	
CROSS-HINDS			PR (PSIA)	75	105	TEST	1	
NO PART #			RH	100%	100%	TEST	1	
			CHEM	Yes	Yes	EVALUATION	27	
			RAD	1.6×10^8	1.17×10^8	TEST	28	
			SUB	NO	—	—	—	
13b. PENETRATION, ELEC.	INSIDE	LONG	TEMP	286°F	340°F	TEST	29, 30	
WESTINGHOUSE			PR (PSIA)	75	75	TEST	29, 30	
NO PART #			RH	100%	100%	TEST	29, 30	
			CHEM	Yes	Yes	EVALUATION	29, 30	
			RAD	1.6×10^8	2.1×10^8	TEST	29, 30	
			SUB	NO	—	—	—	

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document Reference	COMMENTS
14. TERMINAL BLOCK	INSIDE	LONG	TEMP.	286°F	340°F	TEST	22	
WESTINGHOUSE			PR(PSIA)	75	121	TEST	22	
542247			RH	100%	100%	TEST	22	
			CHEM	Yes	YES	EVALUATION	27	
			RAD	1.6×10^8	2×10^8	TEST	22	
			SUB	NO	—	—	—	
15a. CABLE	INSIDE	LONG	TEMP	286°F	318°F	TEST	18,19	
KERITE			PR (PSIA)	75	121	TEST	18,19	
HT			RH	100%	100%	EVALUATION	11	
			CHEM	Yes	Yes	EVALUATION	27	
			RAD	1.6×10^8	2×10^5	TEST	18,19	
15b			SUB	NO	—	—	—	
CABLE	INSIDE	LONG	TEMP	286	318	TEST	18,19	
KERITE			Pr	75	121	TEST	18,19	
HT			RH	100	100	EVAL	11	
			Chem	YES	YES	EVAL	27	
			Rad	1.6×10^8	2×10^8	TEST	18,19	
			Sub.	NO	—	—	—	
15c								
CABLE	OUTSIDE	LONG	TEMP	220	318	TEST	18,19	
KERITE			Pr	17	121	TEST	18,19	
HT			RH	100	100	EVAL	11	
			Chem	NO	YES	EVAL	27	
			Rad	NO	2×10^8	TEST	18,19	
			Sub	NO	—	—	—	
16 CABLE	INSIDE	LONG	TEMP	286	318	TEST	18,19	
COLEMAN CABLE			Pr.	75	75	TEST	18,19	
ROME CABLE			RH	100	100	TEST	18,19	
			Chem	YES	YES	EVAL.	27	
			Rad	1.6×10^8	2×10^8	TEST	18,19	
			Sub	NO	—	—	—	

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT			Qual. Method	Document. Reference	COMMENTS
			Parameter	Spec.	Qual.			
17								
CABLE	OUTSIDE	LONG	Temp	220	AMB	EXP	5.10	
COLEMAN CABLE			Pr	17	ATM	EXP	—	
ROME CABLE			RH	100	AMB	EXP	—	
			Chem	NO	—	—	—	
			Rad	NO	—	—	—	
			Sub	NO	—	—	—	

