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Report to ACRS

THREE MILE ISLAND UNIT 2

U.S. Atomic Energy Commission
Division of Reactor Licensing

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ABSTRACT

The results of our safety evaluation of the Three Mile Island Unit No. 2 are presented in this report. The nuclear steam supply system, engineered safety features, and reactor building are similar to Unit No. 1, Oconee, Crystal River, Rancho Seco, and Russellville reactor plants. Because of the similarity of the Unit No. 2 plant design with these plants, the safety review was primarily directed to those areas which differ from previously approved similar plants. These areas include the application of grouted tendons, the type of control rod drives, aircraft protection, and areas of continuing concern regarding the pressurized water reactor design and construction. Unit No. 2 will have an ultimate power level of 2772 Mwt compared with an ultimate power level of 2568 Mwt for Unit No. 1. We conclude that these matters have been adequately accommodated in the design of Unit No. 2.

The significant safety considerations that will require a continued review following issuance of the construction permit are delineated in Section 19.0. On the basis of our review, we have concluded that the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

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1.0 INTRODUCTION

The Jersey Central Power & Light Company (JCPL) submitted an application on April 29, 1968, for a construction permit and operating license for Unit No. 2 of the Oyster Creek Nuclear Power Station (OCNPS-2). The proposed plant was to have been located adjacent to the existing Unit No. 1 on an 800-acre site in Lacey Township, Ocean County, New Jersey, approximately two miles south of the community of Forked River. On March 10, 1969, Amendment No. 6 was submitted jointly by the Jersey Central Power & Light Company and the Metropolitan Edison Company as co-owners indicating a change in facility sites to the Three Mile Island site about 10 miles southeast of Harrisburg, Pennsylvania. The station was redesignated as the Three Mile Island Unit 2 (TMI #2).

The TMI #2 design is the same as the originally proposed Oyster Creek Unit 2 except for changes incorporated into the TMI #2 design to account for site-related matters such as requirements for aircraft protection and the use of cooling towers as the primary heat sink. The nuclear steam generating supply system will be a two-loop Babcock and Wilcox Company pressurized water reactor. TMI #2 will operate at a core thermal power level up to 2452 (845 Mwe) and will have an ultimate power level of 2772 Mwt. All core physics, thermal and hydraulic characteristics have been evaluated for the 2452 Mwt power level. Engineered safety features, heat removal, and waste handling equipment have been sized based upon the 2772 Mwt ultimate power level. In addition, all accidents which result in releases of radioactivity have been analyzed based on the ultimate core power level of 2772 Mwt.

The architect/engineer will be Burns & Roe, Incorporated. United Engineers and Constructors will be the construction manager. MPR Associates will be used as consultants to assist in executing and auditing the TMI #2 quality assurance and quality control programs. Pickard and Lowe Associates will provide design review consultant services for TMI #2. Schupack and Associates will be structural consultants on the design of the reactor containment building which will be a prestressed, post-tension reinforced concrete structure using grouted tendons. Gilbert Associates, Incorporated will provide architect engineer service for design of the cooling towers and consultant service on site-related engineering including aircraft protection design.

The nuclear steam supply system, engineering safety features, and reactor building are similar in design to the Oconee Nuclear Station, Crystal River Nuclear Station, Rancho Seco Nuclear Station, Three Mile Island Unit No. 1 Nuclear Station, and the Russellville Nuclear Station which have already been issued construction permits.

Because of the similarity of the TMI #2 design with the above previously approved plants, our safety evaluation of the design was principally directed toward safety aspects for multiple plants, shared facility features, and continued review of concerns from previously reviewed plants including radiolytic hydrogen formation and quality assurance and control.

On May 1, 1969 the applicants submitted an exemption request to permit construction of the tendon access gallery prior to issuance of the TMI #2 construction permit. A separate report covering the tendon access gallery exemption was submitted previously to the Committee. A chronology of the action taken on this application to date is presented in Appendix A.

2.0 SITE CHARACTERISTICS

The Three Mile Island Nuclear Station Unit No. 2 (TMI #2) will be located adjacent to Unit No. 1 on Three Mile Island on the Susquehanna River approximately 10 miles southeast of Harrisburg, Pennsylvania (1960 population 79,697). The site is located in Londonderry Township of Dauphin County, Pennsylvania.

