RANDOLPH 6-1200

Commonwealth Edison Company

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September 9, 1966

United States
  Atomic Energy Commission
Washington, D. C. 20025

Attention: Division of Reactor Licensing

Amendment No. 2 to Application for Construction
Permit and Operating License - AEC Dkt. 50-254

In response to your letter of August 24, 1966, requesting additional information to support the Application for Construction permit for Units 1 and 2 at Quad-Cities Station near Cordova, Illinois, Amendment No. 2 to the Plant Design Analysis is submitted herewith.

COMMONWEALTH EDISON COMPANY

By Murray Joslin
Vice President

Subscribed and sworn to
before me this 9th day
of September, 1966.

Notary Public

Photocopy sent to Important Papers

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INTRODUCTION

The information provided herein is in response to the letter from R.L. Doan, Director, Division of Reactor Licensing, dated August 24, 1966. This submittal contains the additional technical information as requested. In addition, several changes and additions are proposed to various sections of the Quad Cities Unit 1 and 2 Plant Design Analysis. For reference, these changes and additions include the following:

(1) Primary System Relief Valves - Section IV-3.3.
Changes are proposed to permit automatic actuation of the primary system relief valves.

(2) Primary System Containment - Section V-1.1
The design basis of the primary containment system is clarified to indicate ability to flood the containment and is modified to delete references to containment capability assoqiated with 100% core melt.

(3) Containment Spray Cooling System - Section V-1.2.7
Modification of system to be comparable with low pressure coolant injection system.

(4) Core Spray System - Section VI-6.0
Changes in design capabilities of the system.

(5) Low Pressure Coolant Injection System - Section VI-7.0
Describes new system added to accomplish core flooding subsequent to coolant loss accident.

(6) High Pressure Coolant Injection System - Section VI-9.0
Describes new system added to accomplish core cooling associated with loss of coolant arising from small line breaks.

(7) Standby Coolant Supply System - Section VI-10.0
Describes new system to provide continuous feedwater capability under emergency conditions. Figure 65 has been revised to indicate the addition.

(8) Standby Diesel Generator System - Section VIII-3.0
Describes additional diesel generator capacity requirements.

(9) RCIC System - Section X-4.0
Describes system change wherein one RCIC system is provided in lieu of two systems. Figure 101 has been revised to reflect the change.

QUESTION

A. General

1. In view of the public water supply intake approximately 20 miles downstream of this facility and the fish and wildlife preserve opposite the plant on the Iowa side of the Mississippi, please elaborate on the precautions to be taken to minimize the expected release of radioactivity for both normal and off-normal operation. In your discussion please supply the information on which the estimates of the expected release quantities of radioactive waste are based. Provide further description of the environmental monitoring systems to be employed.

ANSWER

Minimizing the Amount of Release

There are many precautions that will be taken to minimize the release of radioactive materials into the discharge canal. These precautions apply for both normal and off-normal operating conditions. Each separate batch of liquid to be discharged will be analyzed prior to discharge. The rate at which the batch is released into the discharge canal will be dependent on the coolant flow that is available for dilution.

Continuous sampling of the discharge canal will assure that the daily release does not exceed operating limits. The environs monitoring program will continuously sample the river water downstream of the discharge.

An item that is not an operating precaution but serves an important part in minimizing the release of radioactivity is the Powdex condensate demineralizer system. With a Powdex system there will be no liquid discharge of solutions from the regeneration of resins. Since the solutions from the regeneration process usually comprise the major portion of the liquid wastes, use of a Powdex system will reduce the radioactivity discharge by roughly a factor of 20 as compared to a regenerative system. The spent Powdex resins are shipped from the plant as solids.

Environs Monitoring System

A detailed description of the Dresden Station Environs Monitoring Program has been prepared and submitted in AEC Docket 50-10 by letter of John H. Hughes to Dr. R. L. Doan dated August 11, 1966. The environs monitoring systems planned for installation on and off the Quad-Cities Station site will be quite similar, if not identical, to those currently in operation at Dresden Station.

Onsite Monitoring

On-site environs monitoring will include continuous detection and recording of atmospheric gamma radiation, weekly integrated gamma radiation exposure measurements, and the collection of airborne particulate samples. Plans also provide that these measurements will be made at four locations within the station property boundaries.

The inlet and discharge canals of the cooling water will be continuously sampled to maintain a daily measurement of the radioactivity released to the Mississippi River via the Radioactive Liquid Control System.

Off Site Monitoring

The exact number of offsite sampling locations are undetermined at this time, however, the types of samples to be collected will include air particulate, atmospheric gamma radiation, surface water, well water, fall-out (rain) water, silt, slime, and plankton, vegetation and milk.

Collection frequencies will vary - weekly in the case of air particulate and gamma radiation, monthly for surface waters, and semiannually for well water.

A continuous water sampler will be installed at a suitable location approximately 10 miles downstream of the station's discharge canal.

QUESTION

A. <u>General (Continued)</u>

2. A failure of a turbine rotor could result in a spectrum of missiles of considerable velocity and mass. Please discuss the potential hazards to your facility from missiles generated by this type of failure. In particular:

   a. What velocities, masses and trajectories of missiles are possible from postulated turbine rotor or blade failures with the present turbine orientation? Discuss reorientation of the turbine to reduce the probability of damage to the reactor containment system by turbine-generated missiles.

   b. What energy missile would damage 1) the secondary containment, 2) the primary containment and 3) components vital to a safe shutdown of the plant (including electrical and service water systems) which are exposed to potential missile trajectories?

   c. Discuss the likelihood of turbine rotor failure with respect to the design, selection of materials, quality control during construction, and previous history of turbine failures. At what overspeed would rotor failure be expected to occur?

ANSWER

Since the end of 1954, when the low pressure spindle of the turbine at Commonwealth Edison's Ridgeland Station failed there have been no major turbine rotor failures in the United States or Canada. Following the Ridgeland failure, investigating committees were set up by ASME and ASTM comprised of members from each of the major turbine manufacturers and forging suppliers, which led to the development of production techniques and non-destructive testing procedures to insure a forging quality which made such failures virtually impossible. Since the Ridgeland incident there have been some minor pieces, i.e., blade sections, lashing rings, etc. which failed, but in no instance did any of these parts leave the turbine casing.

The possibility of major failure of one of the spindles in the turbine generator for Quad-Cities has been analyzed in terms of the above question.

Based upon destructive tests at the turbine manufacturers shop, together with a reconstruction of the mode of failure of the Ridgeland rotor, a model of the mode of failure has been established, and repeatedly reproduced by test. If the rotor consists of a central shaft with a relatively large diameter heavy section near the center (as in the case of the Quad-City high pressure turbine rotor) the most probable fracture mode would consist of a quartering of the large center pieces. Such fracture would involve a number of small fragments originating near the center of the shaft, after the large quarter sections split.

The Quad-Cities high pressure turbine spindle is approximately 22'-8" long between bearings. The large diameter center sections are 64-1/4" O.D. x approximately 5'6" wide. These heavy center sections will be machined to hold the high pressure turbine blades. The estimated weight of the entire high pressure rotor is 163,000 lbs. If the center section were postulated to fracture according to the above model, a piece weighing approximately 30,000 lbs could be produced as a missile.

The low pressure turbine spindles are constructed such that separate blade rings are shrunk on to a long cylindrical shaft, rather than machining the blade rings into the forging itself. The shaft for the low pressure sections is approximately 22' long between bearing center lines and the shaft is approximately 32" O.D. Failure of this kind of spindle has not been experienced and, since the blade rings are much smaller separate pieces, fracture of one of these is not considered to be particularly hazardous. Furthermore, major failure of this type of fabrication has not been experienced. In any event, the low pressure rotors are not aligned with the primary containment system and consequently do not pose a hazard in this regard.

The generator rotor is encased in the heavy stator, and any failure of the rotor would not penetrate the casing.

The normal rotating speed of the turbine is 1800 rpm. The overspeed set point, which would cause turbine trip, will be at 8 to 9%. The total calculated rotational energy contained in the high pressure spindle rotating at 1800 rpm, is approximately 2.5×10^8 ft.-lbs. If approximately one-fourth of this energy is available for each 1/4 section in a postulated fracture, each piece would have an energy of 6.25×10^7 ft.-lbs available to it. If no energy were consumed in the fracture and in passing through the casing, the theoretical maximum separation velocity corresponds to about 350 ft/sec.

The potential trajectory of any missiles produced by failure of the turbine rotor or blades consist of the following:

a) The piece leaving its point of origin would, assuming sudden release, be tangential to the rotating element. The probability of direction is equal throughout 360°.

b) Pieces leaving in a direction horizontally or in the downward 180° included angle would follow approximately a straight line path unless deflected by objects encountered in the flight path.

c) Missile pieces leaving in the upward 180° included direction would follow a parabolic path unless otherwise deflected.

d) A piece going straight up would, of course, follow a straight line path unless deflected.

For each potential target point, at the same elevation as the turbine shaft, there are two angular directions that the missile could leave its point of origin. At 45° there is only one target point and this would correspond to the maximum range.

Specifically, the turbine generator center line at Quad-Cities will be located at approximately elevation 639'-0". The refueling floor level of the reactor building will be at elevation 690'-6" and the center line of the turbine generator and the center line of the reactor vessel are separated by approximately 125 ft. (c.f. figure 33, PDAR Amendment 1).

If a potential missile originating at the turbine rotor were released, the following directions can be considered of no consequence:

a) Any direction in the downward 180° included angle, since the approximately 12 ft thick turbine foundation shields the plant areas sideways and approximately down to 60° below horizontal. In the downward directions, the pieces would penetrate the consenser and end up probably on the basement floor. The turbine foundation further serves to shield the main steam and feedwater isolation valves at the primary containment penetrations. A potential missile could rupture a steam or feedwater line, but only on the turbine side of the isolation valves.

b) All directions above the turbine main floor and to the west of the plant, including the straight up direction, since these are directed away from the reactor building.

c) All directions toward the reactor building up to approximately 35° above horizontal. Missiles travelling in this direction would strike the concrete wall separating the reactor and turbine building. This wall varies in thickness from 4 ft. to 2 ft. depending upon shielding requirements, etc. Beyond this wall is the 6'-6" thick concrete cylinder surrounding the primary containment vessel. Because of the double barrier, it is believed that no potential missile could penetrate these two barriers. In addition some non-critical equipment is located in the intervening space which would serve to intercept the missiles.

This leaves the included angle from about 35° above the horizontal to 90°; i.e., straight up, toward the reactor building as an area for milssiles to travel.and land on the refueling floor of the reactor building. It is impossible to estimate the energy of the pieces consumed in either the rotor fracture or by passing through the heavy steel double casing of the high pressure turbine. Furthermore, some energy is consumed by a potential missile traveling through the turbine building roof, reactor building superstructure, etc. It appears that there exists an area approximately 5° in included angle toward the reactor building which could result in a missile traveling in a direction which would land the pieces on the 40 ft. diameter shield plug over the primary containment vessel at the refueling floor. This plug is approximately 6 ft thick and is in place at all times during plant operation.

It is calculated that this plug could accommodate the force imposed by a 30,000 lb piece dropped from a 60 ft height. This would correspond to a height of approximately 110 ft. above the turbine room floor. The calculations developed for this case are based on fracturing of the concrete plug and allowing the reinforcing steel to go to the yield point. No allowance has been made for energy consumed in the plug by elastic or plastic deformation which would take the reinforcing steel beyond the yield point, and which would still permit the plug to remain intact. In this event, it is likely that pieces of concrete would spall from the bottom of the plug and strike the top of the drywell. These pieces would not penetrate the steel cap of the drywell.

Based on the above, the shield plug could safely absorb 1,800,000 ft-lbs with no potential hazard to the primary containment. Undoubtedly, the shield plug could absorb considerably more than this without damaging the primary containment, but it is difficult to assess what this would be. It is believed that the plug would first deform until it rested on the drywell cover. The cover would then bend and would probably deform until it contacted the reactor pressure vessel head. It is likely that the drywell would not tear prior to contact with the reactor vessel. At this point the movement would cease since the vessel is supported on concrete which in turn would transmit further downward forces into the building substructure.

If a missile were generated, it is likely that even small ones could damage the secondary containment, since the superstructure is an insulated metal panel whose thickness is determined by resistance to weather and service conditions. The primary containment, however, is surrounded by at least 6'-6" of concrete plus an additional concrete building which tends to protect it. We, therefore, believe that no potential missile originating at the turbine would penetrate the primary containment from the side and could only damage the primary containment if a massive rotor piece were directed upward in a very specific direction permitting it to come to rest on the shield plug covering the drywell at the refueling floor. Even in this case it would have to be dropped from a distance in excess of 60 ft. above the reactor refueling floor elevation.

There are no components associated with safe shutdown of the plant that could be damaged by potential missiles originating at the turbine.

The turbine generator shaft is oriented parallel to and adjacent to the reactor building. This orientation is selected to permit the most convenient and economical arrangement of the overall plant, and allows the use of a single turbine room crane and other service facilities for both units. Circulating water for the condensers can be ducted to and from the condensers with this arrangement only for multiple unit stations.

At the time that the Ridgeland spindle forging was ordered (1951) such techniques as sonic testing were just being developed. The following summarizes the inspection procedures employed at the time that forging was processed.

a) The center bore was optically inspected.

b) The fillets, where the large diameter segments intersect the main shaft, were magnetic particle inspected.

c) Sonic tests were made, but showed so many inclusions and defects that the equipment was judged to be defective.

d) Sulphur prints were taken on the main forging body.

The failure of the rotor at Ridgeland was finally attributed to hydrogen flake inclusions which developed high stress concentrations which propagated to the surface. In addition, it was later found that the turbine spindle NDT temperature was somewhat higher than its operating temperature and consequently the forging was notch sensitive. This led to brittle fracture at the time of failure.

The Quad-Cities rotor forgings will be subject to the most modern day quality controls and manufacturing techniques possible. In summary these consist of:

a) Vacuum degassing the forging to eliminate hydrogen.

b) The NDT transition temperature will be below ambient.

c) The forging is ultrasonically and heat induction tested at the mill. It is again ultrasonically tested at the factory. After machining, the surface will be magnetic particle inspected. The forging will be again heated after machining to examine for thermal stability. The forging is balanced, tuned and run at 10% overspeed at the factory.

d) The normal design stress levels in forging correspond to 120% overspeed and GE uses slightly lower stress levels for their nuclear turbine than conventional turbines.