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document. Reference	COMMENTS
18								
TRANSMITTER, PRESSURE	OUTSIDE	24 hr	Temp(°F)	AMB	286	TEST	18,19	NOT EXPOSED
FOXBORO 611GM-ASI			Pr(psia)	ATM	75	TEST	18,19	TO DBE
(RWST LEVEL)			RH	AMB	100	TEST	18,19	
			Chem	NO	YES	EVALUATION	27	
			Rad.	NO	3×10^4	TEST	18,19	
			Sub.	NO	—	—	—	
19								
TRANSMITTER, LEVEL	OUTSIDE	24 hr	Temp(°F)	AMB	200	VENDOR DATA	34	
BARTON/289			Pr(psia)	ATM	ATM	EXP.		
(RWST LEVEL)			RH	AMB	AMB	EXP.		
			Chem	NO	—	—		
			Rad	NO	—	—		
			Sub.	NO	—	—		
20								
TRANSMITTER, FLOW	INSIDE	SECONDS	Temp(°F)	286	320	TEST	31	NEMA IV ENCLO-
BARTON/332			Pr(psia)	75	ATM	EXP.		SURE. NOT EXPOSED
(STEAM FLOW)			RH	100	AMB	EXP.		TO DBE WHEN
			Chem	YES	YES	EVAL.	27	REQUIRED TO
			Rad	1.7×10^6	NO	—		FUNCTION.
			Sub.	NO	—	—		
21								
TRANSMITTER, PRES.	OUTSIDE	LONG	Temp(°F)	215	320	TEST	31	FOUR NOT EXPOSED
BARTON 1332			Pr(psia)	16	ATM	EXP.		TO DBE. OTHER
(CONT. PRESSURE)			RH	100	AMB	EXP.		TWO EXPOSED TO
			Chem	NO	—	—		HEAT OUTSIDE
			Rad	NO	—	—		CONTAINMENT
			Sub.	NO	—	—		
22								
TRANSMITTER, PRESSURE	INSIDE	SHORT	Temp(°F)	286	286	TEST	18,19	
FOXBORO			Pr(psia)	75	75	TEST	18,19	
611 GM-DSI			RH	100	100	TEST	18,19	
(PRZR. PRESSURE)			Chem	YES	YES	EVALUATION	27	
			Rad.	1.7×10^6	3×10^4	TEST	18,19	
			Sub.	NO	—	—	—	

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document Reference	COMMENTS
23								
TRANSMITTER, PRESSURE	OUTSIDE	SHORT	Temp(°F)	215	286	TEST	18, 19	EXPOSED TO
FOXBORO			Pr(psia)	16	75	TEST	18, 19	HERB OUTSIDE
611 GM-DSI			RH	100	100	TEST	18, 19	CONTAINMENT
(STEAM Pressure)			Chem.	NO	YES	EVAL.	27	
			Rad.	NO	3×10^4	TEST	18, 19	
			Sub.	NO	—	—	—	
24								
TRANSMITTER, LEVEL	Inside	—	Temp(°F)	286	318F	SPEC	33	NOT REQUIRED
FOXBORO			Pr(psia)	75	105	SPEC	33	TO PERFORM A
613M-MDL Modified			RH	100	100	SPEC	33	SAFETY FUNCTION
(Pzr Level)			Chem	YES	YES	EVAL.	27	
			Rad	1.7×10^6	3×10^4	EVAL.	18, 19	
			Sub.	NO	—	—	—	
25								
TRANSMITTER, LEVEL	Outside	SHORT	Temp(°F)	AMB	318	SPEC	33	NOT EXPOSED
FOXBORO			Pr(psia)	ATM	105	SPEC	33	TO DBE
613 DM-MSI			RH	AMB	100	SPEC	33	
(BAST LEVEL)			Chem	NO	YES	EVAL	27	
			Rad	NO	3×10^4	EVAL	18, 19	
			Sub.	NO	NO	—	—	
26								
TRANSMITTER, LEVEL	INSIDE	—	Temp(°F)	286	AMB			ALTERNATIVE
FOXBORO/613			Pr(psia)	75	ATM			INSTRUMENTATION
(SG LEVEL)			RH	100	AMB			AVAILABLE TO
			Chem	YES	NO			PERFORM THIS
			Rad	1.7×10^6	3×10^4			FUNCTION
			Sub.	YES	NO			
27								
TEMP. ELEMENT	INSIDE	—	Temp(°F)	286	200	SPEC	35	NOT REQUIRED
ROSEMOUNT /			Pr(psia)	75	ATM	EXP.		FOR DBE
176JA			RH	100	AMB	EXP.		
(RTD's)			Chem	YES	YES	EVAL.	27	
			Rad	1.7×10^6	200 R/hr	SPEC	35	
			Sub.	NO	—	—		