2.1 Population Distribution

The nearest residence to the site is 2200 feet east of the containment building. The exclusion distance for the Three Mile Island site is 2000 feet. Based on the combined population of Middleton-Steelton communities (22,450) with their nearest boundary at 3 miles, the applicants have proposed a low population zone radius of 2 miles, the same as approved for TMI #1. The 1967 population within the proposed low population zone was about 2300. We believe appropriate protective measures could be taken on behalf of residents within this radius in the event of an accident.

2.2 Meteorology

In the absence of specific onsite measurements of meteorological characteristics at the site, we have applied the staff model of Pasquill "F" and 1 meter/second wind speed in calculating the two-hour accident doses at the exclusion radius of 610 meters (2000 feet).

The applicants have estimated site boundary atmospheric diffusion values for the two-hour post-accident period which are about 85% of the staff values. For calculating doses at the low population zone distance of 2 miles, we have applied the same meteorological parameters used by the applicants. The parameters used are based on Pasquill "F" conditions with absolutely invariant wind of 1.0 meter per second for the first 12 hours. The same wind direction was assumed for the following 6-1/2 days but with Pasquill "F" and 1.0 meter/second 50% of that time and Pasquill "D" with 4.0 meters/second the remaining 50%. For this second time period the wind direction was averaged over the 22-1/2 degree sector. For the following 24 days the wind was assumed to blow in the same 22-1/2 degree sector 50% of the time with 5% in the Pasquill "C" category with 2.0 meters/second wind speeds; 33% in the Pasquill "D" category with 4.0 meters/second wind speed; and 12% in the Pasquill "F" category with 2.0 meters/second wind speeds. Wind direction was averaged over the same 22-1/2 degree sector. Our consultants on meteorology, ESSA, reviewed the meteorology data submitted with the TMI Unit No. 2 application and have commented that the meteorological documentation and conclusions are identical to those which were presented in the application for TMI Unit No. 1, and that their comments on Unit No. 1 dated June 19, 1967, are directly applicable to the Unit No. 2 application. In their comments on the Unit No. 1 application, ESSA concluded that the applicants' diffusion parameters used in the dose calculations are reasonably conservative. The ESSA comments have been forwarded to the Committee. The applicants have a program of onsite measurements to determine wind speed and direction and stability characteristics of the site. Data collected in this program will be used to make a final determination of the diffusion characteristics of the site, and will be evaluated at the operating license stage of review.

2.3 Geology

Core borings show that the foundation conditions in sound bedrock underlying the site should be adequate for the proposed facility. There do not appear to be any extraordinary geologic engineering problems associated with construction of the TMI #2. We have been informed by our geological consultants in the USGS that their conclusions based upon the TMI #1 site geology review are unchanged and no adverse geological conditions exist at the site.

2.4 Seismology

Based upon the review of seismic history at the Three Mile Island site, the operating basis earthquake (OBE) and design basis earthquake (DBE) were selected to have horizontal ground accelerations of 0.06g and 0.12g, respectively. The design basis for TMI #2 will consider the above OBE and DBE with a vertical acceleration equal to 2/3 the horizontal acceleration.

Our seismic consultants (USC&GS) concluded that the ground accelerations considered for the site are adequate on their review of the seismic history of the site and the surrounding area. We have informed the applicants of our requirements regarding the installation of at least one strong-motion accelerograph. They have indicated that they would comply with our requirements.

2.5 Hydrology

The site is located on an island in the Susquehanna River just above the York Haven Dam. Two hyperbolic closed-cycle cooling towers will be used to remove the heat from the main condenser cooling water. Service water for the heat removal of other normal and emergency equipment is supplied from the river. There is an additional cooling tower that is used to reduce the service water and cooling towers blowdown water temperature before being returned to the river. A low river flow of 1700 cfs, equivalent to failure of the downstream York Haven Dam, would still provide both units of nuclear service pumps a supply of 430 cubic feet per second. This minimum flow available to the nuclear service pumps is five times the flow required for safe shutdown of both units