As indicated above, since the Ridgeland failure in December, 1954, no major turbine rotor failures causing pieces to penetrate the turbine casing have been experienced in the United States or Canada by any turbine manufacturer. Prior to the Ridgeland experience the following occurred:

a) March, 1954. A 147,000 KVA, 3600 rpm, generator spindle burst at 3400 rpm at the factory during balancing. Cracks and inclusions in the forging led to high stress concentrations which led to brittle fracture of the forging.

b) September, 1954. A generator rotor forging failed at the Cromby Station of the Philadelphia Electric Co. on overspeed test at 3780 rpm (normal speed was 3600 rpm in this 216,000 KVA machine). Failure occurred at a row of holes that had been drilled for repair bolts. These were required because of trouble encountered when milling slots in the forging. This technique is no longer employed by turbine manufacturers.

c) January, 1953. A 125,000 KVA, 1800 rpm spindle failed in service at the Tanners Creek station of the Indiana and Michigan Electric Co. A segment of the first stage wheel failed due to improper stress relief during manufacture.

d) The last and most dramatic failure was that at Ridgeland Station of Commonwealth Edison on December 19, 1954. The low pressure spindle failed on overspeed test when the unit was being tested for startup. The rotor forging fractured, in accordance with the model described above and pieces were found as far as 500 ft. from the turbine. The large fragments travelled radially perpendicular to the shaft while many smaller pieces were scattered due to striking the turbine casing and the building. As mentioned above, since these failures and especially the Ridgeland failure, modern quality control and material procurement techniques insure that repetition of these events is virtually impossible. Forgings are no longer notch sensitive nor are they likely to have inclusions leading to high concentrations of local stresses because of rigid quality control requirements.

Each of the above were fully documented in the proceedings of the 1955 ASME Annual Meeting and are listed in the references.

It is difficult to assess the overspeed point which would lead to rotor failure using a modern turbine. Based upon test results at the GE factory it is estimated that a rotor forging such as that to be used at Quad-Cities would not fail at speeds well in excess of 20% overspeed.

REFERENCES

1. ASME Paper No. 55-A-208, "Report of the investigation of two generator rotor fractures" by C. Shabtach, E. L. Fogleman, A. W. Rankin, and D. H. Winne, of the Large Steam Turbine Generator Department, General Electric Co.

2. ASME Paper No. 55-A-210. "Report of the investigation of the turbine wheel fracture at Tanners Creek" by A. W. Rankin and B. R. Seguin, also of the Large Steam Turbine Generator Department, General Electric Co.

3. ASME Paper No. 55-A-172, "Investigation of large steam-turbine spindle failure" by H. D. Emmert, Asst. Chief Engineer, Steam Turbine Department, Allis-Chalmers Mfg. Co.

QUESTION

A. General (Continued)

3. Please provide a description of any tests performed to show that the control rod drive collet fingers have sufficient strength to prevent the ejection of a control rod in the event of thimble failure including the case in which the collet fingers are unlatched at the time of the accident.

ANSWER

Tests have been conducted which demonstrated the ability of the collet assembly to engage and lock the index tube at the maximum velocity expected during failure conditions. These tests were conducted on collet assembly 104B1319, the production assembly used in control rod drive 7R-DB-144-A1. In these tests, the collet was allowed to engage a section of the index tube weighted to 310 lbs. at various velocities up to 15 ft/sec, corresponding to an impact energy of 13,000 inch-lbs. The calculated maximum velocity under failure conditions is 10 ft/sec (refer to Report G.E. GECR-5089)transmitted previously to the AEC by the General Electric Co. The index tube travelled less than one inch after finger engagement.

The tests disclosed that the collet fingers engaged the notches at the maximum speed and remained engaged although the fingers were bent and the retaining cylinder was crushed. As predicted, the fingers bent inward tending to jam the index tube. The walls of the retaining cylinder also deflected inward, further restraining the index tube. Static tests run on the collet subsequent to the impact test demonstrated that it was still capable of supporting the maximum applied load in the damaged condition.

These tests and the results produced are described in Report GECR-5268, which is currently being prepared, and will be transmitted to the AEC when completed.

QUESTION

A. <u>General (Continued)</u>

4. Discuss potential hazards to the facility which might result from chemicals manufactured or stored at or shipped to adjacent industrial plants. Include the effects of accidental upstream dumping of corrosive chemicals and fire and explosion hazards in the discussion.

ANSWER

4. The potential hazards to the Quad-Cities Station from chemicals manufactured or stored at or shipped to (or from) adjacent industrial plants would not be of sufficient magnitude to adversely affect the functional capability of the primary containment, reactor and turbine buildings or radioactive liquid waste control facility.

The manufacturing process and the nature of the finished ammonium nitrate product are such as to preclude any hazards of explosive nature. The product is coated with an inorganic anti-setting agent which differentiates it from that of explosive-type ammonium nitrate. A noteworthy process feature is the detection of any trace of oil. The sensitivity of the detection method is capable of detection of trace quantities of oil and assures the maintenance of these oil concentrations well below 1% (trace concentrations). (Ammonium nitrate-oil mixture of about 6% by weight may become explosive if brought into intimate contact with organic materials and a detonator.)

The actual quantities of finished products stored and shipped from the plant are a function of commercial demands. Although a maximum of 400 railroad cars can be stored on the plant siding, only a total of about 25 cars are actually located there at any given time. This quantity represents an aggregate total of about 1000 tons of ammonium nitrate solids (prills) and nitrogen fertilizer solutions. Under no circumstances are any petroleum or other organic fluids stored on these tracks, thus precluding any interactions between ammonium nitrate and organic materials in the event of a fire.

No organic materials are received at the plant by rail shipment. All shipments of ammonium nitrate from the facility are transferred by rail or truck. No shipments are made by barge. The minimum distance separating the Quad-Cities Station structures from the Chicago, Milwaukee, St. Paul and Pacific Railroad and adjacent Route 84 highway is approximately 4000 feet. In the highly unlikely event a chemical explosion should occur at this closest point, no significant damage would be sustained by the plant structures which would render it incapable of a safe shutdown. Since the safety record compiled for the manufacture, handling and shipping of nitrate products is excellent, there is a minimal amount of experience upon which to base estimates of such damage; however, as a result of the review of available information, it is assumed for the purpose of the foregoing conclusion that the range of damage would be a radius of approximately one half (1/2) mile.

Review of the commodities transported by barge on the Mississippi River past the Quad-Cities Station approximates 9 million tons annually; upstream traffic consists of 40% coal, 30% petroleum and related products and 30% comprised of a miscellany of sulphur, salt, gravel and sand, and very minor quantities of anhydrous ammonia to list just a few. However, none of the cargos are considered to be "hazardous" in any special sense by the agencies having jurisdiction over barge traffic. Downstream traffic is approximately 30% grain and 70% empty barges. This barge traffic occurs only during the 9-month period when the channel is open for navigation.

The near- and long-term effects upon the plant systems which have contact with or utilize river water for cooling purposes by the accidental upstream dumping of corrosive chemicals is judged to be minor if not negligible. The Mississippi River at the low flow rate* of 18,900 cfs represents a water volume of about 8 million gallons per minute passing the plant intake canal. Plant operating experience has shown that condenser tubes, during acid cleaning procedures, are exposed to concentrations of hydrochloric acid of 5% by weight at elevated temperatures (140°F) for extended periods (6 to 14 hours) without damage or significant reduction in tube wall thickness.

If, for the purpose of this discussion, it is assumed that an accidental release (dumping) of acid (HCL) occurred upstream

* minimum average monthly

in such a manner as to preclude a significant degree (full) of mixing, i.e. only 10% of the total river flow was thus contaminated to 5% by weight with the acid (at low river flow), and that this 10% portion of the stream was maintained in tact (channeling effect) until drawn into the plant intake canal, it would be necessary that the release (or dumping) rate equal about 60,000 gpm of concentrated acid. No quantities of corrosive chemicals or the facilities to produce such quantities at this rate are known to exist upstream of the Quad-Cities Station. Furthermore, any stream flow rate in excess of this low flow would further reduce the concentrations present by additional dilution. Finally, the periodic chlorination of process cooling systems which is commonly practiced to improve performance of these systems is not too different from this situation. Chlorination has no significant deleterious effects upon condenser tubes (especially stainless steel to be used at Quad-Cities Station) or other process systems but rather is practiced as a process improvement procedure.

Since it becomes obvious that a 5% concentration in the river could not be sustained for a long period of time (hours or days) simply by the lack of available chemicals or by the fact that loss of quantities of such magnitude could not go unobserved indefinitely, it appears that if a short-term exposure could occur, it would have only a negligible detrimental effect upon the plant system.

In the event of fire occurring on or near the plant site as a result of chemical reactions, the plant has the fire-fighting capability of protecting personnel, equipment and structures by automatic initiation of water deluge and/or carbon dioxide extinguishing systems, and trained fire brigades. An adequate supply of water for fire protection purposes is available in all areas adjacent to the plant structures and switchyards and is maintained at pressure by both electrically powered and diesel driven pumps.

QUESTION

B. Site

1. In view of the relatively high frequency of tornados in the mid-west, please provide the following information:

a. Provide specific tornado design criteria for the Quad-Cities plant with respect to wind loadings and missile protection for the containment, control room, radioactive waste storage tanks, components or systems required for a safe shutdown and primary steam system. Provide justification for the winds and missiles chosen as a basis for design protection.

b. Discuss the location and specific design protection from wind loadings and missiles of the type generated by a tornado provided for the above systems and components.

c. Define the ultimate capability of the above systems and components to withstand wind loadings and missiles of the type generated by a tornado.

d. What effect might a tornado have during a refueling operation? What is the history of tornado effects on small bodies of water? Could fuel in the fuel storage pool be disturbed by a tornado?

e. Provide the design considerations given to the lower compartments of the secondary containment (by structural strength or venting) to withstand the effects of a tornado-created vacuum in the upper reactor building?

ANSWER

a. As stated in Section I-2.7.3 of the Plant Design and Analysis Report the structures are designed to assure that safe shutdown of the reactor can be achieved considering the effects of possible damage to the structures when subjected to the forces of short-term tornado loadings up to 300 mph. This design basis is interpreted to mean that the reactor primary system as described in Section IV 2.0 of the Plant Design and Analysis Report, and components which directly affect the ultimate safe shutdown of the plant are located either under the protection of reinforced concrete or are located underground.

The technical literature provides little information on the wind velocities within a tornado, and the selection of 300 mph as a design value is arbitrary. The U.S. Weather Bureau has issued a publication(1) which states:

"No structural damage is known to have resulted to a reinforced concrete building in a tornado".

A most important design and safeguards consideration is the question of what damage level to evaluate. Recognition is given to the fact that superstructure damage could be incurred to the reactor building, turbine building, storage tanks, and incoming power lines without affecting the ability to shutdown the reactor and to maintain integrity of containment and certain heat removal systems during and following a tornado which might traverse the site. Simultaneous damage to all of these items is not expected, yet as a design objective the power plant may be safely shutdown and maintained in a safe shutdown condition with the loss of all such superficial equipment.

Those items of equipment or systems located either underground or within the protection of the reinforced concrete include the following:

Primary Containment and Isolation Valves
Reactor Primary System
Shutdown Heat Exchangers
Control Rod Drive Hydraulic Equipment
Standby Liquid Control System
Reactor Core Isolation Cooling System
High Pressure Coolant Injection System
Low Pressure Coolant Injection System
Core Spray Systems
Shutdown-Containment Spray System
Service Water System
Station Battery
Diesel Generator

Electrical Controls and Instrumentation for Above Systems Control Room

Although the design basis wind is arbitrarily established at 300 mph, the actual design of the concrete structures is governed by the seismic considerations. Therefore, the 18-inch thick concrete panels of the reactor building are capable of resisting winds in excess of 400 mph within normal allowable stresses and in excess of 600 mph within yield.

[illegible] 13"

The capability of the concrete building panels to resist missiles has been inventigated. A utility pole (35' long x 14" diam.) striking the center of an 18-inch thick panel at 150 mph will not penetrate. It will cause local concrete cracking. Above this velocity, progressively greater yielding of the reinforcing bars will occur, penetration could occur at a velocity of about 200 mph. It should be noted that it is difficult to postulate the vectorial forces necessary to accelerate this missile to the velocities discussed.

A one-ton missile, such as a compact-type automobile traveling at 100 mph will not penetrate the concrete, with no credit being taken for energy absorption of crushing the vehicle. However, at speeds of 150 mph, portions of the vehicle (engine) may be capable of penetration.

Therefore, the station has been designed with sufficient capability to maintain a safe shutdown condition for prolonged periods.

REFERENCES

(1) Gilbertson, V.C. and Magenau, E.F., "Tornadoes," AIA Technical Reference Guide, TRG 13-2, U.S. Weather Bureau.

b. The reactor building walls have been designed to accommodate tornado winds of 300 MPH. The walls which provide the secondary containment are comprised of 1'-6" and 3'-0" thick reinforced concrete walls. Additional protection for the primary containment is achieved by the 6'-6" thick concrete shielding walls surrounding the drywell. The wind design, however, does not exceed the earthquake design requirements, therefore the earthquake design governs. Since the earthquake design governs, the wall can accommodate tornado winds up to 500 MPH without exceeding normal stresses for 4000# per sq. in. concrete. All equipment necessary for the safe shutdown of this plant is housed within these walls. The walls were also analyzed for the effect of missiles generated by a tornado. The capability of the wall to withstand these loading conditions is described in section B. 1. c.

c. ~~The ultimate capability of the reactor~~ building ~~walls to withstand a tornado wind is~~ based on allowing the reinforcing steel to approach yield stress. At this point, the walls can accommodate a wind velocity exceeding 500 MPH.

We have considered two types of missiles that could be generated by a tornado. (1) A utility pole 35'-0" long with a butt diameter of 13" and a unit weight of 50#/ft.^{3} having a velocity of 150 MPH. (2) A 1-ton mass with a velocity of 100 MPH with a contact area of 25 sq. ft.

The walls were analyzed for the effect of these missiles; the analysis was based on ultimate stresses. The telephone pole was considered to have a perpendicular incidence at mid point of a wall panel. Upon impact of the pole on the wall compression waves are transmitted to the opposite face of the struck wall (with a velocity equal to that of sound in reinforced concrete). These waves are reflected as tension waves. Based on this phenomenon, the projectile will pierce the 1'-6" wall without the wall attempting to stop it by its bending resistance. The remainder of the panel may develop cracks but in essence would remain structurally sound. For the 3'-0" wall and column section, the effect of the utility pole projectile would be different. The analysis reveals that a local failure would occur. The nature of the failure would be a deflection of the wall causing cracks and concrete spalling, however this would not impair the structural stability of the entire structure. The effect of a one-ton mass with a velocity of 100 MPH striking the reactor walls would be the same as that for the utility pole hitting the 3'-0" thick wall and column section.

d. The mean recurrence interval for a tornado striking the Quad-Cities site has been evaluated to be of the order of 500 to 700 years. The reactor will normally operate in excess of a year between refueling outages and these outages should be between two and three weeks in duration. The mean occurrence frequency of a tornado at the site during a refueling outage may therefore, be estimated at approximately 10,000 years.