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT Parameter	Spec.	Qual.	Qual. Method	Document. Reference	COMMENTS
28								
BATTERY	OUTSIDE	LONG	Temp(°F)	AMB	110	VENDOR DATA	32	NOT EXPOSED
GOULD / FTA-19			Pr (psia)	ATM	ATM	EXP		TO DBE
			RH	AMB	AMB	EXP		
			Chem	NO	—	—		
			Rad	NO	—	—		
			Sub	NO	—	—		
29								
DIESEL GENERATOR	OUTSIDE	LONG	Temp	AMB	AMB	EXP		NOT EXPOSED
ALCO			Pr	ATM	ATM	EXP		TO DBE
251 F			RH	AMB	AMB	EXP		
WESTINGHOUSE 1900KW			Chem	NO	—	—		
			Rad	NO	—	—		
			Sub	NO	—	—		
30								
MOTOR, FANS	INSIDE	LONG	Temp	286	320	TEST	20	
WESTINGHOUSE			Pr	75	95	TEST	20	
588.5-CSP			RH	100	100	TEST	20	
			Chem	YES	YES	TEST	20	
			Rad	1.6×10^8	2×10^8	TEST	17, 18, 19	
			Sub	NO	—	—	—	
31								
CIRCUIT BREAKER	OUTSIDE	SECONDS	Temp	215	AMB	EXP		FAIL SAFE
WESTINGHOUSE			Pr	16	ATM	EXP		
DB-50A 1600A			RH	100	AMB	EXP		
			Chem	NO	—	—		
			Rad	NO	—	—		
			Sub	NO	—	—		
32								
I+C CABINETS	OUTSIDE	LONG	Temp	AMB	AMB	EXP		NOT EXPOSED
FOXBORO			Pr	ATM	ATM	EXP		TO DBE
			RH	AMB	AMB	EXP		
			Chem	NO	—	—		
			Rad	NO	—	—		
			Sub	NO	—	—		

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EQUIPMENT TYPE	INSIDE OR OUTSIDE CONTAINMENT	TIME NEEDED	ENVIRONMENT			Qual. Method	Document. Reference	COMMENTS
			Parameter	Spec.	Qual.			
33								
HVAC	OUTSIDE	LONG	Temp	AMB	122	SPEC	4, 6	
WESTINGHOUSE /			Pr	ATM	ATM	EXP.		
Z 162			RH	AMB	AMB	EXP		
STURTEVANT /			Chem	NO	—			
8015			Rad	NO	—			
			Sub	NO	—			
34								
SPLICES	INSIDE	LONG	Temp	286	350	TEST	36, 38	
RAYCHEM			Pr	75	85	TEST	36, 38	
WCSF-N			RH	100	100	TEST	36, 38	
(PRZR, FAN COOLER)			Chem	YES	YES	TEST	36, 38	
			Rad	1.6×10^5	2×10^8	TEST	36, 38	
			Sub	NO	—	—	—	

Ginna

(Documentation Reference)

1. NRC Penetration Steam Incident Test Report Crouse Hinds
2. Gilbert Spec. 520
3. Gilbert Spec. 711
4. Gilbert Spec. 5201
5. Gilbert Spec. 5324
6. Gilbert Spec. 5342
7. Gilbert Spec. RO-2239
8. Gilbert Spec. RO-2267
9. Gilbert Spec. RO-2400
10. IPCEA Std. S-61-402, Sect. 3.8
11. Kerite Memo 7/22/68
12. NEMA Std. SG-3
13. Westinghouse Spec. 676258
14. Westinghouse Spec. 676270
15. Westinghouse Spec. 676370
16. Westinghouse Spec. 676427
17. WCAP 7343 June, 1969
18. WCAP 7410-L, Vol. I & II
19. WCAP 7744, Vol. I & II
20. WCAP 9003, January, 1969
21. Nov. 29, 1977 Letter, Westinghouse to CPL, CPL-77-550
22. Westinghouse Letter, A.P. Colaiaco to E.E. Hoallen
23. Vendor Data on Asco Solenoid
24. Vendor Data on Copes Vulcan Solenoid
25. Vendor Data on Laurence Solenoid
26. Vendor Data on Versa Solenoid
27. WCAP 7153
28. Letter from White to Ziemann March 21, 1978
29. Gilbert Spec. 504
30. Technical Proposal for Electrical Penetration for Ginna Containment Structure by Westinghouse - September 4, 1974
31. WCAP 7354-L
32. Vendor Data On Gould Batteries
33. Westinghouse Spec. Sheet
34. Vendor Data on Barton 289 Transmitter
35. Rosemont RTD Spec.
36. Vendor Data on Raychem Splices
37. June 16, 1975 Letter to R. A. Purple from L. D. White
38. Franklin Research Center Final Report F-C5074, Qual. Tests per Splice Sleeves in Accident Environment, 4/79
39. Containment Purge Valves Closure Demonstration, RG&E Report, June 21, 1979
40. NRC Letter from D. Ziemann to L. D. White, Jr., RG&E, 10/23/79
41. RG&E Letter from L. D. White, Jr. to D. L. Ziemann, USNRC, 11/9/79
42. RG&E Letter from L. D. White, Jr. to D. L. Ziemann, USNRC, 12/17/79
43. Design Criteria, Standby Auxiliary Feedwater System, October 24, 1974.

