Commonwealth Edison
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Address Reply to: Post Office Box 767
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January 18, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Station Units 1 and 2
Braidwood Station Units 1 and 2
Advance FSAR Information
NRC Docket Nos. 50-454/455/456/457

Dear Mr. Denton:

This is to provide advance copies of information which will be included in the Byron/Braidwood FSAR in the next amendment. Attachment A to this letter lists the information enclosed.

One (1) signed original and fifty-nine (59) copies of this letter are provided. Fifteen (15) copies of the enclosures are included for your review and approval.

Please address further questions to this office.

Very truly yours,

F. B. Lentine
for T. R. Tramm
Nuclear Licensing Administrator
Pressurized Water Reactors

Attachment

3129N

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ATTACHMENT A

LIST OF ENCLOSED INFORMATION

I. FSAR Question Responses

Revised:	10.47	221.2
	10.50	421.22
	22.06	423.30
	22.30	423.35
	22.73	423.38
	40.143	423.39

II. FSAR Text Changes

Subsection 5.4.7.2.6 (returned to original state)

Chapter 14 preop and Startup tests: Tables 14.2-7, 11, 21, 58

III. Miscellaneous Items

Byron Station Emergency Preparedness Open Item Responses 1-7, 9, 10, 11

Byron Station Emergency Preparedness

Open Item 1

Establish formal letters of agreement with appropriate agencies and organizations including law enforcement, ambulance service, medical and hospital support, fire departments and state and local authorities responsible for implementation of protective measures for the public. These agreements must specify the emergency measures to be provided to the licensee and the mutual acceptable criteria for their implementation. (Sections A & L)

RESPONSE

Byron Station has obtained written agreements for support from the Ogle County Sheriff's Department and the Byron Fire Department. Additional contacts are being made in accordance with NUREG-0654, Section C.4. It is expected that all necessary agreements will be obtained by June, 1982. These agreements will specify the emergency measures to be provided to Commonwealth Edison and the mutual acceptable criteria for their implementation.

Open Item 2

The responsibilities of the station director which may not be delegated must be clearly specified in the plan. Among the responsibilities which may not be delegated is the decision to notify and recommend protective actions to off site authorities. The actual notification and transmittal of these recommendations may be delegated. (Section B)

RESPONSE

The next revision to the Byron Station Emergency Plan Annex will specify those responsibilities of the Station Director which may not be delegated. Those responsibilities which may <u>not</u> be delegated are as follows:

a. Declaration that an Unusual Event, Alert, Site Emergency, or General Emergency Condition exists. General categorization of emergencies may be done by other plant personnel in accordance with approved Station procedures. But the final decision to declare the emergency condition rests with the Station Director; and

Byron Station Emergency Preparedness

Open Item 2 (Response-Continued)

b. Decision to notify and recommend protective actions to offsite authorities in the case where a Site Emergency or General Emergency condition exists and the Recovery Manager or Corporate Command Center Director have not been contacted or are not prepared to make an informed decision. In all other cases, the decision to notify and recommend protective actions to offsite authorities shall be made by the Recovery Manager or Corporate Command Center Director. This responsibility may not be delegated. Actual notification and transmittal of these recommendations may be delegated.

Open Item 3

A description of the 30 minute augmental capability must be provided in the plan, along with a description of the systems to be used to ensure that the goals of shift augmental can be met. (Section B)

RESPONSE

NUREG-0654 Criterion II.B.5 states that "the licensee must be able to augment on-shift capabilities within a short period after declaration of an emergency." It further defines that short period as 30 and 60 minutes. These time frames are considered to be the goal for augmentation. Commonwealth Edison cannot commit to rigid, inviolate times due to the diversity of residential patterns for station personnel, possible adverse weather conditions, and possible road congestion. This position is consistent with that stated in a letter dated 10/26/81, from W. J. Dircks of the NRC to D. F. Knuth of KMC, Inc. Byron Station will strive to meet the 30 minute criteria for approximately one-half of the planned augmentation force. The Station augmentation staff will be able to perform the emergency functions outlined in Table B-1 of NUREG-0654.

To facilitate meeting the augmentation commitment, Byron Station will establish a 24 hour duty-call individual who would be notified first after a station emergency is declared. This individual would initiate a prioritized notification (call-list) procedure. The procedure identifies individuals who are capable of fulfilling the specific response functions listed in the Generating Stations Emergency Plan, Revision 2, Figure 4.2-3. The call-lists are prioritized by least travel time of station staff.

Byron Station Emergency Preparedness

Open Item 3 (Response-Continued)

In addition, Byron Station will initiate unannounced offshift notification drills at least once every six months. These drills will involve implementation of the Station call-list procedure and documentation of the times when persons are notified. These drills will serve to demonstrate the capability to augment the onshift staff within a short period following an emergency declaration.

Open Item 4

Identify the parameter values for each emergency class in the plant. (Section D)

RESPONSE

By June, 1982, the Byron Emergency Plan Annex will be revised to incorporate more detailed emergency action levels. For EALs that involve parameters monitored by available plant instrumentation, parameter values will be identified for each emergency class.

Open Item 5

Include in the plan all of the applicable initiating conditions described in Appendix 1 of the NUREG-0654, Revision 1 as well as those based on FSAR emergency accidents. (Section D).