If a tornado were to strike the reactor building, it has been calculated that damage would result to the structure above the operating floor. The reactor building superstructure siding is designed to accommodate up to 170 MPH wind velocity equivalent to a pressure of 75 psf. When this design velocity is exceeded, the siding will blow off and expose the refueling floor. The structural steel frame will withstand the force of a 300 MPH wind and at this point the steel will be at yield stress. The reinforced concrete structure would not fail. The reactor building and standby gas treatment system would probably be rendered ineffective for fission product retention and elevated release.

During the refueling operation, the primary containment and primary system are separated from the secondary containment by the large volume of water above the reactor vessel and drywell. If a tornado were to strike with these systems open, the greatest hazard appears to be associated with objects being blown into the reactor vessel or fuel storage pool which could result in fuel damage.

The general arrangement for storage of equipment during refueling is as shown on Figure 27 of the PD &A Report. It is planned that the reactors normally would be refueled separately. Therefore, at any one time only half of the equipment and facilities shown would be stored on the floor.

Exceptions to this are the refueling servicing tools which are common to refueling functions for both reactors.

The following material is presented to show the possible extreme effects of a 300 mph wind on some of the more obvious items.

A 300 mph wind will produce a velocity pressure of approximately 230 psf.
The reactor vessel head will provide a horizontal projected area of approximately 200 square feet. The side-on loading with a shape coefficient of 0.60 (hemispherical) results in a total horizontal force of approximately 14 tons. The vessel head weighs approximately 100 tons so that frictional forces should prevent movement. The vessel head is held on three (3) brackets on the floor, each of which is capable of developing a lateral resistance of at least 25 tons. The vessel head would not move.

The drywell head will provide a horizontal projected area of approximately 450 square feet. The total horizontal force resulting from a 300 mph wind will be approximately 36 tons. This assembly weighs approximately 65 tons so that friction alone may not prevent movement. The drywell head will be located to prevent this potential movement.

The spent fuel shipping cask will be located in the decontamination area after withdrawal from the pool. In this location, it is possible that the tornado winds could topple the cask. The geometry of the cask, the location of the decontamination area, and the location of the fuel storage pool and reactor vessel are such that the cask will not topple into the fuel storage pool or the reactor vessel.

Most of the tools and equipment used during a refueling outage are of such a geometry and density that they would not move under the contemplated wind load. If, however, a hand tool or similar sized object were to be blown into the vessel or storage pool it could damage some fuel. Objects of lighter densities such as vessel insulation sections which could be blown into the vessel or fuel storage pool should not cause fuel damage.

The consequences of failing fuel under postulated accident conditions may be approximated by considering the effects of the fuel loading accident analyzed in Section XIV-3 of the Plant Design Analysis Report This analysis postulated the fission gas release from 92 failed rods with the secondary containment and standby gas treatment system operable. The maximum off-site whole body and thyroid doses from this accident were calculated to be 5.5×10^{-3} and 3.0×10^{-3} rems respectively.

The answer to Question H-1 of this amendment discusses these results in conjunction with one of the design basis accidents. Based on calculations performed for Dresden, Quad-Cities and similar other reactor plants, the effect of the stack and standby gas treatment system is to reduce the whole body dose by a factor of about 10 and the thyroid dose by a factor of about 2000. Therefore, if 92 fuel rods were to be damaged during a tornado without the benefit of secondary containment, maximum off-site whole body and thyroid doses as calculated at 5.5×10^{-2} rem and 6 rem respectively. These valves are approximately 500 to 50 below the guideline valves of 10 CFR 100.

A literature search has been made to determine tornado effects on small bodies of water such as swimming pools and pools of similar size. This search uncovered no case where the water level was changed significantly as a result of the tornado.

Because of the reactor building geometry, the methods of construction of the fuel storage pool and the depth of water covering the fuel, it appears the fuel should only be disturbed to the extent that lighter objects may be blown onto the fuel inflicting minor damage.

e. The lower compartments of the secondary containment are constructed of reinforced concrete. The floor slabs are designed for live loads of 350# per sq. ft. up to 1000# per sq. ft. The walls above grade can accommodate exterior tornado wind velocities of 500 MPH, equivalent to 908# per sq. ft., without exceeding normal stresses.

Tornado data on pressure gradients is practically non-existent, but the lowest recorded air pressure during a tornado occurred at St. Louis, Missouri on May 27, 1896. This reading was 26.94 inches and was 2.42 inches lower than the pressure recorded at the same time at the weather bureau office seven blocks away. The 2.42" pressure differential is equivalent to a loading of 170# per sq. ft. which is much less than the design loads. This loading of 170# per sq. ft. would be induced only by an instantaneous pressure drop.

Below grade, the suppression chamber walls are designed for a saturated soil condition of 87# per sq. ft. With an assumed ground water elevation of 590'-0" during an extreme flood of 586'-0", the external wall loading at the bottom of the suppression chamber compartment will be added to the 170# per sq. ft. vacuum load due to the tornado. The walls are designed to accommodate the stress due to this loading condition based on allowable overstress of 33-1/3%.

QUESTION

B. Site (Continued)

2. State the site elevation, maximum expected water level and snow loading design criteria.

ANSWER

The plant site elevation for the Quad-Cities Station, Units 1 and 2 will be 594'-6". All major buildings and structures will be located at that elevation. The land elevation at the site is 605', thus the area occupied by the plant will be excavated to the 594'-6" elevation. The slope between the 594'-6" plant site and the 605' elevation of the surrounding land will be approximately 3:1.

The high flood level was established in April, 1965, reaching an elevation at the site of 586'. This results in the plant site having an 8'-6" margin above maximum flood stage. The plant switchyard will be located at the 605' elevation, 19' above maximum flood stage.

The plant buildings and structures will be designed to withstand, within applicabl code allowance, a snow loading of 30 lbs/ft^2 as recommended by the American Standards Association for the area of the site.

QUESTION

C. Seismic Design

The following Additional information relative to the seismic design of your proposed facility is required:

1. Clarify maximum earthquake ground acceleration values for which the plant is designed.

ANSWER

The seismic design for critical structures and equipment for the station will be based on dynamic analysis of acceleration or velocity response spectrum curves which are based on a ground motion of 0.12 g. To assure that the station can be shutdown with the containment and heat removal facilities intact, the critical structures will be designed to accommodate a ground motion of 0.24 g. Further discussion of the seismic design considerations are provided in Sections I-2.7.5, II-6, and V-3 of the Plant Design and Analysis Report.

QUESTION

C. Seismic Design (Continued)

2. What is the source and applicability of the response acceleration spectrum plot on Plate V of Appendix F? Please provide a more accurate plot on log paper to include the short period range.

ANSWER

The response acceleration spectrum plot on Plate V of Appendix F was developed through studies made of local and area geology, seismology, and seismic history by John A. Blume and Associates, Engineers. Dr. Ross Heinrich was Consulting Seismologist. Because most of the critical components and structures are modeled and subjected to an excursion through ground motion, it is desirable to adopt an actual earthquake record which produces a ~~response in agreement with that postulated on the basis of the seismic~~ investigations. In this case, the 1957 San Francisco Golden Gate Park S80E was chosen.

The purpose of the spectra shown is to provide earthquake response in a dynamic analysis. These are averaged single-mass spectra and are usually employed in a model analysis. These are not generally used in the analysis of important critical components, since computer programs are utilized which follow the actual earthquakes, rather than their response curves, and it is not necessary to combine modes in some approximate manner to arrive at final shears, moments and displacements.

The response acceleration spectra are presented on a log-log plot as shown on the attached figure. As explained above, these curves are not generally used in the analysis of important critical components, since computer programs are utilized which follow the actual earthquakes.

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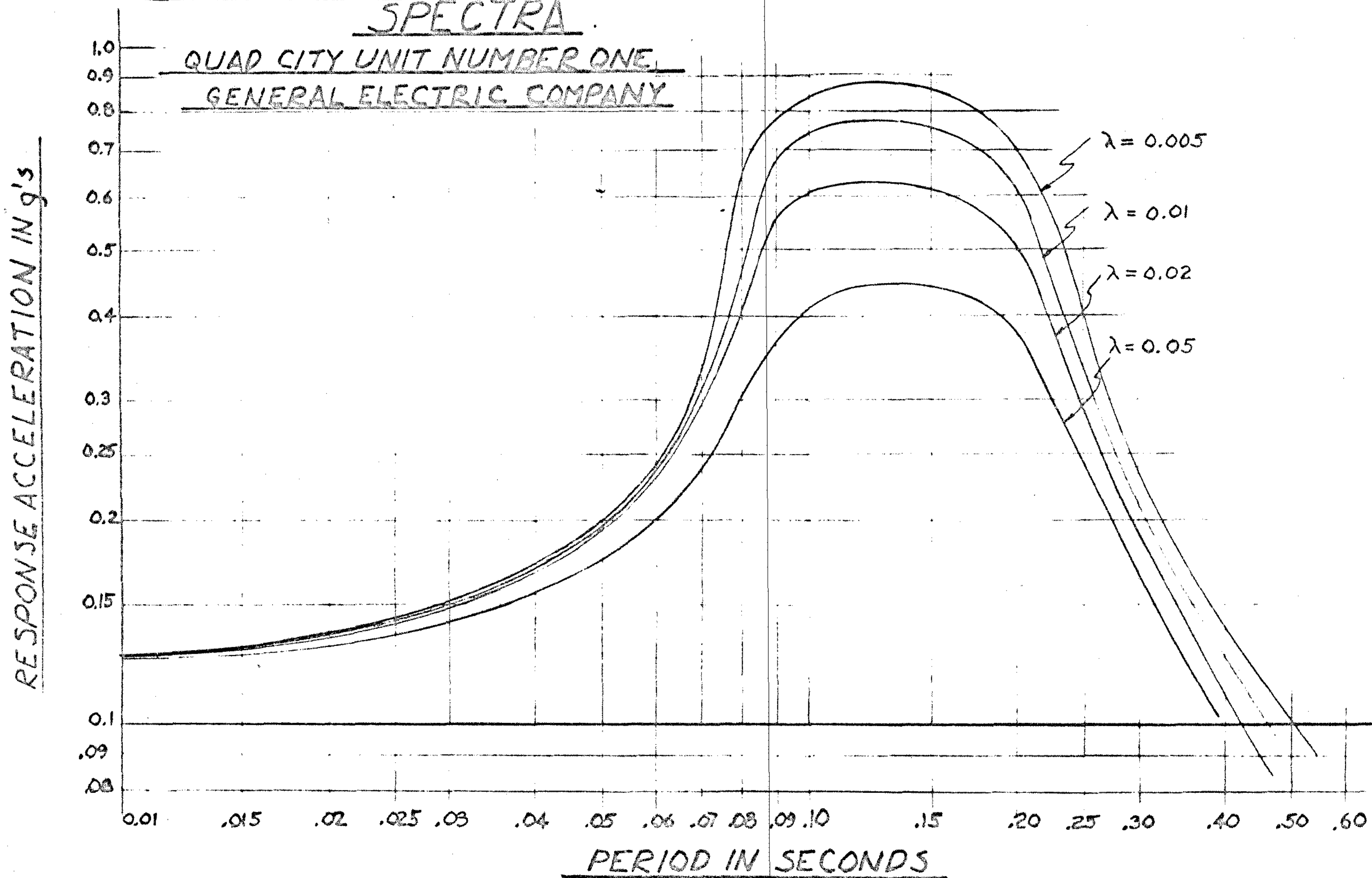
FIGURE C-2-1

RESPONSE ACCELERATION SPECTRA

QUAD CITY UNIT NUMBER ONE
GENERAL ELECTRIC COMPANY



QUESTION

C. Seismic Design (Continued)

3. Provide a table of the damping factors to be used for the Quad-Cities plant. Discuss the advisability of using 2% damping for those structures and components important to safety.

ANSWER

The damping factors to be used are listed below and are the same values given in the "Plant Design Analysis", Volume 1, Quad-Cities Station Units 1 and 2, as amended, Table V-5, Page V-3-2:

ITEM	PERCENT OF CRITICAL DAMPING
Reinforced Concrete Structures	5.0
Steel Frame Structures	2.0
Welded Assemblies	1.0
Bolted and Riveted Assemblies	2.0
Vital Piping Systems	0.5

Most Class 1 structures or equipment, such as the drywell, vents, suppression chamber, reactor vessel, pumps, and other mechanical pieces of equipment are assigned damping values of 2 percent or less. The reactor building and ventilation stack are assigned damping values of 5 percent. Stresses in the walls of the reactor building resulting from earthquake loads are approximately 100 pounds per square inch. The adjacent walls, acting in overturning, are stressed to about

200 pounds per square inch. These values are without consideration of wall reinforcements. Dead load from the walls above only (no floor loads included) results in an additional 100 pounds per square inch. These shear and tension values are doubled for twice the design earthquake condition. Stresses in ventilation stacks due to earthquake forces are of the same magnitude as noted above. It is a well known fact that damping values for concrete structures will vary with the stress level and since the above stresses are near the allowable values it is reasonable to assign a 5 percent damping value.

QUESTION

C. Seismic Design (Continued)

4. Describe the manner in which the seismic response analysis for the facility stack will be carried out.

ANSWER

The stack is treated as an elastic cantilever system with masses appropriately lumped. The resulting masses are considered supported by weightless elastic columns. Natural frequencies, mode shapes and flexural response of the equivalent multi-mass system are computed with the aid of a digital computer utilizing the actual earthquake record rather than response curves.

Stiffness characteristics of the stack include the contribution of shear rigidity in addition to flexural rigidity. Rocking of the stack on its foundation is considered in a coupled system rather than combining a rocking mode with the flexural modes.

Normally at least five modes are considered, and a damping value of five percent is used on all five modes. Curves showing the envelope of maximum shears, moments, and displacements versus height are determined.

QUESTION

C. Seismic Design (Continued)

5. Describe the manner in which Class I components within Class II structures will be analyzed.

ANSWER

The response of the Class II structure will be calculated taking into consideration the characteristics of the structure and its foundation. After the response of the structure has been estimated the Class 1 component will be classified and then designed in accordance with its rigidity as defined in Quad-Cities Station Units 1 and 2 "Plant Design Analysis" Volume II, Appendix F - Pages 7 and 8.

C. (Seismic Design - continued)

6. Describe the inspection and quality control procedures to be followed during the construction of containment structures and other critical structures to insure that the seismic design criteria will be met.

ANSWER

For the primary containment the quality control and inspection procedures are as follows:

1. The seismic design criteria are accommodated in the engineering design of the containment vessels per applicable ASME codes. Calculations for these structures are submitted for approval by the architect-engineer prior to starting shop fabrication.

2. Certified mill tests are required for each heat of steel to be used and are submitted for approval by the architect-engineer.

3. Qualification of containment fabricators welding procedures prior to start of fabrication.

4. Certified charpy impact test reports are required for approval by the architect-engineer on plate and pipe materials.

5. Containment contractor provides quality control for shop fabrication with periodic shop inspection by the purchaser's representative.