ATTACHMENT

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By letter dated December 18, 1972, the Atomic Energy Commission's* Regulatory staff requested Rochester Gas and Electric Corporation (RG&E)[1] (licensee) to submit a detailed evaluation to substantiate that the R. E. Ginna Nuclear Power Plant (Ginna) could withstand the effects of a postulated rupture of any high energy fluid piping outside the primary containment, including the double ended rupture of the largest line in the main steam and feedwater systems. It was further requested that, if the results of the evaluation indicated changes to the facility were necessary to assure safe plant shutdown, information on the design changes and plant modifications be provided. Criteria for performing this evaluation were included in our December 18, 1972 letter. NRC and RG&E representatives met in Bethesda, Maryland, on February 1, July 18 and September 18, 1973, to discuss the NRC request and the scope of the expected RG&E analyses.

In response to our request, RG&E submitted a letter[2] dated November 1, 1973, that included a summary report "Effects of Postulated Pipe Breaks Outside the Containment Building" dated October 29, 1973. The results of this pipe whip and building pressurization analysis indicated that the intermediate building structure at Ginna was generally incapable of resisting pipe whip and pressurization effects of most postulated main steam and feedwater breaks within this building and from the adjacent turbine room. The licensee determined that modification of the structure or pipe encapsulation to provide the required protection was not practical[3] and an extensive volumetric examination program** to provide added assurance that the postulated piping system breaks would not occur was later proposed[4], initiated in 1973 and finally approved[5] by NRC in 1975.

*Currently known as the Nuclear Regulatory Commission (NRC).
**In accordance with the requirements of 10 CFR Part 50, Section 50.55a, paragraph (g), RG&E submitted by letter dated 7/2/79, the "Ginna Station In-Service Program for the 1980 through 1989 Interval".

Additional information[6] was submitted by RG&E's letter dated May 24, 1974. This information was responsive to NRC concerns for postulated high energy line breaks outside containment and potential effects on safety related equipment that might be required to cool the core. The licensee later submitted a schedule[7] for analysis and plant modifications. As a result of the High Energy Line Break Outside of Containment evaluation, plant changes have been made[19] as summarized below:

-- An augmented In-Service-Inspection Program has been initiated[4] to further reduce the probability of a main feedwater or steam line rupture.

-- A Standby Auxiliary Feedwater System has been added to further improve steam generator feedwater reliability and specifically to substitute for the auxiliary feedwater in the low probability that auxiliary feedwater pumps are damaged due to nearby high energy pipe breaks within the intermediate building.

-- Check valves have been added to existing auxiliary feedwater lines near the connections to the main feedwater lines to minimize the auxiliary feedwater piping that is pressurized during normal operation.

-- Two parallel remotely operated valves have been added to a crossover line between the motor driven pump discharges to provide additional auxiliary feedwater makeup capability.