RESPONSE

By June, 1982, the Byron Emergency Plan Annex will be revised to incorporate more detailed emergency action levels. The EALs will include applicable initiating conditions described in Appendix 1 of NUREG-0654. The revised EALs will also incorporate FSAR analyzed accidents for which there are no similar conditions described by Appendix 1 of NUREG-0654.

Open Item 6

A summary of the descriptions of the prompt alert and notification system must be included in the plan. The full system meeting the design objective of appendix 3 in NUREG-0654, Revision 1, must be developed and installed. (Section E)

Byron Station Emergency Preparedness

Open Item 6

RESPONSE

Outlined below is the proposed Prompt Notification System for the Byron Station. This system is designed to meet the requirements as specified in NUREG-0654, Appendix 3.

The system consists of three parts:

a. A permanently installed outdoor notification system within the 0 to 5 mile radius around the station. The 0 to 5 mile radius around the station is primarily an agricultural area with a population density well below 2000 persons per square mile. The installed notification system will essentially cover all inhabited areas with a minimum noise level of 60 db using an attenuation factor of 10 db loss per distance doubled. For the possibility of a dwelling not being exposed to a 60 db minimum noise level, a local coverage siren or an inhouse warning receiver will be utilized.

b. A permanently installed outdoor notification system covering the heavily populated areas within the 5 to 10 mile radius. The area outside the five mile radius and inside the ten mile radius contains a number of communities that will be covered by installed notification. These systems will utilize existing sirens plus additional sirens to ensure complete coverage with either a 60 db minimum or 10 db above daytime background.

c. A mobile notification system for the remainder of the area within the 5 to 10 mile radius. The total 1980 population of the area outside the five mile radius and inside the ten mile radius is approximately 12,000. Of this total, that population residing in communities within the 5 to 10 mile radius will be covered by the installed notification system mentioned in part b above. The remaining population live in rural residences or farm steads and will be alerted by a mobile system including sirens and public address. The plan for a mobile notification system includes the use of law enforcement vehicles with siren and portable announcing systems. Contained in the Emergency Plans of each of the counties associated with Byron Station, will be the general guidance for early notification of the population within the county. The routing of law enforcement vehicles through the 5 to 10 mile EPZ will be accomplished by the county sheriff dispatcher aided by the state police dispatcher if needed.

Byron Station Emergency Preparedness

Open Item 6 (Response-Continued)

The purpose of the prompt notification system is to direct residents indoors, where they are instructed to tune to designated emergency information radio stations. The mobile system proposed will adequately perform this function for those personnel not already indoors. Thos personnel already inside (i.e. during foul weather, family dinner, etc.) are already in the desired location.

A second portion of the prompt notification system required instructional messages to be given to the public. The state procedures for providing these messages will be contained in the local Emergency Plans for Byron Station.

Open Item 7

Provide a copy of the public information brochure for review and approval for the NRC and FEMA prior to distribution. Distribution of the brochure must occur prior to fuel load. (Section G).

RESPONSE

A pamphlet entitled "Byron Station -- What to do in Case of a Nuclear Station Emergency," will be provided to the NRC and FEMA prior to distribution. This brochure will follow the same format as those already distributed for other Commonwealth Edison nuclear stations. The first nine pages of the brochure are essentially generic to all Commonwealth Edison stations and have already been reviewed by the NRC and FEMA. The brochure will be distributed prior to Byron Unit 1 fuel load.

Byron Station Emergency Preparedness

Open Item 9

Describe in the plan the non-radiological process monitor that will be used under accident conditions (e.g., reactor coolant system pressure and temperature, flow rates, liquid levels, etc. and the hydrological monitors that will be used to assess Rock River flood or low water conditions. (Section H).

RESPONSE

Section 7.3.4 of the Byron Emergency Plan Annex will be revised by June, 1982, to describe onsite process monitors used to properly assess plant status following an accident. The instrumentation includes that listed below:

POSTACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Reactor Coolant Outlet Temperature - T_{HOT} - Wide Range	1/Loop
3. Reactor Coolant Inlet Temperature - T_{COLD} - Wide Range	1/Loop
4. Reactor Coolant Pressure - Wide Range	2
5. Pressurizer Water Level	2
6. Steamline Pressure	2/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator
8. Refueling Water Storage Tank Water Level	2
9. Boric Acid Storage Tank Level	2

The hydrological characteristics of the Byron Station vicinity are described in Section 2.4 of the Byron FSAR. The river screen house is the only structure that could be affected by flooding on the Rock River and is designed for a combined event flood, wehre a combined event flood is defined

Byron Station Emergency Preparedness

Open Item 9 (Response-Continued)

as a flood on the Rock River having a $1x10^{-6}$ annual probability of being exceeded at a 90% confidence level. All other Byron Station structures are 161 feet or more above the Probable Maximum Flood level of the Rock River.

The maximum water requirement for Byron Station is 88.7 cfs. In the unlikely event that emergency requirements cannot be satisfied by surface water withdrawals from the Rock River, groundwater wells will serve for makeup to the essential service water cooling towers. The minimum design operating level of the essential service water makeup pumps is 3.8 feet lower than the water level for the 1-day 100-year low flow drought condition.