6. Hartford inspection for both shop and field work performed by containment vessel fabricator.

7. Radiography per applicable ASME code os shop and field welds. Repairs where necessary are made per ASME code.

8. Pneumatic overpressure and leak rate tests on completed containment vessels. The overpressure test will be conducted with the suppression chamber filled with water to the normal operating level.

For the concrete secondary containment, the quality control and inspection procedures are as follows:

1. Certified mill tests are required for each heat of A432 reinforcing steel.

2. Inspection by independent laboratory services of all concrete and steel supplied to the jobsite as follows:

a. Physical and chemical tests on all reinforcing steel.

b. Soundness tests on all aggregates.

c. Make and test concrete cylinders for each 100 CY of concrete placed to determine compressive strength. Two cylinders are made for 7, 28 and 90 day tests for each 100 CY.

d. Slump tests of all concrete are made at time of placing concrete.

e. All compressive tests are submitted for review after cylinders are broken.

3. Curing compounds are used to insure minimum loss of water during curing. Stripping of forms by subcontractors is specified and controlled by construction manager to assure appropriate initial strength in concrete.

4. Preliminary inspection of concrete supplier's source of aggregate and routine checks on aggregate quality by construction manager during course of job.

5. Control of subcontractor's concrete placement by the construction manager in the event of cold weather. Cold weather precautions are specified if temperatures go to 40°F or below.

QUESTION

C. Seismic Design (Continued)

7. Describe the response of the overhead crane to the design earthquake.

ANSWER

The bridge crane in the reactor building will be designed to adequately withstand the plant design earthquake forces described in Section V-3, Plant Design Analysis Report. The structural steel supporting these cranes will also adequately withstand these earthquake forces and would also remain intact.

The crane bridge wheels are each double flanged and ride on rails firmly attached to the superstructure steel. Although the crane would be displaced during an earthquake, it would not leave the crane rails. The trolley will be attached to the bridge with clips which will run under the rails to prevent the trolley from being thrown off the bridge rails.

The bridge wheels on the main runway are equipped with solenoid-actuated spring loaded brakes. Upon loss of power or when the crane is parked, the spring actuates a brake locking the wheels firmly in place to prevent the crane from rolling out of position.

QUESTION

D. Containment

1. To protect the containment penetrations for large lines, pipe anchors and stops are provided to limit movement in the event of postulated pipe breaks. Although it has been stated that anchors and stops will be provided to withstand axial forces, it is not stated that such anchors can accommodate bending and twisting moments associated with pipeline failure without stressing the containment penetration. Please submit criteria to indicate the design requirements to preclude pipe failures occurring directly at the penetration. Also provide the design and location of the principal pipe anchors inside and outside containment on the main steam lines. Demonstrate how these restraints prevent stressing of the containment penetration.

ANSWER

The containment penetrations for large lines, such as steam piping are protected by pipe anchors and guides to limit movement of the pipe in all directions.

The piping design criteria accounts for normal thermal expansion, live and dead loads, seismic loads and accident loads. The criteria are as follows:

a) Major line movement is restricted to movement in the axial direction by the guides.

b) Guide stops are provided to prevent movement in excess of the normal thermal expansion.

c) Piping must not impose stresses on the penetrations greater than those permitted by code allowable under conditions of a pipe break combined with a design earthquake.

d) Piping must not impose conditions on the penetration which will affect the penetration integrity under conditions of a pipe break combined with two times the design earthquake load.

Bending and torsional movements on the containment penetration are precluded by the use of pipe guides and anchors located in a manner to satisfy the design criteria described above. Pipe supports provided with stops and guides will allow axial movement for normal pipe line growth.

The analyses and the design of the pipe anchors and supports inside and outside of the containment have not been completed to date. Such designs when completed, will satisfy the above design criteria.

QUESTION

D. <u>Containment (Continued)</u>

2. Under the conditions of a steam line break, the steam line isolation valves would be required to close under higher than operating flow conditions. In view of this, describe the test conditions under which the valves will be tested for closing time. How will these tests be related to accident and operating conditions?

ANSWER

The specifications for the steam line valves require that each valve be tested by the manufacturer to demonstrate that the valve will close within three seconds against a line pressure of 1000 psig.

The design basis for these valves requires that they be designed to permit periodic tests including the following: testing under appropriate power operation conditions or during shutdown such as leakage testing of the valve at containment design pressure; testing of the various automatic and manual modes of actuation and operation; checking set points of actuating sensors; and testing over the valve full stroke range.

At the present, definite operational test procedures for these valves are in the formative stage. Meaningful test procedures will be developed prior to plant startup. One possible test of closure time, which would minimize effects on plant operation, would be to reduce total steam flow to 75% then trip one valve closed and determine its closure time.

The relationship of this test to accident conditions may be judged by comparing the following flows and pressures:

Test - Steam Flow	∿75%
Test = ΔP on Valve	5 to 40 psi
Line Rupture - Steam Flow	∿200%
Line Rupture - ΔP on Valve	∿1000 psi

Although the test conditions are only partial conditions with respect to line rupture flow and pressure, nevertheless, they are deemed to be sufficient to insure operation of the valves when required.

QUESTION

D. Containment (Continued)

3. The containment spray system, by virtue of its function does not require automatic isolation valves. However, the system components are shared with the shutdown cooling system, which in turn has provision for connections to the fuel pool. Unless the isolation valves in the interconnected system are subject to leakage tests, the boundary of containment spray systems may be extended to the fuel pool. Please specify the containment isolation criteria applicable to systems with shared components.

ANSWER

The containment isolation criteria for shared systems is the same as for non-shared systems with respect to isolation valve testing. The fuel pool connections shown on the shared shutdown/containment cooling low pressure coolant injection system are normally sealed off with blind flanges. This connection capability is provided for additional fuel pool cooling in the event it is necessary to unload the entire reactor core and store it in the fuel pool. Temporary connecting piping would be installed to utilize the shutdown heat exchangers if the fuel pool heat exchangers could not hold the fuel pool to the desired water temperatures.

QUESTION

D. Containment (Continued)

4. How will leakage external to the primary containment (e.g. containment spray pump seals) be factored into the containment leakage rate and what provisions will be made to enable the measurement of such leakages in those systems and components external to the primary containment?

ANSWER

Refer to Dresden Unit 2, (AEC Docket 50-237) Plant Design and Analysis Report, Amendment No. 4, Question 13. The leakage from pump seals is insignificant in comparison to the design primary containment leakage. All pumps in the core spray system and the containment spray system have leakoffs, which drain to the radioactive waste disposal system. Therefore, any leakage from these pumps can be collected and measured to determine their contribution to the containment leakage rate.

QUESTION

D. <u>Containment (Continued)</u>

5. Please specify the isolation criteria and bases applicable to closed systems within and outside the containment.

ANSWER

The isolation criteria applicable to closed systems is given in paragraph V.1.2.4c of the Quad Cities Station, Units 1 and 2, as amended, Plant Design Analysis. All of the lines for which this criteria is applicable are closed outside the containment. Even if a loss of coolant accident inside the containment caused failure of one of these lines within the containment, the line is still closed outside the containment and the (redundant) isolation valve can be closed to isolate the line. The closed segment of each of these lines outside the containment are designed to withstand the containment design pressure and therefore will not rupture on communication with the containment.

QUESTION

D. <u>Containment (Continued)</u>

6. Since the secondary containment is relied upon to significantly attenuate off-site doses for the design basis accidents, please provide the following information to enable an evaluation of the capability of the system to perform its intended function.

   a. Discuss the reliability and capability of the means provided to maintain a negative pressure under various external wind and barometric conditions. What means will be provided to determine that a minimum negative pressure of 0.25 inch of water will exist at any location of the entire peripheral area of the building structure? Describe the test procedure planned to demonstrate periodically the leak-tightness of the building structure and the instrumentation to be used in such measurements.

   b. What provisions are planned to limit in-leakage to the specified rate in the event of increased infiltration from the development of fissures, increase in differential negative pressure or the inadvertent opening of access doors?

   c. The accident analysis assumes that fission products leaked from the primary system are evacuated from the secondary containment at the rate of 100% of the containment volume per day. This holdup of fission products has a significant effect on reducing the calculated two hour dose in the maximum hypothetical accident. Discuss the potential for inadvertent evacuation of compartments near the primary system at a rate greater than 100% per day thus reducing fission product holdup in the secondary system and increasing calculated off-site doses.

   d. Discuss the possibility of diffusion of fission products through fissures in the building walls against the small pressure gradient proposed in your design.

   e. Describe the design of the joints in the steel superstructure of the refueling building which insure no significant impairment of the leakage characteristics by the design earthquake.

ANSWER

a. The secondary containment ventilation and standby gas treatment system is discussed in Section V-2 of the Plant Design Analysis Report.

The reactor building (secondary containment) is designed so that its in-leakage rate is not greater than 100 percent of the building volume per day under neutral wind conditions when the building is subjected to an internal negative pressure of 0.25 inch of water. The behavior of the reactor building leakage rates as a function of various external wind and barometric conditions is discussed in Section V-2.2.2 of the Plant Design Analysis Report. In summary, the studies show that at wind velocities less than 35 to 65 mph there would be little if any exfiltration from the reactor building. Maximum and minimum calculated reactor building exfiltration rates as a function of wind velocity are shown in Figure 119 of the Plant Design Analysis Report. The calculations which were made also suggest that exfiltration rates are almost directly proportional to the initial in-leakage rate for a given negative building pressure. Calculations show that the exfiltration rate could be many orders of magnitude larger and occur at much lower wind speeds without increasing the post accident doses above 10 CFR 100 guidelines. This is discussed further in answer to question H-1 herein.

The standby gas treatment system has been designed to have a capability of maintaining a 0.25 inch of water negative pressure for a building in-leakage rate of 100 percent of the building volume per day. The in-leakage rate and differential pressure may be measured during any normal mode of plant operation. A test section will be installed in the common exhaust duct upstream from the two gas treatment units. The test section will have been calibrated using a standard pitot tube with the number and location of traverse points as recommended in the "Standard Test Code for Air Moving Devices" but permitting subsequent testing through the use of a single point velocity pressure reading on an inclined U-tube manometer. The test instrument board will also include indication of air temperature immediately downstream of the test section. The reactor building static pressures will be transmitted from a point in the general open area of each floor. Because the reactor building above the concrete structure is such an "open" volume, significant pressure differentials within this volume are not expected to occur. The system may be tested as follows:

a. Shutdown the reactor building ventilation supply fans and close the supply isolation valves.
b. When the building static pressure reaches 0.25 inches of water subatmospheric pressure, start either unit of the emergency gas treatment system.
c. Shutdown the building main exhaust fans and close the exhaust isolation valves.
d. Position all control dampers to permit all building exhaust to go through the operating emergency gas treatment system. After equilibrium conditions have been reached, obtain the static pressure, air temperature, and pitot tube manometer readings. From this information compliance to the in-leakage specification may be determined.

Humboldt Power Plant Unit No. 3 operated by the Pacific Gas and Electric Company has the same in-leakage requirements as those specified for this station. Experience at the Humboldt Plant suggests that the design criteria may be readily satisfied.

b. In the event of in-leakage rates in excess of the allowable rates the source(s) of the in-leakage will be located and repaired as required to meet the design criteria. Repairs would be made in accordance with approved procedures. After repairs are completed the in-leakage will be determined to assure compliance with the accepted criteria.

An increase in differential pressure could result from increased flow through the standby gas treatment system or decreased in-leakage. Since each standby gas treatment system fan has a fixed capacity and is sized for 100 percent of the building volume per day the maximum flow through the system is fixed. Total flow through the system is, therefore, expected to be determined by the in-leakage and a decrease in in-leakage should result in an increase in differential pressure. This presents no problem and no corrective action is required.

All personnel and equipment access openings are airlocks provided with mechanically interlocked double doors, with weatherstrip type seals. The interlocks on these openings provide a means for assuring that building access will not result in loss of containment integrity. Possible deterioration of seals on the air locks will be detected through periodic inspections and tests.

c. The standby gas treatment system takes suction from many areas of the reactor building. Reactor building in-leakage will occur over the whole surface of the reactor building, but predominantly through the sheet metal walls above the operating floor rather than through the thick concrete walls which includes all surface below the operating floor. The major part of the standby gas treatment system flow will come from above the operating floor. Since leakage from the primary containment will generally occur below the operating floor, the effective ventilation rate of fission products will be well below 100%/day when the whole reactor building ventilation rate is 100%/day. Therefore, the holdup for fission product decay will be greater than that assumed in the Plant Design and Analysis Report.

The major effect of the reactor building and the standby gas treatment system in reducing accident doses is filtration and release from an elevated point. The holdup for decay is very secondary. For example, even if half the volume into which the primary containment leakage leaked were exhausted at 200%/day (increasing the effective reactor building leakage rate by 50%) the resultant off-site doses would be only about 6% greater. This would be a negligible increase to a negligible dose.

d. A consideration of the volumetric leak rates due to pressure differences and concentration differences shows that there is a potential for leakage of fission products through fissures in the building walls against a small negative (0.25 inch of water) pressure gradient. The data below shows the relative contribution of diffusion of krypton, xenon, and iodine against a pressure gradient.

Table D-1

Calculated Relative Diffusion of Gases

Gas	Relative Contribution of Diffusion
Krypton	5.1 (%)
Xenon	3.3
Iodine	2.1

The relative contributions of diffusion given above would represent maximum values since the concentrations outside the building were considered zero, thereby giving a maximum concentration gradient.

The volumetric flow rates were calculated using the equation on page II-2 of NAA-SR-10100. The following numbers were used in the equation:

Crack width, b = 1.4 mil
Crack length, L = 330,000 feet
Wall thickness, X = 1 foot
Pressure difference, Pi-Po = 0.25 inch water

The volumetric flow rate for the diffusion case was:

$$q_c = \frac{bLD\ (Ci-Co)}{X}$$

where Ci = concentration inside (Co assumed zero)

D = diffusivity (evaluated by the method on pages 512-13 of <u>Transport Phenomena</u> by Bird, Stewart and Lightfoot; Wiley and Sons, 1960.

b, L, x are same as before.

The final model used by combining the two equations was:

$$R_T = \left[\frac{Ci \; q_c \; b^3 L \; (Pi - Po)}{12 \; \mu x} + \frac{bLD}{x} \right]$$

where q_c = gravitational conversion factor, and μ, the viscosity of air. The relative effect of the diffusion term was indicated in table above.

In summary, there is a possibility of some diffusion occurring against the small pressure gradient, but at a maximum this effect is only on the order of 2-5 percent of the total in-leakage.

e. The structural steel frame of the refueling building will deflect when subjected to the ground motions due to the design earthquake, however, the structure is designed to accommodate the forces caused by these ground motions. The siding will be designed with caulked interlocking vertical joints and overlapping horizontal joints. The horizontal joints will be overlapped sufficiently to insure leak tight integrity. To further insure the leak tightness of the siding, all joints will be provided with mechanical fasteners to provide positive connections to the structural steel. In the event of a design earthquake causing movement of structural steel frame, some of the siding connections may be distressed, however, it is expected that the mastic joint sealant will provide sufficient resilience to prevent the exfiltration of air.