-- A large metal plate jet shield has been installed underneath the main steam header in the Intermediate Building to protect the service water piping from a postulated crack in the main steam line. Jet Impingement Shields have been added to protect vital equipment including containment isolation valves, motor generators, transfer switches, cable trays, terminal boxes and wiring, pressure transmitters and reactor trip breakers. Also jet shields have been added to protect main steam bypass valves and piping and other locations listed by RG&E.

-- Instrument cabling has been relocated to areas that will not be affected by postulated high energy pipe breaks.

-- The heating and ventilation system has been modified to withstand postulated high energy pipe breaks without further endangering the capability to safely shut down the plant.

-- The east end of the cable tray that connects the Intermediate Building and the Relay Room of the Control Building has been sealed to prevent damage that could result from a postulated high energy line break.

-- Openings around pipes and cable trays that pass through the areas required for safe shutdown of the plant have been sealed to prevent steam leakage into these areas in the unlikely event of steam or feedwater line breaks in the Turbine Building.

-- Steam generator blowdown lines have been rerouted through the sub-basement to minimize the potentially detrimental effects of breaks in these lines within the Intermediate Building.

-- Sufficient floor grating has been installed at manholes to guard against flooding of safety related equipment in the Intermediate Building resulting from an assumed feedwater line break.

-- Steam line pressure and feedwater flow transmitters have been relocated away from the locations that could be affected by postulated high energy line breaks.

-- Pressure shielding steel diaphragm walls are being installed at selected locations in the Turbine Building to assure continued operability of safety related equipment following a postulated high energy pipe break in the Turbine Building.

-- RG&E committed, by letter dated June 27, 1979, to provide jet shielding for one atmospheric steam dump valve, all steam generator code safeties, and the two main steam bypass valves and their associated 3-inch piping. This shielding would be provided in conjunction with the Systematic Evaluation Program (SEP). Furthermore, modifications to the Intermediate Building wall resulting from analysis of high energy line breaks in the Turbine Building will be made as necessary upon completion of the SEP.

Discussion

Ginna is a pressurized water reactor that utilizes a reinforced concrete containment which contains the entire primary coolant system, including the steam generators.

The criteria and requirements used by the licensee and the staff for evaluating the high energy line breask outside containment are summarized as follows:[9]

1. Equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of essential equipment, should

be protected from all effects of ruptures in pipes carrying high energy fluid, up to and including a double-ended rupture of such pipes, where the service temperature and service pressure conditions of the fluid exceed 200°F and 275 psig. Breaks should be assumed to occur in those locations specified in the "pipe whipe criteria". The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement), and environmental.

2. In addition, equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of essential equipment, should be protected from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes normally carrying high energy fluid routed in the vicinity of this equipment. The postulated size of the cracks was either 1/2 the pipe diameter in length and 1/2 the wall thickness in width (critical crack size) or equivalent pipe flow cross section in area.

The licensee evaluated all piping outside containment that contains high energy fluid and is in the same building with or in the proximity of safety related equipment required for safe shutdown. These lines are:

Main Steam
Feedwater
Auxiliary Feedwater
Steam Supply to Auxiliary Feedwater Pump Turbine
Steam Generator Blowdown
Charging Line
Plant Steam

The licensee's evaluation postulated longitudinal and circumferential breaks at high stress locations specified by the NRC criteria for piping break locations and considered the effects of pipe whip, jet impingement, pressurization, environment and flooding. For the evaluation of piping cracks, effects of pipe whip and pressurization were not applicable. The licensee described the course of events following various size breaks of the main steam and feedwater lines at different reactor operating conditions. The equipment necessary to bring the plant to a safe shutdown was listed. The licensee's analyses indicate that the Intermediate Building, through which the main steam and feedwater lines pass from the Containment Building to the Turbine Building, cannot withstand most main steam line and feedwater line breaks. This results from the pressurization of the building following the postulated high energy pipe break exceeding the design pressure for the concrete block walls and the roof, and from the structural capabilities not being sufficient to withstand the effects of pipe whip. Some equipment that is used to maintain the reactor in a safe shutdown condition is located in this building and might be rendered inoperable. This equipment includes instrument channel cables, service water piping, and the auxiliary feedwater system. A number of alternatives to the final plant modifications[2] were evaluated and considered by the licensee to be impractical[3].