Because of the site hydrological characteristics given above, plant operation will not be affected by Rock River water level conditions and therfore, hydrological monitors have not been installed. The Rock River is not used for any public water supply. There are no recorded plans for any future public water supply usage from the Rock River. The nearest surface water users downstream from Byron Station are on the Mississippi River over 115 miles away. This allows for sufficient mixing that makes permanently installed hydrological monitors unnecessary. Provisions have been mdae for grab samples to establish release rates. In performing dose calculations from liquid releases, Byron Station uses a historical average river flow value, Fw, as a parameter in the liquid release model.

Open Item 10

Within one year before the issuance of the operating license for full power operation, successfully complete a full scale exercise. (Section N)

RESPONSE

A full scale exercise will be conducted within one year prior to the issuance of the Byron Unit 1 operating license for full power operation. The exercise is currently scheduled to be held in October 1982.

Byron Station Emergency Preparedness

Open Item 11

Indicate in the plan that communication drills from Byron Station to Wisconsin will be tested quarterly; and that communications with the NRC headquarters in Region III Operations Centers from the TSC, EOF, and Control Room will be tested monthly. (Section N)

RESPONSE

The Byron Emergency Plan Annex will be revised to indicate that communications will be tested quarterly with the State of Wisconsin. Communications with the NRC Region III OFfice and the NRC Operations Center will be tested monthly from the TSC, EOF, and Control Room. The Plan will be revised accordingly, by June, 1982.

5.4.7.2.6 Manual Actions

The RHRS is designed to be fully operable from the control room for normal operation. Manual operations required of the operator are: closing the suction valves to the RWST, opening the suction isolation valves, positioning the flow control valves downstream of the RHRS heat exchangers, and starting the RHR pumps.

Manual actions required outside the control room, under conditions of single failure, are discussed in Subsection 5.4.7.2.5.

5.4.7.2.7 System Operation

Reactor Startup

Generally, while at cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At initiation of the plant startup, the RCS is completely filled, and the pressurizer heaters are energized. The RHRS is operating and is connected to the CVCS via the low-pressure letdown line to control reactor coolant pressure. During this time, the RHRS acts as an alternate letdown path. The manual valves downstream of the residual heat exchangers leading to the letdown line of the CVCS are opened. The control valve in the line from the RHRS to the letdown of the CVCS is then manually adjusted in the control room to permit letdown flow.

After the reactor coolant pumps are started, the residual heat removal pumps are stopped but pressure control via the RHRS and the low-pressure letdown line is continued until the pressurizer steam bubble is formed. Indication of steam bubble formation is provided in the control room by the damping out of the RCS pressure fluctuations, and by pressurizer level indication. The RHRS is then isolated from the RCS and the system pressure is controlled by normal letdown and the pressurizer spray and pressurizer heaters.

Power Generation and Hot Standby Operation

During power generation and hot standby operation, the RHRS is not in service but is aligned for operation as part of the ECCS.

Reactor Cooldown

Reactor cooldown is defined as the operation which brings the reactor from no-load temperature and pressure to cold conditions.

The initial phase of reactor cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators.

TABLE 14.2-7

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

(Preoperational Test)

Plant Condition or Prerequisite

Prior to core load, RCS at ambient temperature and pressure.

Test Objective

To verify operation of the safeguards logic systems for all conditions of trip logic.

Test Summary

Prior to core loading the operation of the Engineering Safety Features Actuation System will be demonstrated. It will include actuation of containment systems, emergency core loading systems (ECCS), habitability systems, fission product removal, and control systems. It will demonstrate redundancy, coincidence and safe failure on loss of power. It will demonstrate that both trains are independent, and that the response time from the measured variable to the sensor meet the Technical Specifications.

Acceptance Criteria

The safeguards logic system operates in accordance with acceptance criteria developed from the system design criteria, safety analysis report, and plant installation.

TABLE 14.2-11

AUXILIARY POWER SYSTEM

(Preoperational Test)

Plant Condition or Prerequisite

Prior to core load, RCS at ambient temperature and pressure.

Test Objective

To verify proper operation of the auxiliary power transformers, breakers, switchgear, and other components.

Test Summary

Prior to core loading the auxiliary power system will be tested and verified that all interlocks, protective features, alarms, and indications are operational. It will demonstrate that a loss of offsite power will transfer to onsite power and function as per its design capabilities. It will also demonstrate that the capacity of the system auxiliary transformer (starting transformer) to supply power to its unit's vital buses while carrying its maximum load of plant auxiliaries and the other unit's vital buses and testing of the automatic transfer capabilities. Tests of the vital buses will be performed as early as the necessary components become available for testing, but not during theperiod when electrical separation requirements are in effect for Unit 1 operation and Unit 2 construction. It will also demonstrate that the two ESF Divisions 11 and 12 are completely independent. All metering, voltages, and proper phase rotations will be demonstrated to perform as per its design. Full accident load testing will be performed using the system auxiliary transformer, reserve feed, and diesel-generator. The voltage levels at the vital buses are predicted throughout the anticipated range of voltage variation of the off-site power source by an engineering analysis. During full load accident testing, voltage readings of the off-site power source and voltage readings of the vital buses will be taken. These readings will be correlated with analysis results to verify the voltage levels at the vital buses are optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the off-site power source.

Acceptance Criteria

Each 480V or 4-kV Auxiliary Power bus can be supplied from the appropriate power sources. Transformers, breakers, relays, and meters function in accordance with system design requirements.