QUESTION

E. Instrumentation

1. Describe the design of the rod-block system with respect to designing against a single failure.

ANSWER

The rod block system is not intended as a safety function to protect against serious reactivity transients. It is, however, a backup to procedural control to prevent inappropriate sequences of control rod withdrawal. The most probable failures (relay coil malfunctions, loose connections and solenoid failures) will cause relay drop out and result in a rod-block. Also, the neutron monitoring trips are arranged so that those associated with protection logic channel A can give a rod-block independent of those for logic channel B. This is accomplished by having relay contacts from logic channel B open the rod select bus circuit whereas relay contacts from logic channel A neutron monitor trip and all other rod-block trips open the circuit to the rod withdrawal bus. These two buses are electrically independent such that no single short in the wiring could prevent a rod-block from occurring if one of the four neutron monitor (APRM) from each of the two protection logic channels give a trip signal.

QUESTION

E. Instrumentation (Continued)

2. Please describe in detail the power/flow instrumentation and associated circuits. Include an analysis which shows that no single failure within this system will nullify rod-block capability in response to off-normal power/flow conditions.

ANSWER

The flow is sensed by two ΔP transmitters (an A and a B) on each of the 2 recirculation loops flow nozzles. Summation of the outputs of the two A transmitters is accomplished by a two-point summer to give total recirculation flow. The B transmitters are similarly added in a second summer to give a second total recirculation flow. The outputs of the two summers are compared and a pre-established differentiation between them causes a dual rod-block signal (one from each of two comparator circuits).

The A summer feeds into the individual bias units for APRM channels 1, 2, 3, and 4 and the B summer feeds the bias units APRM channels 5, 6, 7, and 7 where channels 1 and 5 are in the same quadrant of the core, etc. Thus, failure of one bias unit can affect only one APRM, failure of one summer can affect only one half of the APRMS, but this failure will cause a rod-block by virtue of the comparator circuits mentioned above.

Failure of one ΔP transmitter or square root connector will cause mismatch. Incorrect lineup of the instrument values at the ΔP transmitters will cause an apparent flow mismatch and cause rod-block. Closing of one set of root valves, for the ΔP transmitter, could give an erroneous reading (up scale or downscale) on both transmitter A and B. The root valves will be locked in open position and checked prior to operation of the system and not distributed for normal calibration.

Closure of a flow check valve by excessive blow-down during calibration would only affect one transmitter. This condition will be corrected before completion of the calibration process and could not occur during normal operation without having an instrument line break of sufficient magnitude to close an excess flow check valve.

QUESTION

E. Instrumentation

3. Please provide an analysis of the refueling interlock system which demonstrates that no single failure within the system will allow more than one rod at a time to be fully withdrawn. Mode switch circuitry should be included in the analysis.

ANSWER

The refueling interlock system is designed to function as a operator-backup during the refueling process. It is brought into service to monitor the operator during the period when he is performing the stuck rod criteria. The system is designed to have maximum reliability using standard components and is readily testable for operation. In the unlikely event that the system should fail to operate it is only one event in a series of events that must occur before withdrawing multiple rods would be of any consequence.

QUESTION

E. Instrumentation (Continued)

4. Please discuss the effects of over-temperature on control room instrumentation. Discuss provisions to terminate reactor operation in the event of excessive control room temperature.

ANSWER

All components of the safety system in the main control room can tolerate higher ambient temperatures than a human operator can. The temperature ratings of the radiation and neutron monitoring amplifiers and trip circuits are $\underline{50}^{\circ}$C with a degrading of accuracy of only 1% which does not constitute a safety problem but merely a probable 1% shift in trip points.

In the event personnel had to evacuate the control room the procedure for a safe plant shutdown as outlined in question E-5 would be followed.

QUESTION

E. <u>Instrumentation (Continued)</u>

5. Assume that fire occurs in the control room during full power operation and that, as a result, immediate evacuation of the control room is required. Please provide an analysis to show that an orderly shutdown can be accomplished under these circumstances. A discussion of manual-override capability of controls (e.g. valve controls) which might be accidentally actuated by spurious signals resulting from the fire should be included.

ANSWER

The probability of fire occurring in the control room is quite remote for the following reasons:

(a) furniture and cabinets will be constructed principally of non-flammable materials, (b) paper records will be stored in metal cabinets, and (c) only minor quantities of flammable material in the form of records, reports, food containers, etc. will be brought into the control room during the course of unit operations.

Therefore fire in the control room which could cause evacuation of personnel before they could initiate the immediate shutdown of the reactor is difficult to conceive. However, assuming that such a situation could occur, it is possible to effect a safe plant shutdown by use of local control panels located in the turbine and reactor buildings. A reactor scram and turbine trip can be accomplished by opening power breakers at the D.C. power centers. The depressurizing and cooling of the reactor system is possible by operation of the reactor auxiliary systems from their local control panels. Motor operated valves which have become inoperative can be operated manually. Instrumentation on local control panels would provide the information needed to assess the condition of the reactor and turbine during the shutdown operation.

QUESTION

F. Reactor Core Isolation Cooling System

1. Under what conditions and restrictions would the RCIC system be used during the operation of the plant? What limit is proposed for radioactive isotopes in the pressure suppression pool water and air volumes to insure access to the containment? What provisions are proposed to reduce activity levels in the suppression pool air and water volumes should this become necessary?

ANSWER

The RCIC system is to provide reactor primary coolant inventory makeup under conditions of isolation of the reactor from the main condenser (which is the primary heat sink) coupled with loss of capability of inventory makeup with the boiler feedwater pumps. This system is automatically actuated by a low-low reactor water level signal or by manual signal. If coolant inventory can be maintained by means of the feed pumps, the RCIC system would not be operated. Relief valve actuation would be required in either case to control pressure in the primary system. Except for testing it is not intended that the RCIC system be actuated during power operation of the reactor.

The RCIC system operation will result in slow heatup of the suppression pool water. One of the two shutdown/containment cooling systems must be placed in service one hour after initiation of the RCIC system to limit pool temperatures to 140°F. If in the very unlikely event that a cooling system could not be placed into service by this time the reactor primary system would have to be depressurized by use of the relief valves. The rate of depressurization would be consistent with normal cool down rates. Since two cooling systems of 100% capacity are provided and are powered from the emergency power bus it is unlikely that a cooling system would not be available.

Use of the RCIC system with makeup from condensate storage would be restricted by water inventory in the suppression pool. High water level in the suppression pool would require that suction for the RCIC pump be from the suppression pool thus maintaining inventory equilibrium.

Under normal continuous operation of the plant, the activity levels in the suppression pool are not expected to exceed background levels by more than a few mrem per hour. The permissible activity level would be determined by access requirements for inspection and maintenance during scheduled outages and would be governed by limits applicable to limited access areas to be established prior to startup of the units.

Clean-up of the suppression pool water can be accomplished by circulating the suppression pool water through the waste water filter and demineralizer system located in the radwaste facility. A pipe connection would have to be made to either the core spray system or the low pressure coolant injection system to accomplish such clean-up. This is not provided as part of the normal operating piping in order to minimize the potential of inadvertent draining of the suppression pool. Clean water would be returned to the suppression pool through any of the supply headers from condensate storage.

QUESTION

F. Reactor Core Isolation Cooling System (Continued)

2. Discuss provisions for manual operation of critical valves in the RCIC and suppression pool cooling system.

ANSWER

Only the D.C. powered valve in the RCIC turbine inlet steam line is required to open to put the RCIC system into operation. This valve serves also as the outboard isolation valve and will be located in an accessible area in the reactor building. It will be capable of manual operation from the control room, although the normal mode of operation is automatic actuation.

All valves in the suppression pool cooling system are accessible and provided with manual operating features.

QUESTION

F. Reactor Core Isolation Cooling System (Continued)

3. What is the criteria for protection of vital equipment from possible missiles from the RCIC turbine?

ANSWER

Vital equipment in the engineered safeguards systems will be located and arranged so that such equipment is protected to the maximum extent possible and that a single failure will not destroy the capability of the system to perform its intended function. Components of each system will be examined for the potential of the generation of missiles.

The RCIC turbine and pump for Units 1 and 2 are located in their own con~~crete shielded compartment. There is no other equipment which is vital to~~ safe plant shutdown located within these compartments. See revised Figure 29 as submitted with this amendment.

Turbines of the type used for the RCIC system have shells that are sufficiently strong as to contain any potential missiles. Failure rates for these turbines is very low and there is no case of a missile escaping the shell of one of these units.

QUESTION

F. Reactor Core Isolation Cooling System (Continued)

4. State the criteria which would determine the time at which the reactor would be depressurized if the suppression pool coolers became inoperative during operation of the RCIC system. If the suppression pool is allowed to exceed about 140°F (the upper limit of blowdown test data), what is the basis for assuming complete condensation of steam during a coolant loss accident?

ANSWER

Two separate suppression pool cooling systems are provided;each is capable of handling the design heat load. In addition to the high reliability of A-C power to the station, either of the two systems can be operated from the diesel emergency bus. Therefore, it is highly unlikely that the suppression pool coolers would become inoperable.

The containment cooling systems will be normally actuated within one hour after initiation of the RCIC system. As shown in Figure 106 of the Plant Design and Analysis Report, the pool will not exceed 140°F due to the ability of the containment cooling systems to dissipate the decay heat being removed by the RCIC. In the highly improbable event of a design basis accident following significant operation of the RCIC system, the temperature at the end of blowdown would not exceed 170°F.

The upper pool temperature limit at which steam no longer condenses properly has not been established experimentally, but the tests conducted at Hanford Operations (described in reply to Question F-5) have shown that steam quenches without difficulty even at pool temperatures well in excess of 170°F.

General Electric's experience with both the Bodega and Humboldt Bay pressure suppression tests indicate that condensation will occur in or near the temperature range of interest. Initial pool temperatures were varied during these tests to determine the effect on steam condensation. Vapor condensation was rapid and complete throughout the entire range tested.

Humboldt Bay Test #26, conducted with an initial pool temperature of 141°F and the design basis break, resulted in a final pool temperature of 170°F. Also significant was Bodga Test #39. This had an initial pool temperature of 120°F and a final pool temperature of 163°F after the design break blowdown.

Therefore, it may be concluded that the quenching of steam will occur even at the maximum expected pool temperatures.

QUESTION

F. Reactor Core Isolation Cooling System (Continued)

5. Discuss the effect of water temperature during blowdown on vibrations observed in the Moss Landing tests. What stresses might these vibrations impose on the vent pipes and how would containment integrity and suppression pool capability be affected?

ANSWER

In a 1959 ASME paper, 59-A-215, Pressure Suppression Containment for Nuclear Power Plants, C. C. Whelchel and C. H. Robbins, it is stated that, "Test results showed that tank vibration began when the water was 120°F - 130°F or hotter. It was most severe at high steam flows."

Subsequent tests at Moss Landing and at Hanford in connection with the NPR did not repeat these vibrations with hot pool water. It is concluded now that if the pool walls and piping are properly designed no excessive vibrations will occur and no significant vibration-induced stresses will exist.

The vibrations observed in the early Moss Landing tests were most likely the result of hitting a natural frequency of the tank itself. This tank was made of one-quarter inch plate, 20 feet in diameter and 24 feet high. In later tests at Moss Landing of the Humboldt and Bodega design the suppression pool wall was comprised of concrete lined with steel plate and the vent pipes were securely anchored. No significant vibrations were noted in the Humboldt or Bodega tests. These tests were more representative of current actual suppression pool design.

Other significant results relating to pressure suppression are contained in HW 68609, NPR Primary Loop - Emergency Dump Tests, 1961. The tests described in HW 68609 were run specifically because of the vibration phenomenon which was mentioned in the 1959 ASME Paper noted above. Hanford ran a number of tests to determine whether vibration would be excessive in a pressure suppression type system planned and built for NPR. They ran tests with final water temperatures as high as 200°F and found that vibrations were either nil or minor.

Several of the NPR conclusions are quoted below:

> "The tests did not uncover any unsafe conditions when quenching hot pressurized water in pools of subcooled water."

> "Serious quench tank vibrations were not observed in any of the runs which included quench water temperatures of 70° to 200°F. Vibrations were nil with the dumps of 455° to 600°F water. Minor vibrations developed during the dumps of 270°/275°F water with the vertical, horizontal and cross-distributors. Vibrations were also nil with the ring sparger and draft tube during the 270°/275°F dumps."

In summary, the results of pressure suppression tests which have been conducted over the range of parameters of interest to the Quad Cities Units 1 and 2 design support the conclusion that the current design is proper and adequate to eliminate blowdown vibrations which might impose restrictions on suppression pool capability.

QUESTION

F. Reactor Core Isolation Cooling System

6. Will sensors to initiate the reactor building emergency ventilation system be located in the suppression torus to allow actuation of this system should high activity levels be present due to operation of the RCIC system?

ANSWER

Radiation sensors are not provided in the suppression torus. The suppression chamber has the capability of limiting the leakage of fission gases into the secondary containment under peak transient pressure conditions. During operation of the RCIC system the pressure difference between the suppression chamber and the secondary containment will be essentially zero. However, if there were leakage of fission gases to the secondary containment it would be sensed by the radiation monitors in the reactor building exhaust ducts. The standby gas treatment system is then placed in service automatically on the signal of high radiation in the reactor building exhaust ducts.

QUESTION

G. System Analysis

1. To achieve consistency in the structural integrity of all components in the primary system, comparable quality in design and manufacture must be employed. Due to differences in code requirements, however, it appears that certain primary system components could be designed to different quality standards unless additional requirements are imposed. Please submit the following information on the primary system components:

   a. Design specifications, including design life.

   b. Vessel classification selected.

   c. Lifetime design cycles for each transient

   d. Requirements imposed over and above code rules.

ANSWER

The probability of failure of the piping and other primary loop components is approximately equivalent to the failure of the reactor vessel. However, in order to establish a rational basis for containment design, the design basis accident is assumed to be the mechanical failure of the reactor primary system equivalent to the circumferential rupture of one of the main recirculation pipes. This by no means was ever meant to infer that pipe or component rupture was considered credible - this assumption only provided a rational design approach.

As stated in the Plant Design and Analysis Report, paragraph 2.2.2 on page IV-2-2, the primary piping system (both recirculation system and primary steam system to the first isolation valve) is designed, fabricated and constructed to meet, as a minimum, the requirements of the American Standard Association Code for Pressure Piping, ASA B31.1 and the applicable State Regulations which require that the piping meet the requirements of Section I of the ASME Boiler and Pressure Vessel Code. In addition, the piping and valves of these systems must meet certain additional requirements imposed by the General Electric Company [illegible] specifications. The intent of these additional requirements is to provide piping systems of such quality which are at least equivalent to the reactor vessel to which it is attached. As with the vessel, the piping system components have a 40 year design life.