The Steam Line Pressure and Feedwater Flow Transmitter Signal Cables have been relocated to areas with no high energy lines. The sensing lines for the transmitters are susceptible to damage since they connect to high energy lines. However if they rupture, the channels fail downscale and since low steam line pressure and low feedwater flow produce the trips for protective action, the channels fail in the safe direction. In addition, the signal cables for a cold leg reactor coolant temperature channel from each loop have been rerouted outside the Intermediate Building to provide the operators with additional information to follow the course of the accident.

The following instrument channels are isolated from the effects of high energy line break outside containment.

Instrument	No. of Protected Channels	No. Required to Trip
Steam Generator Level	2 per loop	2 per loop 1 per loop with Steam Flow-Feed Flow Mismatch
Steam Line Flow	2 per loop	1 per loop
Feedwater Flow	2 per loop	1 per loop
Steam Line Pressurizer	3 per loop	2 per loop
Pressurizer Pressure	2 per loop	1 per loop
Pressurizer Level	1 per loop	1 per loop
Reactor Coolant Temperature	2 per loop	NA

The instrument channels or signal cables that remain in the unprotected areas of the Intermediate Building are likely to perform their trip function by providing protective action signals for the steam or feedwater line breaks either in the normal fashion or by the fail-safe trip. This is because any failure which could occur would most likely be a separation of the sensing line or signal cable and, except for the steam flow channels, loss of signal trips the channel. Also the required protective actions can be initiated by the response of a single one of the parameters monitored by the channels above, such as low steam pressure on two channels in one loop, or by a number of diverse responses, such as low pressurizer pressure and level on one channel. Therefore, the protected channels and those remaining in the unprotected area maintain the required diversity and redundancy for reactor protection systems. In addition, the protected channels will ensure that the operator is provided with information for the course of the accident. On this basis, we find these modifications acceptable.

Evaluation

The Augmented In-Service Inspection Program[4] proposed and implemented by the licensee consists of radiographic examination of all welds at the design basis break locations in the main steam and feedwater lines and at other locations where a failure would result in unacceptable consequences. The examination techniques, procedures, and inspection intervals are based on the requirements of Class 2 components of Section XI of the ASME Boiler and Pressure Vessel Code. The program* is based on ten year inspection intervals with the first interval running from 1973 to 1982. The extensive in-service inspection program is designed to preclude design bases or consequential main steam or feedwater pipe breaks.

During each third of the first inspection interval, the program provides for examination of all welds at specified design basis break locations and one-third of all the welds at specified locations where a weld failure could result in unacceptable consequences. During each one-third of the succeeding 10-year intervals, the program provides for examination of one-third of the welds at design basis break locations but continues unchanged with one-third of the welds at locations where a weld failure could result in unacceptable consequences. This program is designed to detect flaws capable of causing pipe failure. The frequency of reinspections is designed to detect any change in condition in advance of a potential failure. We have concluded that this augmented inspection program is a prudent measure to ensure a very low probability of any break in the main steam and feedwater lines. The inspection requirements for this program have been incorporated into the Technical Specifications[5].

The Instrumentation Channels that initiate the protective action in the event of a main steam line or feedwater line break are: Pressurizer Pressure, Steam Line Pressure, Steam Line Flow, Feedwater Flow, Pressurizer Water Level, and Steam Generator Water Level.

The pressurizer pressure, steam line flow, pressurizer water level and steam generator water level transmitters are located inside containment and, therefore, their operability would not be affected by a high energy line break outside the containment. Some of the signal cables from these transmitters, however, are routed through cable trays in the Intermediate Building. To ensure that the minimum number of these channels required to produce the protective actions (safety injection, reactor trip, and feedwater and steam line isolation) are not adversely affected by a high energy line break in the Intermediate Building, their signals have been rerouted out of other containment penetrations and do not pass through the Intermediate Building.

*RG&E letter dated 7/2/79 presents the 1980-1989 Ginna ISI Program.