During our consultants' (USGS) review of the Three Mile Island site hydrology, a check was made with the Baltimore District of the U.S. Corps of Engineers to obtain verification of the probable maximum flood (PMF) discharge of the Susquehanna River at the site. The Corps of Engineers indicated they have just completed their calculation of the PMF discharge for the Susquehanna River basin and the PMF discharge would be about 1.75×10^6 cfs at the Three Mile Island site. This flow exceeds the preliminary value of 1.1×10^6 cfs that was used to determine flood protection requirements for TMI #1. We have discussed this with the applicants. The final value for the PMF will not be available until about June 1970. However, they indicated that the flood protection dikes and any equipment required for protection of the TMI 1 and 2 would be designed to be protected to the flood height which results from the PMF. In addition, other means could be used to protect against a larger PMF discharge if required. The bridges onto the TMI site would not be changed since they are already built. However, the applicants indicated if the bridge were lost, access to the site could be maintained by use of a helicopter or boat.

We will review the design during the post-construction review stage to assure that protection is provided against the flood height developed from the PMF.

2.6 Environmental Monitoring

Metropolitan Edison Company has been conducting a preoperational environmental monitoring program since January 1968. The program, as described in the TMI #2 application, has included samples of well and river water, river sediments and fish, and soil and vegetation from the site area. A total of 84 samples have been taken in the program. The Metropolitan Edison Company has met with the Health Department of the Commonwealth of Pennsylvania and has submitted their program to that agency for its review and recommendations.

We have received comments from the U. S. Fish and Wildlife Service regarding the TMI Unit No. 2 application. These comments have been forwarded to the applicants. We have discussed the comments with the applicants and they indicated that they would comply with the recommendations of the Service.

We have informed the applicants that additional information is necessary including the type and location of samples of aquatic biota from the Susquehanna River; the studies that have been made to determine critical radionuclides and critical food chain pathways to man; the use of Susquehanna River water for irrigation; and the possible need for monitoring of crops raised on irrigated land.

With respect to the sampling of aquatic biota, the applicants have informally advised us that an ecological survey has been underway for some time now to determine what aquatic life is present in the river and how these may be involved in the food chain to man. They have also indicated that the possible incorporation of radionuclides into food crops via irrigation water will be investigated. Results of the ecological survey and the study of irrigated food crops will be used to establish appropriate environmental monitoring procedures. These items are discussed in Amendment 10 to the application. On the basis of our discussions with the applicants and this information, we conclude that the applicants will have an acceptable environmental monitoring program. We will follow up on this aspect of the site review to assure that an adequate program has been established at the operating license stage of review.

2.7 Airports

The Three Mile Island site is located two and one-half miles from the eastern end of the single 8000-foot-long runway of the Olmstead State Airport. The Harrisburg-York Airport is located about eight miles WNW of the site.

During the review of the TMI #1 facility and s[illegible] applicants provided an analysis indicating the probability of an ai[illegible] the facility should be about 10^{-6} per year based on 40,000 take-offs and

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landings at the end of the runway nearest the site. This aircraft traffic is about four times the current aircraft traffic at the Olmstead State Airport. Because of the proximity of the site to the airport, we concluded that the effects of aircraft impacting on the facility must be included in the design bases for the Unit 1 facility. The same conclusion applies to the design of Unit 2. The design aspects are discussed in Sections 5.3 and 11.0.

Using the May 6, 1969 Proposed Criteria for Siting Power Reactor Facilities Near Airports, the TMI site would be subjected to the requirements of section 100.10 (b) 3 because the plant lies within one mile laterally along the flight paths. However, based upon these proposed criteria and an aircraft traffic of four times the 1969 traffic at Olmstead State Airport, the TMI site would require protection against an air strike of the DC-9 (98,000 lbs) aircraft instead of the 200,000-lb aircraft which is discussed in Section 11.0.

We have checked with the Chief Controller at the Olmstead Airport regarding the present air traffic at the airport. There has been no significant change in the frequency or type of aircraft using this airport which would affect the aircraft protection requirements established during the TMI #1 review. The airport usage is principally for light commerical aircraft. Heavy commercial usage of the large jet class planes is not anticipated. The following table lists the class and flight frequency distribution at the Olmstead Airport.