An ASME Boiler and Pressure Vessel Section I, Power Boiler classification is used for the piping system since Section III, Nuclear Vessels, specifically exempts itself from jurisdiction over piping systems. Section I is used for this specific plant because it is to be located in the State of Illinois whose boiler inspection laws do not recognize the ASA Code for Pressure Piping. The National code bodies are working on a Nuclear Piping Code which, when completed, will serve as a companion to the ASME Section III Nuclear Vessel Code. Moreover, these bodies have agreed that when the Nuclear Piping Code is issued, piping which is designed and constructed in accordance with it will be stamped with a Section III stamp. In the interim, piping systems are designed and constructed to meet the requirements of Section I, or where permitted by law, are designed and fabricated to the ASA B31.1 Code for Pressure Piping plus additional requirements which permit it to be classified as State Specials.

The piping system must meet the fabrication requirements of Chapter 6 of the ASA B31.1 Code for Pressure Piping. These requirements include an analysis of the expansion stresses and flexibility. The prescribed method imposes a stress reduction factor of 1.0 for all full temperature cycles over the expected life of 7000 or less. Since the expected number of full temperature cycles of the plant is on the order of 100, the unity stress reduction factor is appropriate. There are no other life time design cycles which need be considered for piping systems except in the area of branch connections through which flow of water at a markedly different temperature may occur. Where this occurs, a complete thermal stress analysis is made similar to the manner in which vessel nozzles are analyzed. This stress analysis is made by the methods prescribed by Section III of the ASME Boiler and Pressure Vessel Code. The allowable stress rules of Section III must be met. Thermal sleeves are utilized in such connections whenever the calculations establish their need. As with the vessel, first-of-a kind configurations are analyzed and the results are used to justify the same arrangement in subsequent plants when geometric symmetry exists and the expected design conditions are the same.

The following cycles are considered in the reactor vessel design:

Cycle	Number
Start-up, 100°F/hr.	120
Shutdown 100°F/hr. (Including Steam Space Spray and Flooding)	119
Blowdown 1000 psig to 160 psig in 10 min.	1
Reduction to 70% Power	10,000
Reduction to 50% Power	2,000
Rod Worth Tests	50,000
Feedwater Heater Loss	80
Feedwater Flow Loss	80
Scrams (from 1000 psig)	200
Turbine Trip	40
Pressure Excursion to 1250 psig	1
Pressure Excursion to 1375 psig	1
Hydrostatic Test 1562 psig	3
Hydrostatic Test 1250 psig	130
Core Spray Nozzle Flow	10
CRD Return Water Stoppage & Re-initiation	250
CRD Housing Cooling Flow Stoppage and Re-initiation	50
Sudden Start of Recirculation Loop	10
Back Flow in Recirculation Outlet Nozzle	10
Core Flood Nozzle Flow	10

The General Electric Company purchase specifications for the piping and the valves impose the following requirements over and above Code Rules:

1. Full penetration welds on all attachments and reinforcements.
2. All inspection, test and fabrication procedures must be submitted to General Electric for approval.

3. Liquid penetrant testing of all longitudinal and circumferential welds and the entire surfaces of valve body castings.
4. Check analysis of all fitting materials.
5. Backing rings with their inherent corrosion trap and incipient crack are prohibited. All circumferential joints must use consumable insert rings which result in a smooth inner bore, achieving the equivalent of a double butt weld.
6. Special bevel and radius requirements on all weld preparations to insure highest quality welds.
7. Liquid penetrant testing of all weld seams of fabricated fittings.
8. Complete radiography of all welds, cast fittings and valves.
9. Socket welded fittings are limited to 2-1/2" and smaller. (Section I allows 3").
10. Ferrite control of all austenitic stainless steel welds and castings.
11. ~~Controls are established for descaling, cleaning and heat treatment~~ of pipe, fittings and valves.
12. Heat treatment of cast austenitic fittings and valves is required initially and after major repairs.
13. An allowance for shrinkage is required on length of austenitic piping to reduce the weld shrinkage stresses.
14. Heat treatment of austenitic piping is required after hot forming operations. In addition, all surfaces must be liquid penetrant tested after forming operations.
15. Acceptance standards for liquid penetrant testing are more restrictive than code requirements.
16. All drawings and code calculations of austenitic piping and all valves must be approved by the General Electric Company.
17. Liquid penetrant or magnetic particle examination of all bolting.

In addition, it is General Electric Company's practice to include the major piping systems in the seismic analyses of the plant by outside consultants which have national stature in this field.

As stated in the Plant Design and Analysis Report on page IV-2-2, the casing of the reactor recirculation pumps is designed in accordance with the requirements of ASME Code Section III Nuclear Vessels Class C, plus certain additional requirements imposed by the General Electric Company purchase specifications. As with the piping and valves, the intent of these additional requirements is to provide assurance of a pressure containing component of such quality which is at least equivalent to the reactor vessel. As with the vessel, and the rest of the primary system, the design life is 40 years.

The recirculation pumps are classified as machinery, and as such are specifically exempted from the jurisdiction of any section of the ASME Boiler and Pressure Vessel Code or of the ASA Code for Pressure Piping. The Standards of the Hydraulic Institute are the only standards which are really applicable; however, they are more cogent to the testing and performance of the pump and consequently provide little or no guidance in the areas of casing quality and structural integrity. Therefore, in order to assure the General Electric Company and the applicant that the pump casing is an adequate pressure container, the General Electric Company insists that the pump designers utilize the allowable stresses and fabrication rules of the ASME Boiler and Pressure Vessel Code, Section III Class C. This Class is used because the pump casings do not experience the pressure transients and particularly the temperature transients that the reactor vessels and certain piping connections experience, and therefore it is not necessary to make the cyclic analysis required by Class A of Section III.

The thermal inertia of the system is so large that the pumps do not experience the rapid thermal transients that other portions of the system do. The use of Class C is a conservative approach because the allowable stresses of Class C are lower than those of Class A.

The General Electric Company purchase specifications for pumps impose the following requirements over and above Code Rules:

1. 100% radiography of all pressure containing parts to Class 2 standards and of all welds.
2. Liquid penetrant testing of the entire surfaces of the pressure containing parts and all welds.
3. Submittal for approval of all inspection, test and fabrication procedures.
4. Liquid penetrant or magnetic particle examination of all bolting which join pressure containing parts.
5. Solution heat treatment and ferrite control of all pressure containing castings.
6. Full penetration welds on all attachments and reinforcements.
7. Controls are established for descaling, cleaning and heat treatment of castings.
8. Heat treatment of castings after major repairs.
9. Acceptance standards for liquid penetrant testing are more restrictive than code requirements.
10. Drawings and code calculations must be approved by the General Electric Company.

QUESTION

G-2 In view of the influence of in-service inspection and testability on the vessel and primary system design, please provide the inspection and testability criterion for the primary system and its bases. In particular, discuss the proposal to limit vessel hydro-tests during the service life to 80% of design pressure.

ANSWER

Definitions: For clarification purposes, the applicant desires todefine "in-service" as that period in the life of a nuclear generating unit for which an operating license is obtained, i.e., after construction and before retirement. Therefore, in-service is applicable to the reactor, the primary system and the unit commencing with the issuance date of the operating license and has no bearing upon whether the licensed unit is operating or is not operating.

For further clarification, in-service inspection and testability can be subdivided further into (a) in-service surveillance and (b) in-service testing.

(a) In-service surveillance can be accomplished while the unit is operating by the study of operating records and performance of operating tests. Operating records can be obtained from instrument logs or, in the case of the reactor pressure vessel material, from "surveillance samples" of the actual materials of construction of the vessel, strategically placed in the vessel prior to the start of operation and removed during outages for physical tests to provide specified "in-service" condition information. Operating tests can be carried out, in many cases, without interrupting or stopping operation of the unit.

(b) In-service tests may be performed during outages, i.e., when the unit is not operating. These tests, made to demonstrate the pressure integrity and operational condition of the parts of the various systems will also demonstrate these characteristics of the systems and the unit. For example, specified welds in the recirculating water, feed-water and steam piping systems can be uncovered and visually inspected for external cracks or other signs of distress and visual inspection can be augmented by ultriasonic, dye-penetrant, magnetic particle or gamma radiographic examination, if in the opinion of the inspector, use of these non-destructive testing methods is warranted. Pumps can be run to check their pressure integrity and operational ability and valves can be pressure tested and operated to gain similar information .

Specified areas of the inner and outer surfaces of the reactor vessel can be visually inspected, directly or remotely using optical aids and this visual inspection can be augmented with one or more of the previously named non-destructive test methods if the inspector thinks it is desirable. The previously-mentioned (under (a) above) tests of the pressure vessel material surveillance samples are intended to supplement and augment the information gained by direct examination of the vessel and reduce the need for direct examination to a minimum.

Criteria

1. In-service testing will be performed at specified intervals.

2. Maximum use will be made of the design and manufacturing staffs and facilities of the organizations furnishing and installing the parts of the various systems to specify and assist in the performance of the necessary tests throughout the service life of the unit.

3. In-service tests will be designed and performed to demonstrate the functional ability and integrity of the operating parts of the systems and the pressure integrity of the systems in their entirety or of those affected parts when their function in the unit's operation requires pressure integrity.

4. Records of the results of the in-service inspections will be prepared, studied and kept available for study by others so that trends can be recognized as they develop during the service life of the equipment.

Bases

1. In-service inspection and testing will be carried out at specified intervals to assure safe future operation. The development of records of results from these tests will provide us with a basis for forecasting the continued safe operation of the unit.

2. The use of the equipment manufacturer's and installer's, staffs and facilities to develop and perform in-service inspections with our engineers insures their continuing interest in the service life of the equipment as well as assuring us of obtaining the benefits of the latest developments in the various arts of manufacturing, inspection, installation, etc., contributing to the equipment at our time of purchase and throughout its life. It will also be beneficial to the manufacturers and installers as they will be provided with information regarding the performance of their equipment throughout its operating life which will help them to design better equipment for future units.

3. The in-service tests necessary for safety assurance, not only for ourselves, but also for the various regulatory agencies will have to be carefully considered during the design of the units so that necessary access, readily removed and replaced insulation and shielding and significant numbers of points of inspection can be provided. Further, piping systems will have to be provided to furnish facilities for in-service and in some cases operational demonstration of the pressure and operational integrity and ability of selected pumps and valves.

4. The development and recognition of trends in operating performance of equipment is of major importance for planning purposes. Most of the unit's parts are not shelf items and maintenance and replacement must be carefully planned for the future to maintain maximum availability and as a corrollary, safety.

   The same importance, but with safety foremost, attaches to the recognition of trends developed as a result of in-service inspection of the parts of the systems which comprise the unit for pressure integrity. The excellent design, fabrication, manufacturing and erection efforts that will contribute to the superior initial condition of these systems must be continued throughout service life of the unit.

Proposed Hydro-Tests Pressure

The design of the Dresden 1 system required the use of 1000 psi as the nominal operating pressure and the early state of the nuclear art dictated the choice of 1250 psi as the design pressure. Although this established a much wider range than had been considered normal in fossil equipment, it gave ample room for changes that might be dictated by operating experience.

Proposed Hydro-Tests Pressure (cont'd)

When testing the system after an outage which has involved removal and replacement of the vessel head or another pressure containing closure, it is reasonable to use the operating pressure (1000 psi) as the hydrostatic test pressure to test for tightness. This use of the nominal operating pressure (1000 psi) as a hydrostatic test pressure for tightness, before returning the unit to service after an outage, has been accepted by the State's and the Insurance companies' Boiler Inspectors because the 1966 edition of the Boiler Safety Act and Boiler Rules and Regulations of the State of Illinois, Section 7, Part 6, Page 51 "Hydrostatic Pressure Tests" under "Note" reads:

> "Note: When hydrostatic test is to be applied to existing installations, the pressure shall be as follows:
>
> (a) "For all cases involving the question of tightness the pressure shall be equal to the release pressure of the safety valve or valves having the lowest release setting."

This is the background of the proposal to limit vessel hydro-tests during the service life to 80% (1000 psi/1250 psi = 80%) of design pressure.

The inspector and the operator performing a hydrostatic test for tightness during the service life of the vessel are only concerned that the vessel and its parts do not leak during operation. Safety valves are set to relieve at pressures no lower than the maximum working pressure (1080-1120 psi for Dresden 1) or design pressure. Demonstrating tightness does not require raising the pressure of the equipment so high that the safety valves operate. Further, a great deal of damage can be done to the seats of the safety valves if foreign objects are on the seating surfaces and the valves are gagged (mechanically restrained) so that their seating surfaces are overly jammed together. In the case of nuclear units, there is an additional feature which makes discharge of the safety valves undesirable: it is the fact that the safety valves discharge into the containment and the clean-up after such a discharge would be very difficult and expensive.

> Note: Dresden Unit #1 safety valves were tested and set on a fossil fuel boiler at Joliet Station to avoid discharging them into the containment at Dresden. At present, they are being tested and set using nitrogen in the test set-up outside the containment at Dresden. The use of the test installation at Joliet Station is now limited to setting blow down rates on repaired or rebuilt valves. All of this testing has been carried out with the approval of the Insurance companies' and the State's Boiler Inspectors.

QUESTION

G. <u>System Analysis (Continued)</u>

3. Provide the design basis energy removal capability for the shutdown cooling system. If the shutdown cooling system takes water from only one recirculation loop as indicated in the drawings, provide the basis for this design change.

ANSWER

The design basis energy removal capability for the shutdown cooling system is equivalent to the decay heat being generated in the core at the time the shutdown system is put in operation. This decay heat value is approximately .75 - .80% of full reactor power. The system is designed to be capable of removing not only the decay heat generated in the core, but also the sensible heat which is present in the vessel, piping, and the primary fluid.

The shutdown cooling system volume flow requirements for the Quad Cities units permit the use of a single suction line of adequate diameter. This is not considered a design change, as other General Electric Boiling Water reactors have this same feature.

QUESTION

G. System Analysis (Continued)

4. Discuss the reason for the lower control rod worths specified for this plant.

ANSWER

It is assumed that the question is concerned with the control rod system worth entered in the reactivity balance in Table III-1-1 in the Plant Design and Analysis Report. This shows a control rod system worth of -.17 Δk. Previous tables, such as the one submitted for Dresden Unit 3 indicated a system worth of -.18 Δk.

Reactivity balances are not unique representations of the system because the reactor conditions assumed can affect the magnitude assigned to each Δk increment. The design process does not use, or result in a reactivity balance. Balances are, therefore, only a guide.

The change in the system worth is a direct result of the change in the basic fuel k_∞. Since there has been no physical change in the control blades, the effect of the system expressed as $\Delta k/k$, to a first order approximation, is independent of the basic k_∞. Thus, if the basic k_∞ decreases, the Δk due to control will also decrease to maintain a constant $\Delta k/k$.

It should be noted that the design shutdown margin is unaffected by this change.