The Auxiliary Feedwater System is also located in the Intermediate Building with all three pumps in the same vicinity. There are two motor driven pumps and one steam driven pump. These pumps are only used during start-up and normal or emergency shutdown of the plant. The pumps are susceptible to damage from the effects of breaks in the main steam and feedwater lines and the auxiliary steam and feedwater lines. To ensure the heat removal capability for core cooling, the licensee proposed and later installed a Standby Auxiliary Feedwater System adjacent to the Auxiliary Building along the south wall. The Standby Auxiliary Feedwater Pumphouse is a seismic Class I concrete structure supported by caissons[9].

The Standby Auxiliary Feedwater System consists of two, independent 100 percent capacity subsystems in a new structure remote from high energy lines. The discharge piping from the pumps was routed through the Auxiliary Building, enters the containment through penetrations remote from the main steam and feedwater lines, and connects to the feedwater lines near each steam generator with check valves near the connection to minimize the amount of line pressurized during normal plant operation.

The pumps take suction from the Service Water Loops inside the Auxiliary Building, are motor driven from the Engineered Safety Features busses, and are manually started from the control room in the event that the Auxiliary Feedwater pumps, which start automatically, are not operable. The analysis performed by the licensee assumes that feedwater is not available for 10 minutes following the worst case line break. This is ample time for the control room operator to take action since alarms and indications are available in the control room to alert the operator to the lack of effective auxiliary feedwater flow and the standby pumps can be put into operation from the control room.

Our concerns for the structural, mechanical and material aspects of the modifications were adequately addressed[10] by the RG&E letter dated July 28, 1978, in response to our request dated June 21, 1978.

In the event of loss of off-site power, the pumps would be powered by the diesel generators. The diesel generators have sufficient capacity for this additional 225 Kw load. However, to prevent an overload of the feedbreakers tying the diesels to the buses, an interlock has been installed to prevent starting a standby pump when its associated auxiliary pump is running on the diesel.

The Standby Auxiliary Feedwater Pump Building and System design satisfied the codes and standards applicable in 1974 when the building was designed[9,10]. We conclude that these modifications provide an acceptable backup to the Auxiliary Feedwater System for maintaining the plant in a safe shutdown condition. The scope of the Safety Evaluation of the Standby Auxiliary Feedwater System is presented in the enclosed Appendix 1. On the basis of this evaluation, the Technical Specification changes proposed by RG&E[8], which we revised with RG&E concurrence, are acceptable. Also, the same operating procedure requirements for the prevention of water hammer in the Auxiliary Feedwater System should be applied to the Standby Auxiliary Feedwater System.

The Ventilation Systems were evaluated to determine whether the steam from high energy line breaks would intrude into an area where personnel or equipment important to safety would be endangered. It was determined that modifications were necessary to the control room lavatory exhaust, the control building ventilation equipment room relief opening, the relay room cable tray openings and tunnel, the battery room exhaust and cable tray openings, the diesel generator room piping and cable tray openings, and some interconnecting ventilation ducts between the Intermediate Building and the Auxiliary Building. All of these openings have been sealed and the exhausts have been ducted to areas not subject to intrusion of the steam from a high energy line break. Based on the above, we conclude that these modifications reduce the probability of adverse consequences from the postuated high energy line breaks and are, therefore, acceptable.

Pressure Shielding Steel Diaphragm Walls were proposed by RG&E's letter dated February 6, 1978[11]. The steel diaphragm walls were to have been erected between the Control Building and the Turbine Building and between the Diesel Generator Rooms and the Turbine Building. The walls would:

-- Comply with the requirements for physical protection of licensed activities against industrial sabotage (10 CFR Section 73.55)

-- Provide protection from postulated fires on the operating level of the Turbine Building

-- Provide protection from postulated high energy line breaks in the Turbine Building

We met with representatives of the licensee in Bethesda, Maryland, on February 15, 1978, to discuss fire protection and structural aspects of the diaphragm wall and on January 30, 1979, to discuss structural design criteria for the wall. On the basis of information provided by the licensee[12, 14] we have concluded[15] that the steel wall and door that have recently been added between the Control Room and the Turbine Building are designed for high power rifle resistance (level IV bullet resistance) and, therefore, meet the requirements of 10 CFR 73.55.