TABLE 14.2-21

LEAK DETECTION SYSTEM

(Preoperational Test)

Plant Condition or Prerequiste

Prior to core load, RCS at ambient temperature and pressure.

Test Objective

To ensure the Containment Sump and Reactor Cavity Sump leak detection systems are functional.

Test Summary

Reactor Cavity Sump and Containment Sump level and flow monitoring instrumentation shall be functionally tested to verify proper operation.

Acceptance Criteria

The leak detection system is verified to function in accordance with specifications derived from system design criteria and plant installation.

TABLE 14.2-58

INTEGRATED HOT FUNCTIONAL HEATUP

(Preoperational Test)

Plant Condition or Prerequisite

Prior to core load following hydrostatic testing of primary and secondary systems.

Test objective

To demonstrate ability to heat primary system to normal operating temperature and pressure.

Test Summary

The reactor coolant system will be taken to normal operating temperature and pressure using reactor coolant pump heat input. Tests will be performed to demonstrate operation of excess letdown and seal water flow paths and letdown flow rates. Thermal expansion checks will be conducted. The Snubber thermal monitoring program will be completed. Isothermal calibration of resistance temperature detectors and incore thermocouples will be performed.

Acceptance Criteria

Preoperational tests to be performed during plant heatup are accomplished and the reactor coolant system taken to normal operating temperature and pressure in accordance with Subsection 3.9.2.

QUESTION 010.47

"Your response to Q010.30 has not provided an adequate analysis to demonstrate that drainage of leakage water away from safety-related components or systems is adequate for worst case flooding resulting from postulated pipe breaks or cracks in high or moderate energy piping near these safety-related components or systems. The analysis must shown that drainage by natural routes such as stairwells or equipment or hatches by the non-seismic Category I drainage system under failed conditions is adequate to prevent the loss of function of safety-related components and systems. As an example, show that a crack in one essential service water pump room will not flood out the other redundant pump before operator action can be taken to isolate the leak assuming a failed non-safety grade sump alarm system. Worst case locations should be assumed for this example and for other safety-related systems listed in FSAR Table 3.6-1.

It is our position that unless drainage capability by natural or by failed non-siesmic Category I drainage systems can be demonstrated, you should provide the following for all areas housing redundant safety-related equipment.

1. Leak detection sumps shall be equipped with redundant safety grade alarms which annunciate in the control room. Verify that if operator action is required on receipt of alarm that flooding of redundant safety grade equipment will not occur within 30 minutes; or

2. Provide separate watertight rooms and independent drainage paths with leak detection sumps for each redundant safety-related component."

RESPONSE

The following report outlines a confirmatory study of the flooding which would result in the auxiliary building following postulated high or moderate energy line breaks.

AUXILIARY BUILDING FLOODING ANALYSIS

I. INTRODUCTION

One potential problem which could result from a break of a high or moderate energy line is flooding of safety-related equipment. This report will address flooding

of safety-related equipment in the auxiliary building. The containment flooding concerns have been addressed previously. Turbine building flooding is not a safety concern because safety-related equipment is not located in the turbine building (see response to NRC Question 010.50).

II. PLANT FLOODING DESIGN BASIS

The auxiliary building design prevents flooding from affecting safety-related equipment required to safely shut the plant down. The auxiliary building floor drain system, although designated Category II Safety Class D, is designed and supported to withstand seismic loads. In addition, all other Category II Safety Class D piping including building roof drains is designed and supported to withstand seismic loads where these piping or drains are in areas that contain safety-related equipment. In many areas, the pipe is actually imbedded in the concrete floors and walls of the auxiliary building. If the floor drains are not available or adequate, water will leave rooms through doorways and flow down stairways until it reaches the auxiliary building basement.

To ensure against flood damage, certain areas such as the essential service water pump area have been designed with watertight penetrations to prevent leakage from adjacent flooded areas. The auxiliary building has high capacity Category I sumps and pumps with Category I level indication to prevent floods from causing high water levels in the basement. In addition, leak detection sumps with Category I level indication are located throughout the various levels of the auxiliary building in areas that contain safety-related equipment.

III. SCOPE

This flooding study investigates potential breaks in the auxiliary building to show that the postulated high and moderate energy line breaks will not result in flood levels high enough to adversely impact the ability to safely shut the plant down.

The most limiting potential breaks will be investigated. It will be assumed that isolable breaks will be isolated within 30 minutes. This is reasonable because the auxiliary building sumps are equipped with Category I level indicators which will quickly signal unusual water leakage. In addition, continuous on-shift operator surveillance in the auxiliary building will ensure that unusual leakage conditions are observed.

The auxiliary building floor drain system will be specifically reviewed to identify any areas where vertical risers could back-up and cause potential flooding in other areas. In cases where the potential for this situation is possible, check valves will be added in the vertical risers as necessary. The potential for room to room communication by way of the floor drain system on the same level will also be reviewed.

IV. PROCEDURE

The following steps are being followed in the flooding analysis:

1. Each floor of the auxiliary building will be zoned based on the location of walls, flow areas, and potential line break locations.