QUESTION

G. System Analysis (Continued)

5. Discuss the ability of the systems provided for core cooling and inventory control to provide protection for the core over the entire spectrum of primary system breaks up to the design basis break.

ANSWER

As noted in previous applications of Commonwealth Edison Company for construction permits and facility licenses, the Dresden Units 2 and 3 (AEC Docket Nos. 50-237 and 50-249) and the Quad Cities Station Units 1 and 2 (AEC Docket 50-254) are substantially similar in design. Each unit will utilize a single cycle, forced circulation, boiling water reactor furnished by the General Electric Company. Amendment No. 5 to the Plant Design and Analysis Report for Dresden Unit 3 dated August 12, 1966, presents a description, discussion, and analysis of certain provisions for emergency core cooling which are designed to assure adequate core cooling over the full spectrum of postulated primary system loss of coolant accidents. Those systems have also been incorporated into the design of Quad Cities Station Units 1 and 2 and include the following with reference to the appropriate section of the Plant Design and Analysis Report for Quad Cities Station Units 1 and 2:

Item	Section Reference in Unit 1 and 2 Plant Design and Analysis Report
High Pressure Coolant Injection System	VI-9.0
Low Pressure Coolant Injection System	VI-7.0
Core Spray Systems	VI-6.0
Primary System Relief Valves	IV-3.0
Emergency Coolant Supply System	VI-10.0

An exception to the above is the provision for use of the isolation condensers in Dresden Units 2 and 3, and the reactor core isolation cooling systems in the Quad Cities Units 1 and 2. However, the performance requirements with respect to core cooling are the same for the two different systems.

The integrated ability of these systems to protect the core under postulated coolant loss conditions has been fully discussed in the aforementioned amendment No. 5 for Dresden Unit 3. Such analyses and discussions are equally applicable to Quad Cities Station Units 1 and 2.

QUESTION

G. System Analysis (Continued)

6. Discuss design provisions to limit leakage of the vessel shroud due to thermal stresses during core flooding and thus maintain the integrity of the "inner vessel".

ANSWER

Two types of joints connecting the jet pump diffusor to the shroud support are being considered to insure that any potential leakage is within design limits. That is, the leakage around the seal will be equal to or less than 3,000 g.p.m. when the water level is at the two-thirds core elevation.

One joint incorporates a welded construction utilizing a flanged stainless steel insert welded to the hole in the shroud support. In the event that some separation between the insert and shroud support would be initiated during core flooding, the flange would limit the leakage path such that the total integrated leakage would be within design limits. The connection of the jet pump to the insert would be at a sufficient distance from the shroud-insert junction that any deformation of the shroud support plate would be sufficiently attenuated at the connections to minimize tearing.

The second method of connection being considered incorporates a mechanical joint allowing relative motion between shroud support plate and jet pump with a minimum of restraint. One method by which this can be accomplished is threading the outer diameter of the jet pump and threading two collars one above and one below the shroud support plate effectively clamping the jet pump diffusor to the shroud support plate. (The jet pump would not be threaded to the shroud support plate.) In this manner, relative motion between collars and shroud support plate can occur without a gross distortion of the jet pump body.

Both of the above designs can insure that the leakage is within design limits if the core flooding system is activated. The mechanical joint is as of this date the preferred design. Analyses of the static and dynamic structural effects of the mechanical joint on the jet pump assembly are needed before a final selection can be made.

QUESTION

G. System Analysis (Continued)

7. Discuss the use of a single header beneath the suppression pool as a water source for several core and containment cooling systems. Are the reliability of the several systems compromised by use of this single header? What design criteria and protection are specified for this component? Provide the reference design of the screened intakes to this header and discuss the basis for the number of intakes versus the number of systems served.

ANSWER

The header beneath the suppression pool is considered a logical extension of the pressure suppression chamber and, as such, must meet the same design criteria, surveillance, and testing as the primary containment. (See Section V-1.0).

The suction piping is constructed of heavy-walled pipe firmly supported from the lower portion of the suppression chamber at fifteen positions around the circumference. Maximum protection of the suction pipe is afforded by its physical location; i.e., adjacent to the suppression chamber in a room containing no high pressure piping or mechanical equipment.

A single header design with three penetrations to the suppression chamber was chosen to provide selection flexibility of specific equipment suction points. It is considered that the reliability of each connected cooling system is not compromised by this arrangement in that the suction header is strong, located in a highly protected area, and subjected to the same design, inspection, and periodic testing as the adjacent suppression chamber.

Three intakes, 120° apart, are provided in the header. Stainless steel strainers located on each intake are designed to screen out particles greater than 1/8 inch diameter. Each of the three intakes will be designed to handle the total suction requirements of the low pressure coolant injection system and one core spray system at a total maximum entrance head loss across the screen of one foot of water. The total designed suction capacity of the header system is over twice the total flow demand from the connected cooling systems.

QUESTION

H. Accident Analysis

1. Provide an analysis of the consequences of violation of the secondary containment by tornadic action as a function of time after the design basis accident.

ANSWER

Based upon data relating to variation in tornado frequencies in specific locations in the United States, compiled by the U.S. Department of Commerce - Weather Bureau, it is estimated that the mean recurrence interval for a tornado striking the Quad-Cities site is of the order of 500-700 years. This low probability of occurrence of a damaging natural phenomena, coupled with the extremely low probability of a postulated mechanical failure of a recirculation pipe in the drywell, results in an insignificant probability of simultaneous occurrence of both events at the same place at the same time.

However, the question has assumed that such an event occurs. If a tornado were to strike the reactor building, it has been calculated that damage would result to the superstructure above the operating floor. It is expected that the steel panels of the building might be torn away from the structural steel members of the building. The massive reinforced concrete walls below the operating floor would not fail. Thus the reactor building and standby gas treatment system would be partially or totally ineffective for fission product retention and elevated release. Several design basis accidents are postulated as a means of evaluating the effectiveness of the engineered safeguards for mitigating accident consequences.

Table I-6-1, page I-6-11 of the Plant Design Analysis for Quad-Cities Units 1 and 2 summarizes the maximum offsite doses from postulated accidents. For the design basis accidents these doses, representing maximum off-site exposures, range from 3.6×10^{-4} to 3.0×10^{-3} rems thyroid and from 8.6×10^{-4} to 8.3×10^{-2} rems whole body.

Based on calculations performed for Dresden, Quad-Cities, and other similar reactor plants, the effect of the stack and the standby gas treatment system is to reduce the thyroid dose by a factor of about 2000. Similarly the Stack reduces whole body dose by a factor of about 10 under postulated accident conditions. Thus for the accident case cited above the whole body dose for the worst case accident would be increased to approximately 0.8 rem, and the thyroid dose would be increased to approximately 6 rem. These values are still approximately a factor of 30 to 50 below the guideline values of 10 CFR 100.

QUESTION

H. Accident Analysis (Continued)

2. What is the maximum permissible radioisotope inventory in the waste storage tanks? Describe the environmental consequences of failure of the closure valves to the stack and the environmental consequences of a tank leak, resulting in a ground level release.

ANSWER

The total capacity of the waste storage tanks is 130,000 gallons and the maximum concentration is 10^{-3} μC/ml. Therefore, the maximum radioisotope inventory, assuming that all the tanks are full at the same time, would be about 4.9×10^{-1} curies.

The question of the mobility of the stored wastes is discussed on page VII-4-4, as revised on August 18, 1966, of the Quad Cities Station Plant Design analysis. If a leak were to occur in a tank outside the building, the liquid would be confined to the area of the tank by a retention curb. A drain leads from the retention pad to the discharge canal which is provided with a continuous water sampler. If that were to happen, the concentration of wastes in the canal would be less than 10^{-7} μC/ml.

The vents on all the storage tanks are not fitted with closure valves but have breather pipes which are equipped with high efficiency filters.

QUESTION

H. Accident Analysis (Continued)

3. Please provide an analysis of the physical consequences of a refueling accident wherein a fuel bundle is dropped into a just-critical array. We believe that an analysis of the potential effects on the secondary containment should be considered (not withstanding the small probability of the accident) because of the magnitude of the consequences if the containment were violated.

ANSWER

This hypothetical accident has been analyzed using the basic excursion analysis models developed at APED [1].

The neutron kinetics model indicates that the peak fuel enthalpy at the termination of the prompt power burst is approximately 400 calories per gram. This model also calculates that approximately 450 kg of fuel has enthalpies above 170 calories per gram. For this fuel, approximately 260,000 BTU must be transferred to the water to subcool the fuel to 100 calories per gram below the melting point. The rate at which energy is transferred to the water as a function of time was calculated using the fuel failure model and a characteristic thermal time constant for finely dispersed fuel.

A thermodynamic-hydrodynamic analytical model was then used to calculate the steam generation & water expulsion assuming that the energy released by the dispersed fuel was transferred uniformly into the water contained in the cell associated with the failed fuel pin.

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1. "Nuclear Excursion Technology", H. A. Brammer, R. J. Mc Whorter, J. E. Wood, Seminar to members of Licensing and Regulatory Staff and Advisory Committee on Reactor Safeguards, June 11, 1965.

Figure 1 shows the resultant water spillage following the nuclear excursion. The maximum surface velocity of the water in the reactor well is approximately 16 feet per second. This produces a water spillage rate of approximately 20,000 cubic feet per second.

At the termination of the hydrodynamic disturbance, approximately 3000 to 5000 cubic feet of steam at 40 to 60 psia will be in the core of the reactor. It is anticipated that the bulk of this steam will condense as it diffuses through the 40 to 50 foot water column remaining in the vessel and reactor well.

Based on this analysis, no structural damage to the vessel or refueling building is expected to occur.

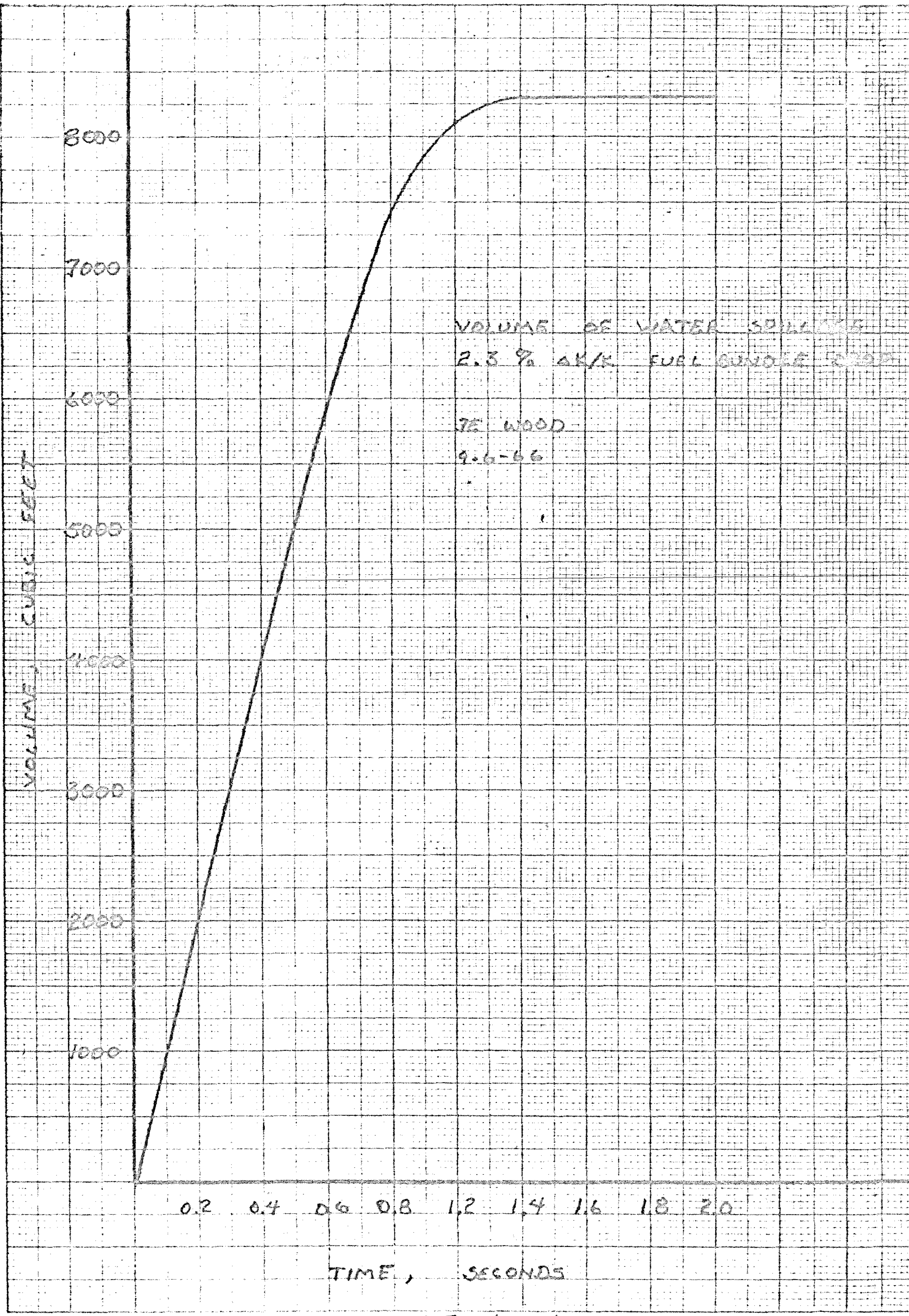
8000
7000
6000
5000
4000
3000
2000
1000
VOLUME, CUBIC FEET
VOLUME OF WATER
2.3 % ΔK/K FUEL BUNDLE
JE WOOD
9-6-66
0.2
0.4
0.6
0.8
1.2
1.4
1.6
1.8
2.0
TIME, SECONDS

FIGURE H-3-1

QUESTION

H. Accident Analysis (Continued)

4. Justify the identification of the height of the steam cloud after a steam line break with the BNL data which were for small explosions. What energy dissipation effects of the turbine building have been taken into account?

ANSWER

A copy of the BNL explosion test program or data resulting from such a program as referenced in the question is currently unfamiliar to the applicant. Until such data relating to small explosions have been obtained and analyzed, no comparison can be made of steam clouds and explosion products.

A previous discussion of the ability of a steam cloud to rise subsequent to release from the turbine building is contained in the Dresden Unit 2 (AEC Docket 50-237) Plant Design and Analysis Report, Amendment No. 2, page III-4-2, question III-4-C. In these evaluations, no credit has been taken for energy dissipation by or in the turbine building, or for plateout or washout of halogen in the turbine building.

QUESTION

H. Accident Analysis (Continued)

5. Discuss the ultimate capability of the reactor vessel to retain portions of the core which are released to the lower plenum as a result of partial ineffectiveness of the spray system. Could this quantity be increased significantly if the sacrificial shield were flooded, allowing heat transfer from the control rod drive thimbles? What amount of the core, including UO_2, could react before the containment reached its design pressure?