We have reviewed the adequacy of the Steel Diaphragm walls between the Control Building and the Turbine Building and between the Diesel Generator Rooms and the Turbine Building with respect to fire protection. Based on the information provided by the licensee[11, 12, 13, 14], we have concluded[16] that the concept of a steel diaphragm wall between the Turbine Building and the Control Room protected by an automatically actuated water curtain is acceptable, but the details of the water supply and actuation system must be submitted for our review. Concerning the Pressure Shielding Steel Diaphragm Turbine Building walls adjacent to the Diesel Generator, Relay and Battery Rooms, the licensee has agreed[16] to conduct studies to determine what active and passive systems should be installed to prevent structural failure from fire that would jeopardize safe shutdown of the plant. We have also identified[16] the requirements for fire doors in the areas where the steel diaphragm wall is being constructed.

The NRC Safety Evaluation of the structural adequacy of the Pressure Shielding Steel Diaphragm Walls is presented in the enclosed Appendix 2. We have concluded on the basis of information presented in licensee letters[11, 12, 13, 14] and during a meeting with NRC representatives[17], that the structural criteria and design methods for the steel diaphragm walls are adequate to assure safe shut down of the reactor following a high energy pipe break in the Turbine Building. However, our conclusion is based on the premise that the peak Turbine Building pressure and temperatures that the Turbine Building steel diaphragm walls adjacent to the Diesel Generator Rooms and the Control Building (Control Room, Relay Room and Battery Room) must withstand, results from a postulated rupture of the 20" Feedwater Line. Since the licensee had previously reported[6] that the pressure on the operating level of the Turbine Building as a result of a break in the 24" or 36" steam line peaked at 0.098 psig with steam relief through the building exhaust fans in the wall and roof, and later reported 0.70 psig pressure peaks[17], resulting from a break in the 20" main feedwater line, we requested RG&E to submit additional analysis. The licensee's basis[10] for using the Main Feedwater 20" pipe break to determine peak Turbine Building transient pressure and temperature for the structural design of the new steel diaphragm walls was justified because of the augmented In-Service Inspection of all welds in the steam lines in the Turbine Building and the resultant low probability of a large break in the steam lines. Nevertheless, at our request, by letters dated May 17, 1979 and July 6, 1979, the licensee provided supplementary information[18,20] which in addition to the Turbine Building pressure transient analyses for postulated feedwater pipe breaks, also included steam line breaks in the Turbine Building. As expected, these calculations showed that the steam line break pressure transients were significantly greater than originally reported[6].

The following additional information[17, 18, 20] provided by RG&E:

-- The peak pressure transients in the Turbine Building calculated by the licensee are less than the 0.7 psig structural design pressure for the steel diaphragm wall on the mezanine floor along the control room wall and less than the 1.14 psig structural design pressure for the steel diaphragm walls on the operating floors at the relay, battery ahd diesel generator room walls.

-- The new steel diaphragm walls are at nearly opposite ends of the Turbine Building from the high energy piping thereby providing adequate separation to preclude wall damage at these locations because of pipe whip or jet impingement that could accompany a high energy pipe break in the Turbine Building.

Based on this information and our detailed Safety Evaluation of the pressure shielding steel diaphragm walls in the Turbine Building which is included as Appendix 2 to this Safety Evaluation, we have concluded that the structural adequacy of the steel diaphragm walls as described by the licensee[10] is acceptable. A schedule for completion of the installation of the steel diaphragm walls, in accordance with the information provided, should be submitted within 60 days.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Enclosures:
1. Appendix 1, "Detailed Evaluation of the Standby Auxiliary Feedwater System - R. E. Ginna"
2. Appendix 2, "Detailed Evaluation of the Pressure Shielding Steel Diaphragm in Turbine Building - R. E. Ginna"

Date: August 24, 1979