2. The limiting line break will be defined for each area.

3. The potential flow paths will be investigated to ensure that drainage to the auxiliary building sumps is available.

4. The transient break flow will be determined.

5. The maximum flood level in all areas involved will be determined.

6. The location of safety-related equipment will be compared with the maximum flooding heights.

V. CONCLUSION

The procedure described herein will verify that high and moderate energy line breaks will not cause flooding which impairs the ability to safely shut down the plant.

and non-seismic piping

QUESTION 010.50

"Your response to Q010.33 concerning the affects of flooding resulting from a failure of the circulating water system transport barrier is incomplete. You have not provided an adequate response to items (4) & (5) of Q010.33. Our concern is for the consequences of a major circulating water system leak in the turbine building caused by failures of such non-seismic Category I components as the main water headers or expansion joints to the condenser coupled with failures of their corresponding butterfly isolation valves. The potential exists to flood the turbine building basement to the water level elevation of the cooling tower basin (Byron) or the cooling pond (Braidwood) by simple gravity draining from these large reservoirs.

"Describe the designs and locations with the aid of drawings, if necessary, of the watertight barriers provided to prevent floodwater leakage from the turbine building to the auxiliary building or any other safety-related enclosure. Include a discussion of the consideration given to passageways, pipe chases and/or cableways joining the flooded space to space containing safety-related system components. As an example, discuss the means of preventing floodwater from entering the main steam tunnel and eventually reaching the auxiliary building at its termination with the main steam tunnel near the safety valve room. Include in the discussion water exiting the turbine building at or above grade level and entering other safety-related enclosures through watertight barriers removed for maintenance."

RESPONSE

In the event of a circulating water line break which cannot be isolated, the turbine building could theoretically be flooded to grade level at Byron and to 5 feet below grade at Braidwood. Once the water reaches the grade floor of the turbine building at Byron, it will flow outdoors and will not enter the auxiliary building except possibly for small amounts of seepage under closed doors. Damage to turbine building equipment will not prevent safe shutdown of the plant because no essential equipment is located in the turbine building.

The auxiliary building is completely watertight below grade at the turbine building auxiliary building interface except for the main steam tunnel from affecting the auxiliary feedwater tunnel, the containment, or any other auxiliary building areas.

The only safety-related items which will be affected by turbine building flooding are the main steam isolation valves (MSIV's). With a loss of power, the MSIV's will fail as is.

The steamlines are automatically isolated on high containment pressure or low steamline pressure signals. If the event which damages the circulating water piping also causes a significant break in a main steamline, the resulting decrease in pressure will cause MSIV closure prior to MSIV inoperability.

In the event that the turbine system remains intact following a circulating water piping failure, the turbine will be tripped and the turbine stop valves will isolate the steam system. Failure of the MSIV's will not have an adverse affect. The only lines which will not be automatically isolated on a turbine trip are the takeoffs to the gland sealing steam and steam jet air ejectors. The 4 inch gland steamline utilizes a motor-operated isolation valve which will be closed by the operator after a turbine trip. The line to the steam jet air ejectors contains a 2-inch manual isolation valve. Failure to close this valve will result in a blowdown from the main steamline of approximately 1000 lb/hr. One train of the auxiliary feedwater system is capable of supplying the steam jet air ejector and maintaining the plant in hot shutdown. Two trains of auxiliary feedwater are capable of supplying the steam jet air ejectors, gland sealing steam and maintaining the plant is hot shutdown in the event the motor-operated isolation valves for the gland sealing steam fails to close. There are no main steam or turbine valves associated with lines branching off the main steam header between the MSIV's and the turbine stop valves that are below grade except for the MSIV's. Valves in these lines are of a high quality as identified in the response to Question 040.143.

One fully severed circulating water pipe would provide sufficient flow into the turbine building to flood the MSIV's in about 10 minutes. This is not realistic because a guillotine break of this pipe would require an 8 foot lateral motion of the pipe. A large break in the circulating water system would quickly be evident to operating personnel and action would be taken to secure the main steam system. However, as discussed above, the only concern in this case is the possibility of gross failure of the main steam piping in conjunction with the circulating water pipe failure and this event results in MSIV closure prior to flooding. If the main steam system is intact, the MSIV's may fail open without impact on plant safety.

supplier has qualified the valves for mechanical and seismic loading by analysis, and has proven the operability of the valves through normal and emergency environmental conditions by actual test.

B.1.d: The containment isolation provision for the purge system lines are designed to Section III, Class 2, and Category IE electrical requirements. Inboard and outboard isolation valves (redundant valves) are supplied by Division 11 and 12 power respectively. Operators are of an air/spring design, fail the valve to the closed position upon loss of air or power, and are testable from the Control Room. The containment isolation provisions of the purge system therefore, meet all standards appropriate to Engineered Safety Features.

B.1.e: The purge system isolation valves close automatically on receipt of an ESF actuation signal. No external energy source is required to close the containment isolation valves. They are of a spring return design and will fail to the closed position upon loss of air pressure or electric power.

B.1.f: The specified maximum closure time for the containment purge isolation valves is 5 seconds.

B.1.g: The containment mini-flow purge exhaust intake is 8 inches in diameter, located 73 feet above the operating floor and approximately 2 feet 6 inches from the face of the containment wall. Due to this distance, it is unlikely that following an accident, any debris would blow as high as the mini-flow exhaust intake.