ANSWER

Even though the design basis requires the provision of engineered safeguards with both redundancy in equipment and technique for emergency cooling so as to preclude situations of partial ineffectiveness of emergency cooling for loss of coolant situations, evaluations of the primary system behavior in the more degraded partial effective safeguard condition were made.

In attempting to determine the tolerable percent core meltdown for a particular set of conditions, the following are the more important parameters which must be considered: Area onto which the melt would fall; thickness of melt, conductivity of melt; allowable vessel structure temperature; fraction of fission products remaining in the melted fuel; and the time after scram from an assumed full power operation.

Because of the difficulty of establishing a realistic mathematical model and lack of appropriate phenomenological information on which to base such a model, any estimate must necessarily remain speculative and highly qualified. The model examined is as follows: The molten core portion consisting of a mixture of UO_2 particles, zircaloy and stainless steel, falls to the bottom of the vessel into water at a temperature of 250°F. The bottom of the vessel is insulated, precluding effective vessel cooling. Heat is removed by conduction through the melt and through the surface film boiling coefficient under quasi-steady state conditions.

For this model and the stated assumptions, the interrelationship between percent of core melt and the volume of melt at the bottom of the vessel can be established. If no heat is transferred through the vessel structure on which the melt falls and if film boiling is always present on top of the melt (as will be the case for the heat fluxes of interest) a temperature drop across the thickness of the melt and the vessel wall temperature can be determined as a function of the melt power density. The power density in the melt is a function of the fraction of fission products remaining in the melt, and the time after scram. Thus, all the pertinent variables can be related in a manner to obtain a solution over a range of the possible values of the variables and the physical properties of the melt.

The amount of fission products remaining in the melt was estimated to be in the range of 50 to 80% and an effective melt conductivity was estimated to be between 2 and 6 BTU? Hr ft. for the Zircaloy, stainless steel, UO_2, mix. An allowable vessel wall temperature of 1200°F was selected as a reasonable limit. A time after scram between 1 and 4 hours was considered appropriate for a partially effective core cooling case.

Throughout the range of variables, it was determined that the probable range of in-vessel melt retention capability would be between 5% and 15% of the total core inventory.

Since the reactor vessels are well insulated, immersion in water does not have significant effect on the heat loses out of the vessel. Heat would have to flow by conduction through the vessel wall and then through the reactor vessel insulation or to the control rod thimbles and then to the water surrounding them. Since the thimbles are occupied by concentric cylinders with water spaces between them and because the portion of the core resting on the vessel must be below melting temperatures if vessel integrity is to be maintained, most of the fin effect will be due to the outside cylinder of the thimble.

The thimbles represent about 18% of the area under the head. Thus, even if their fin efficiency were as high as 200%, the net effective heat transfer area added to the vessel would only be (1-.18) + (.18x2)= 1.18, or 18% increase in area. This would be equivalent, then, to having roughly 20% of the bottom head exposed to the heat sink and the other 80% insulated. If this heat transfer process through the bottom head by fin effectiveness were as efficient as the heat transfer occurring out of the top surface of the melt, the amount of core melt that could be tolerated without vessel failure would be increased by only 20%.

A chemical reaction of UO_2-H_2O can occur under ideal conditions, but the kinetics of the reaction are relatively slow in the temperature range of interest. Markowitz[1] has examined the possibility of a UO_2-H_2O reaction and recognized that the oxidation of UO_2 by water was thermodynamically reversible and that UO_{2+x} could be reduced to UO_2 if hydrogen were present. He concluded that "-----While oxidation of UO_2 can proceed in an atmosphere of pure steam----a trace of hydrogen inhibits the reaction; a partial pressure ratio of hydrogen to steam as low as 10^{-3} can make the mixture reducing----", that is, as little as 0.1% of hydrogen in the steam will suppress the reaction completely. Essentially, the same conclusion was also reached by investigations at the Argonne National Laboratory.[2] However, even if the reaction were to occur, the maximum amount of hydrogen associated with the complete reaction of UO_2 to the end point oxidation state ($UO_{2.17}$) amounts to only an 8% increase of that associated with the zircaloy-water reaction for any given portion of the core. Conservatively assuming that the energy release is the same for both, the containment capability[3] curves would shift downward approximately 8% of the value shown. For an upper limit case involving a hydrogen release of two hour duration, the containment capability would drop only from 65% to 60%.

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1. Markowitz, J.M., "Internal Zirconium Hydride Formation in Zircaloy Fuel Element Cladding Under Irradiation." Report #WAPD-TM-351, Bettis Atomic Power Laboratory, May, 1963. pp. 26
2. Adams, R.M., Glassner, H., Reactor Development Progress Report, ANL-7115 October, 1965, pp. 81-83
3. Figure 108, Quad-Cities Plant Design Analysis, Volume II

QUESTION

H. Accident Analysis (Continued)

6. To what level could the containment be flooded before design pressure was reached due to compression of non-condensibles after a 25% metal-water reaction?

ANSWER

The design basis for the Quad-Cities Units 1 and 2 emergency core cooling systems is to prevent fuel clad melting for the postulated loss-of-coolant accident. This is accomplished by means of several engineered safeguards features with specific performance characteristics to achieve the design function over a complete spectrum of postulated break sizes. These engineered safeguards include the two core spray systems, the low pressure coolant injection system, the high pressure coolant injection system, automatic relief valve actuation, coolant water availability and emergency electrical power.

The results of calculations and experimental work provide a basis for the firm conclusion that fuel cladding can be maintained below melt temperatures with the above emergency core cooling systems, and that the extent of metal-water reaction under accident conditions would be approximately 0.5%. As noted on page VI-6-2 of the Plant Analysis Report for Quad-Cities Units 1 and 2, this metal-water reaction would result in a hydrogen concentration in the primary containment of approximately 2.5%, and would provide a minor contribution to containment pressurization.

A 25% metal-water reaction is not considered as the design basis for the containment because of the core cooling systems provided for this station as noted above. However, if it is postulated that an amount of hydrogen equivalent to a 25% metal-water reaction were introduced in the containment, the drywell could be flooded to within about 10 feet of the bottom of the reactor vessel as shown in Figure H-6-1 This assumes isothermal compression because the time required to compress the gases is sufficiently long that most of the heat of compression will be dissipated. The water temperature will be approximately 100°F. Neither decay heat nor stored vessel heat significantly affects the water temperature. At this water elevation, the stresses at the bottom of the suppression chamber would reach the design pressure of 62 psig.

In the absence of hydrogen, consistent with the core cooling systems provided, the containment can be flooded to about five feet above the bottom of the reactor vessel.

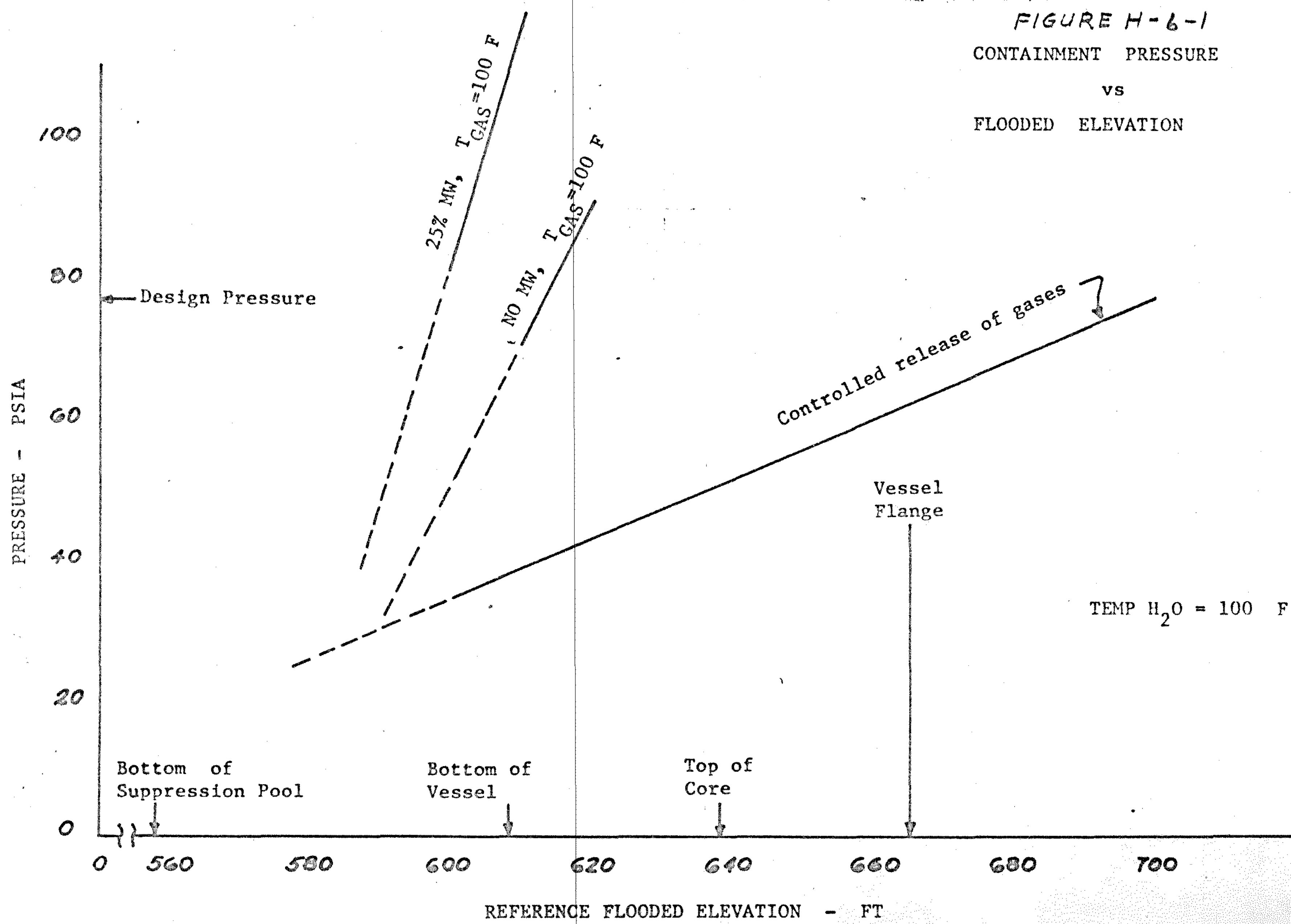
If the containment non-condensibles were released in a controlled manner, the water level in the drywell could be raised to flood the reactor core without exceeding yield point stresses. Hydrostatic loading is recognized as being a different type of load than a simple gas pressure loading and this is considered in the design specifications for the containment.

FIGURE H-6-1

CONTAINMENT PRESSURE

vs

FLOODED ELEVATION



QUESTION

H. Accident Analysis (Continued)

7. Discuss the magnitude of nuclear excursions which might: (a) impair the effectiveness of core quenching systems either by movement of the core or damage to piping or shroud or (b) cause vessel rupture. Could the excursion which caused damage to the vessel internals result in a loss of primary system integrity?

ANSWER

It should be understood that a preventative design approach is taken with respect to nuclear excursions. A limitation has been placed upon the amount of excess reactivity and the rate at which it can be added to the core, rather than designing to contain a specific nuclear excursion. Thus, sufficient safety barriers (housing supports, rod worth minimizer, velocity limiters, plus operator procedures) have been provided to preclude an excursion of such magnitude that could cause serious mechanical damage. However, these considerations have been in the interest of defining specific functions, i.e., the spectrum of excursions, energies, and pressures, and probability of mechanical failures. Specific containment design bases and concepts for these exercises have not been set for this type of accident.

With this in mind, the following values have been established: Given a 2.5% δk rod at hot standby and withdrawal velocity of 5 feet/second, the resulting excursion yields an energy release of 4000 MW-S, and a peak fuel enthalpy of 220 cal/gm and only minor fuel damage. As previously reported*, the resulting pressure rise across those reactor internals critical to the core cooling systems was only a few psi. Therefore, the core cooling systems would remain unimpaired.

If a higher reactivity addition rate was considered possible, the energy release would approach 7600 MW-S, with a peak fuel enthalpy of 340 cal/gm and probably a significant pressure rise. The behavior of the coolant (transient heat transfer, vaporization, pressure pulses, water slugs) in this transitional range is not well known and would require additional analytical and experimental work. A small amount of the fuel may be damaged

*Plant Design and Analysis Report, Dresden Units, Amendment 5 and some dispersed, however, not resulting in prompt dispersal or a fine fragmentation. The rate of pressure rise is more rapid than in the 4000 MW-S excursion, but not in the realm of a pressure pulse or water slug. The pressure rise for this postulated condition cannot be accurately calculated using presently available models. Some internal structures may be damaged if significant fuel dispersal were to occur, however, the primary system remains intact.

However, with no primary system damage, supplementary core cooling is not required to limit further core damage. The LPCI and HPCI systems remain capable of injecting water into the primary system in addition to a core spray operating capability.

If, for example the heat produced from the latter excursion were to be transmitted to the water through the fuel timé constant, i.e., insignificant dispersal) calculations indicate that the peak transient pressures occurring across any of the structural components are under 15 psi, more than a factor of ten under the lowest capability of the internal structures. (Table II 3.2-1).*

An excursion producing an energy release of 21,000 MW-S and a peak fuel enthalpy of 470 cal/gm would result in a prompt dispersal of a small amount of finely divided fragmented fuel, producing an undetermined but high pressure transient rate and probable damage to the internal structures and possible the primary system as well. The maximum pressure rise rate, determined by the peak fuel enthalpy in this excursion range, is not dependent on total energy release. However, severe vessel movement is not probable in view of the large forces required (Figure II 3.3.8). Hence, it is probable, since the piping may remain intact, that the core cooling systems retain significant capability to cool the core.

Additional analyses in this area is being performed. The staff will be advised of any significant conclusions.

QUESTION

H. Accident Analysis (Continued)

8. If one class of instruments (such as level detectors) did not function during an intermediate sized break, could the initiation of low pressure systems be delayed? What assurance is there that vital instruments will work on diesel power?

ANSWER

It is improbable that instruments that are a part of the reactor protection system would not function as required since such instruments must comply with the design basis for that system. However, as presented in Amendment No. 5 of the Dresden Unit 3 Plant Design and Analysis Report, Table II.3.9-1 Intermediate Breaks, the time delay for initiation of the low pressure systems is a function of the break area. To summarize herein, the following are the delay times:

Break Area - ft^2	Additional Time to Uncover Core, Minutes
0.03	30
0.06	20
0.15	6

The actual sensing of low-low level is a prerequisite to initiation of the core and containment spray systems. Thus, if all level switches are assumed to fail in the "normal" condition, then low pressure alone could not initiate spray action. However, manual operation would not be affected, and could be used to actuate the systems.

The only instruments associated with the initiation of the core and containment spray systems are relays which inherently have a very broad voltage and frequency tolerance. The sensing devices are direct acting mechanical devices, thus no power supply sensitive equipment is involved. Furthermore, it is most probable that the relays involved will be powered from the station battery rather than the emergency diesel power.