TABLE 2.7

OLMSTEAD STATE AIRPORT USAGE (1969)

Type Aircraft	Wt. (lbs)	Flights	Average* Daily Frequency
Boeing (707)	257,000	1/day	2
Boeing (727)	200,000	2-3/mo.	0.2
Douglas (DC-9)	98,000	8/day	16
Convair 580 (C-5)	58,000	26/day	52
Other	50,000	8/day	16

*Landings and Take-offs

Based upon our proposed criteria for aircraft protection requirements for sites near airports, we conclude that the TMI siting considerations must include the effects of aircraft impact.

2.8 Cooling Towers

The TMI #1 and #2 heat removal systems will not discharge facility heat loads into the river except under emergency conditions, or during the winter months when the discharge will be used to prevent icing at the river water intake structure. Normal heat loads of both Units 1 and 2 will be discharged to the atmosphere by two hyperbolic natural draft cooling towers per unit. The blowdown from the cooling towers and all secondary river water coolant will be circulated through an additional cooling tower prior to discharging to the river.

Operation of the additional cooling towers does not affect the safety of the plant.

A program is currently in progress to study the effects of large cooling towers on the local meteorology. This study is being made at the Keystone Fossil Plant, located in Chelocta, Pennsylvania. The facility has four cooling towers and a total plant capacity of approximately 1300 Mwe. We have discussed this program with the applicants and they have indicated that the program was initiated to determine the effect of fogging since it could affect operations at the Olmstead Airport. We shall follow this program during the operating license review to assure there are no new significant factors resulting which could affect plant safety or public health.

3.0 COMPARISON OF TMI #1 AND #2

The following table lists the major differences in system designs for TMI #1 and #2. In many instances the differences noted are primarily the result of design approaches taken by Gilbert Associates and Burns and Roe, the architect-engineers for Units 1 and 2, respectively.

TABLE 3.0

MAJOR DESIGN DIFFERENCES BETWEEN UNITS 1 & 2

	TMI #1	TMI #2
Turbine	Non-reheat (GE)	Reheat (West.)
Condensate Demineralizers	1/2 flow	full flow
Nuclear Service Cooling Pumps	3	4
Nuclear Service Coolers	4	2
Decay Heat River Water Coolers	2	0
Ultimate Core Power Level	2568 Mwt	2772 Mwt
Reactor Building		
Emergency Cooling Pumps	2 (river suction)	2 (booster)
Fan Coolers	3	5
Wall Thickness (feet)	3.5	4.0
Dome Liner Thickness (inches)	3/8	1/2
Tendons	Greased 170 wire	Grouted Strand
Channels over liner welds	Inaccessible welds	All welds
Access Openings (dia. feet)	22.3	23.0
Design Pressure* (psig)	55	60
Separate Spent Fuel Cask Pool	No	Yes
Auxiliary Building Aircraft Protection	Below ground floor	Complete

*For discussion on this matter, see Section 5.6.

4.0 UNIT NO. 1 AND NO. 2 SHARED SYSTEMS

The applicants have indicated that the following systems will be shared between Unit 1 and Unit 2:

1. fire protection system
2. miscellaneous waste concentrator
3. fuel handling building crane
4. electrical switchyard
5. auxiliary steam generator
6. new fuel storage

A criterion used in the design of TMI #2 requires that no system shall be shared with TMI #1 unless failure of the shared system would not impair the safety of either unit. This independence of Unit 1 from Unit 2 makes it possible to exclude multi-unit type accidents. The applicants have located critical features for both Units 1 and 2, such as the control building ventilation intake and emergency diesel building, to preclude the possibility of either a single aircraft strike affecting both units or impact on shared features that would affect safe shutdown capability.

The applicants have indicated, and we agree, that the sharing of the above systems will not compromise the independence of either Unit 1 or Unit 2 regarding safety of either unit.

5.0 CONTAINMENT AND BUILDINGS STRUCTURE DESIGN

5.1 General Structural Design

Principal Class I and II structures will be founded on bedrock of Gettysburg shale. No liquefaction potential exists. We and our consultants find the foundation provisions acceptable.