To ensure that debris or damaged ductwork does not impair the isolation of the miniflow system following a LOCA, a debris screen will be added to the miniflow supply duct. The debris screen and the duct between the screen and the isolation valve will be Seismic Category I and will be designed to survive transient differential pressure due to LOCA. The debris screen will be located 1 foot or less away from the isolation valve.

To L. KRIPPS
FROM J. LaVALLEE

QUESTION 022.30

"Penetrations P2, 3, 7, 9, 14, 15, 5, 6, 8, 10, 22, 25, and 48 have been listed in FSAR Table 6.2-58 as having met the requirements of GDC 57. However, these penetrations service systems inside containment that include components which are not Seismic Category I and/or not Safety Class 2. Therefore, in accordance with the provisions of SRP Section 6.2.4.11.9.c and d, the systems inside containment associated with these penetrations are not acceptable as closed systems and cannot be considered as one of the isolation barriers. Provide information demonstrating that the design provisions for these penetrations meet the requirement for two acceptable isolation barriers in series."

RESPONSE

Penetrations P-2 and P-3 have been sealed closed because the station heating is no longer utilized in the containment.

Penetrations P-7, P-9, P-14, and P-15 are the inlet and outlet penetrations for the essential service water system. These penetrations meet the isolation requirements of General Design Criterion 57 for closed systems. The essential service water penetration piping and isolation valves are Seismic Category I Safety Class 2. The essential service water cooling coils and associated piping inside containment are Seismic Category I Safety Class 3. Safety Class 3 design provides adequate assurance that the system will remain closed to reactor coolant and the containment atmosphere. For this system, there is no difference in allowable stresses or design and fabrication procedures between Safety Class 2 and Safety Class 3. The only substantial difference is that Safety Class 2 requires more detailed QA documentation and restricts the nondestructive testing to volumetric methods while Safety Class 3 will allow liquid penetrant or magnetic particle testing as alternatives. The applicant will reexamine the existing essential service water piping associated with the reactor containment for coolers (RCFC) using volumetric techniques, either radiographic or ultrasonic, to assure high quality welds in the portion of the piping system which is designated ASME Section III, Class 3. Based on this reexamination, the essential service water piping associated with the RCFC can be considered a closed sytem and no further isolation provisions are required for penetrations P-7, P-9, P-14, and P-15. The safety classification of the Reactor Containment Fan Coolers is discussed in the response to Question 022.73.

Penetrations P-5, P-6, P-8, and P-10 serve the chilled water system. This is a Category II Grade D system and, as such, is not considered a closed system. Penetrations P-6 and P-10 currently meet GDC 56 requirements with an automatic isolation valve outside containment and a check valve inside containment. Table 6.2-58

has been updated to include these valves. Penetrations P-5 and P-8 will meet GDC 56 requirements with the addition of a second automatic isolation valve inside containment. This additional valve will be either a fail closed hydraulically actuated valve or a motor operated valve powered from a different emergency power division than the existing isolation valve.

Penetrations P-22, P-25, and P-48 serve the component cooling system. This system has been upgraded. The piping, valves, and both the tube and shell side of the excess letdown heat exchangers are now Seismic Category I Safety Class B. FSAR Sections 3.2 and 9.3 will be updated to reflect this revision to the safety class of this equipment. As a closed system, these penetrations meet the requirements of GDC 57 and are listed as such in Table 6.2-58.

QUESTION 022.73

"FSAR Table 3.2-1 indicates that quality measures equivalent in intent to those in Quality Group C will be applied to the reactor containment fan coolers. It is our position that the reactor containment fan coolers must be designed, fabricated, erected and tested to Quality Group B standards, as recommended by Regulatory Guide 1.26 (SRP Section 6.2.2.11.6). Provide information on how you will comply with this position."

RESPONSE

The reactor containment fan cooler (RCFC's) coils are ASME Section III, Class 3 components.

The difference between Quality Group B and C (ASME Section III, Class 2 and Class 3) is in the type of nondestructive testing required. ASME Section III, Class 2 requires radiographic testing, ASME Section III, Class 3 nondestructive testing requiring magnetic particle, dye penetrant or radiographic testing. Allowable stresses, design and fabrication requirements for ASME Section III, Class 2 and Class 3 are the same.

The fan coolers have been designed to meet seismic and other safety-related Quality Group C requirements. In addition, the coils are functional during normal operating conditions and are redundant, such that only two of four coolers are required for post-accident heat removal.

The RCFC service water coils are designed to ASME Section III Class 3 requirements. Following is our justification to demonstrate that by performing magnetic particle examination on the fillet welds and radiographic examination on the butt welds these coils will meet Class 2 NDE requirements:

1. The RCFC coils are made of seamless copper tubes, formed and machined in one piece with end tube sheets.

2. The return bends are brazed to the tubes on one end. The NDE requirements are same for brazing processes for Class 2 and Class 3.

3. The water boxes are made in one piece with no joints and bolted to the tube sheet on the other end of the coil. Welded baffles in the box are internal to the box and hence are not in containment pressure boundary.

RESPONSE

The analysis of a main steamline break is presented in Subsection 15.1.5. The detailed description of main steamline isolation is in Chapter 10. Three lines branch off the main steam lines between the MSIV's and the turbine valves (ref. Figure 10.3-1).

A four inch line supplies approximately 16,000 lb/hr of steam to the gland steam system. A four inch motor-operated gate valve (GS001) is used for isolation. The valve is rated at 900 lbs. and is designed in accordance with ASTM standards.

This valve does not automatically close on a turbine trip but must be closed by the operator, if necessary.

A 28 inch line branches off each main steam header for the steam dump system and extraction to the second stage of the moisture separator reheater. The two 12 inch branch lines supply approximately 790,000 lb/hr. to the moisture separator reheater. Two 10 inch motor-operated gate valves (MS009A/B/C/D) on each moisture separator are used for isolation. The valves are rated at 900 lbs. and are designed per ASTM standards. The valves must be closed by the operator.

All of the valves are Category II, Quality Group D. All main steam and turbine valves that are in the main process flow paths are of a similar quality to the valves identified in Issue No. 1 of NUREG-0138.

The system sensitivity of the type loose parts monitoring system (LPMS) used for Byron/Braidwood Stations has been demonstrated and certified in tests at an operational nuclear power plant. Test results indicate that a sensitivity 0.1 ft-lbs within 3 feet of a sensor is within the capability of the monitoring system. At an impact level of 0.5 ft-lbs a signal to noise ratio of 6 can be maintained for normal plant operation conditions.

A seismic qualification program is currently being prepared for the Byron/Braidwood Stations LPMS. The qualification will demonstrate the capability of the LPMS to meet the seismic requirements of Regulatory Guide 1.133, Revision 1.

Subsequent to final preoperational tests and calibration, a comprehensive report will be prepared and submitted per Regulatory Guide 1.133, Revision 1, May 1981.

7. See item 6.

8. This is included in Item II.70.a.

9. This is included in Item II.64.b of Table 3.2-1.

10. This is already included in Table 3.2-1 as Item II.64.

11. These components are Category I. As previously noted, Table 3.2-1 illustrates the Safety Category and Quality Group breakdown for principle structures, systems and components only.

12. This is included in Item II.16.

13. These valves are Category I. As previously noted, Table 3.2-1 illustrates the Safety Category and Quality Group breakdown for principle structures, systems and components only.

14. These components are Category I. As previously noted, Table 3.2-1 illustrates the Safety Category and Quality Group breakdown for principle structures, systems and components only.

15. These valves are Category I. As noted in Section 3.2.1.1, the safety category of piping and valves may be found on the system P&ID's. As previously noted, Table 3.2-1 illustrates the Safety Category and Quality Group breakdown for principle structures, systems and components.

16. These valves are Category I. As noted in Section 3.2.1.1, the safety category of piping and valves may be found on the system P&ID's. As previously noted, Table 3.2-1 illustrated the Safety Category and Quality Group breakdown for principle structures, systems and components only.

17. This piping and valves are Category I. This is noted on the system P&ID's. Table 3.2-1 is intended to list principle structures, systems and components only.

18. As noted in Table 3.2-1, Item II-11, the Condensate System is Safety Category II. However, pertinent requirements of the operational QA program will be applied to this system. As shown in Item II.3, the Auxiliary Feed Water System is Category I.

19. No air operated valves perform a safety function on the Byron/Braidwood stations.

26. The cask is handled by the Fuel Handling Building crane which is listed in Table 3.2-1 (Item II-31e).

27. See item 23.

28. See item 23.

29. As noted in Section 3.2.1.1, the safety category of piping and valves may be found on the system P&ID's. As previously noted, Table 3.2-1 illustrates the Safety Category and Quality Group breakdown for principal structures, systems and components only.

30. See Item 1.

31. The safety category for supports is the same as the pipe or equipment being supported.

32. See Item 29.

33. a. H_2 monitoring system will be added to Table 3.2.1. This is a Category I system.
    b. H_2 analyzer will be added to Table 3.2-1. This is a Category I system.
    c. See Item 31.

34. a. As shown in Table 3.2-1, Item II-98n, the RCFC Essential Service Water Coils are Category I.

    b. As discussed in response to Item 31, the dampers and supports necessary for operation of Category I systems are designed as Category I.

35. a. The hydrogen recombiners listed in Table 3.2-1, Item II-53a.

    b. Not applicable.

    c. See Item 29.

36. See Item 31.

37. This equipment is already covered by Item II.19.a of Table 3.2-1.

38. This equipment is covered under Item II.20 of Table 3.2-1.

39. Item II-44a includes all equipment necessary for Category I Instruments and Control to perform their safety function.

40. Although not all components in the Control Room HVAC system were designed as Category I components, the pertinent provisions of the operational phase QA program will be applied to this entire system. Control room HVAC equipment is treated as follows:

    h. Humidifier will be added to Table 3.2-1.

    i. The Condenser is included in Table 3.2-1, Item II-88.d.

    j. Charcoal Filter housing will be added to Table 3.2-1.

    k. Ductwork and dampers are not included in this table. The safety category may be found from the P&ID's or equipment lists as noted in Section 3.2.1.1.

    l. The control room HVAC does not have safety isolation valves.

    m. The utility exhaust fan will be added to Table 3.2-1.

    n. Electrical modules with safety functions are included in Table 3.2-1, Item II-88f.

    o. Cables with safety functions are included in Table 3.2-1, Item II-88f.

41. See Item 29.

42. The fixed equipment is covered under Item II.6 of Table 3.2-1. Although not all components of this system were designed as Category I components, pertinent requirements of the operational phase QA program will be applied to this system. Portable equipment is purchased under site QA and thus is not included in Table 3.2-1. Pertinent requirements of the operational phase QA program will be applied to this equipment.

43. The fixed equipment is covered under Item II.61 of Table 3.2-1. Although not all components of this system were designed as Category I components, pertinent requirements of the operational phase QA program will be applied to this system. Portable equipment is purchased under site QA and thus is not included in Table 3.2-1. Pertinent requirements of the operational phase QA program will be applied to this equipment.

44. This equipment is covered under Item II.61 and 62 of Table 3.2-1. Although not all components of this system were designed as Category I components, pertinent requirements of the operational phase QA program will be applied to this system.

45. This item does not constitute principle structures, systems, or components of the plant, and thus is not included in Table 3.2-1. This activity will be conducted under the pertinent requirements of the operational QA program.

46. See Item 45.

47. See Item 45.

48. See Item 45.

49. See Item 45.

50. See Item 45.

51. See Item 45.

52. This is included in Item II.74.a of Table 3.2-1.

B. NUREG-0737 ITEMS

NUREG-0737 Items are addressed in Appendix E. As discussed previously, Table 3.2-1 gives general information on the safety classification of equipment corrections required as a result of NUREG-0737 items have been made to Table 3.2-1 but not all NUREG-0737 items correspond to Table 3.2-1 items. The items listed have been investigated and are discussed here.

1. The plant safety-parameter display console is being installed at Byron/Braidwood stations. (See Section E17) This is included in Table 3.2-1, Item II.60.a.

2. The Reactor Coolant System Vents are discussed in Section E.19. This is included in Table 3.2-1, Item II.64.a.

3. NUREG-0737, Item II.B.2, Plant Shielding, is a design review of the plant shielding and postaccident. radiation levels and therefore, does not directly involve plant structures or equipment. As discussed in response to Part A.1 of this question, safety category of shielding is equivalent to that of the building in which it is located.

4. The post accident sampling capabilities are discussed in Section E.21. This system is included in Table 3.2-1, Item II.62, and is a Category I system.

5. The indication of relief and safety valve position is explained and the safety category requirements discussed in detail in Section E.24. The instrumentation is the same safety category as the valves it is provided for, i.e., Category I.

6. The auxiliary feedwater system is discussed in Section E.25 and included in Table 3.2-1, Item II.3.

and recovery from natural circulation mode. Operators should be able to recognize when natural circulation has stabilized, and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.

If these tests have been performed at a comparable prototype plant, they need be repeated only to the extent necessary to accomplish the above training objectives.

5.w. Containment penetration cooling system. On those penetrations where coolers are not used, provide a startup test description that will demonstrate that concrete temperatures surrounding hot penetrations do not exceed design limits.

RESPONSE

(1.n.7) Fire protection testing is conducted concurrently with ventilation pre-operational tests as delineated in Tables 14.2-35, 14.2-36, 14.2-37, 14.2-38, and 14.2-39.

Fire protection equipments and components not identified in the pre-oprational tests are conducted in system demonstration tests.

(4.t) The referenced letter (NS-EPR-2465), from E. F. Rake (W) to H. R. Denton, dated 7-8-81 provides the reactor manufacturer's recommended program for natural circulation related testing. The program includes four tests, one of which is item 4.t.

The four recommended tests will be conducted during the Byron preoperational and startup test programs as modified to incorporate plant specific design features (viz the Byron design incorporates a diesel driven AF pump not a steam driven pump).

(4.w) Table 14.2-16 has been revised to include testing of concrete penetrations cooled by component cooling and testing of non-cooled penetrations.

(5.1) Later

QUESTION 423.35

"The response to Item 423.14 is inadequate. Modify the initial test program to provide a description of the inspections or tests that will be performed following system operation to assure that all snubbers are operable."

RESPONSE

Refer to revised Table 14.2-58. Responses to Questions 110.63 and 110.37 provide a summary of the snubber operability and thermal monitoring program.

QUESTION 423.38

"The initial test program should verify the capability of the offsite power system to serve as a source of power to the emergency buses. Tests should demonstrate the capability of each starting transformer to supply power (as the alternate supply) to its unit's emergency buses while carrying its maximum load of plant auxiliaries and the other unit's emergency buses (as preferred supply). Tests should also demonstrate the transfer capabilities of the unit's emergency bus feeders upon loss of one source of offsite power. These tests should be performed as early in the test program as the availability of necessary components allows. Provide descriptions of the tests that will demonstrate these capabilities."

RESPONSE

The initial test program will verify the capacity of the system auxiliary transformer (starting transformer) to supply power to its unit's vital buses while carrying its maximum load of plant auxiliaries and the other unit's vital buses. Tests will be conducted to demonstrate transfer capabilities. These tests will be performed as early as the necessary components become available for testing, but not during the period when electrical separation requirements are in effect for Unit 1 operation and Unit 2 construction. The testing is delineated in revised Table 14.2-11.

QUESTION 423.39

"The test descriptions are not sufficiently detailed to ascertain if the voltage levels at the safety-related buses are optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Provide a description of the method for making this verification."

RESPONSE

The Auxiliary Power System Test Abstract, Table 14.2-11 has been revised to include voltage optimization testing vital buses.