Class I structures and components will be designed to withstand the effects of horizontal ground accelerations of 0.06g (Operating Basis Earthquake) and 0.12g (Design Basis Earthquake) with simultaneous vertical accelerations of 2/3 of the horizontal accelerations. These values are the same as are being used for TMI #1. Tornado loading is based on the model as discussed in Section 7.1 which we have accepted on previous plants. Combined loading conditions include the simultaneous occurrence of the maximum earthquake and LOCA.

A major portion of the plant, generally including the reactor, auxiliary, fuel handling and control buildings, is also hardened against direct aircraft crashes and their side effects, such as fire from spilled airplane fuel. Further discussion on aircraft hardening is presented in Section 5.3 and Section 11.0.

We and our dynamic design consultants have reviewed the loading criteria presented for Class I structures and found them to be acceptable.

5.2 Containment Description

The reactor containment building will be a prestressed concrete cylinder with a flat foundation mat and a dome roof. The mat will be 10 feet thick, conventionally reinforced, with a two-foot-thick, reinforced concrete slab above the bottom liner plate. The cylinder will be vertically and horizontally prestressed, with an inside diameter of 130 feet, 4-foot thick walls, and a height of 157 feet from top of mat to spring line. The dome will have a 110-foot radius and be prestressed with a three-way system. The liner plate material will conform to ASTM A-516, Grade 55 and will be 1/2 inch thick in the dome, 3/8 inch thick at the cylindrical walls, and 1/4 inch thick in the base. The containment is sized to provide a free volume of 2 million cubic feet.

The design philosophy under all load combinations, except concurrent maximum temperature and pressure, is that the containment will be in general membrane compression and not crack, except at discontinuities.

The loading combinations have been reviewed by us and found to be acceptable.

The design of the large openings (equipment hatch and personnel lock) will be based on a finite element analysis. The tendons will be draped around the openings and a thickened section will reinforce the wall at the openings. The applicants have adequately described the design parameters by which their analysis and design will be guided. Properly utilized, we believe that these can result in a satisfactory design.

Sufficient information has been presented to indicate that the criteria to be used in designing the tendon anchorage zone concrete can result in a proper design.

The liner design will incorporate sufficient considerations for buckling through the criteria presented, and the liner anchorage system will be so designed that no chain reaction failure of anchors could take place should one anchor fail. A maximum strain limit of 0.5% has been established for the liner. We find these criteria to be satisfactory.

5.3 Aircraft Hardening

The design of TMI #2 against aircraft impingement is based on the same data as were presented in Supplement 5 to the Unit #1 PSAR. Gilbert Associates, Inc., A/E on Unit #1, is acting as consultant for aircraft crash review and design for Unit #2. Burns & Roe is the design A/E for Unit #2. The models used in design are all taken at 200 knots impact velocity and are listed as follows:

Case	Weight	Effective Impact Area
A	6,000#	5' diameter
B	4,000#	3' diameter
C	200,000#	19' diameter

The reactor building will be evaluated for Cases A, B, and C with impingement at the dome apex springline, and on the cylinder walls at or between buttresses. Attachments to the containment structure will also be evaluated for these effects of the above cases.

The control building will be checked for impact on the roof and side walls. Present design indicates that the control room floor will be isolated from the walls, in order to lessen the impact loading on equipment, instrumentation and personnel. The control room will also remain habitable during and after a crash on it.

The fuel handling and auxiliary buildings will be designed to prevent penetration or collapse and for protection against secondary effects due to fire, missiles, etc.

The intermediate building enclosing the main steam lines and feedwater pumps will also be evaluated so as to provide protection below elevation 340'-0".

The loading and impact criteria, as well as potential secondary effects have been reviewed by us, our dynamic design consultants, and by our explosives and shock phenomena consultant. This review is a follow up on a similar one done for Unit #1. The criteria, as presented by the applicants, have been accepted by us and our consultants. The design implementation will be under continuing review during the forthcoming final design stage, at the applicants' option, and in any event evaluation of this implementation of the criteria will be completed during the review for the operating license.

5.4 Containment Prestressing System

A major portion of our structural review has been directed toward the prestressing system. The applicants have not yet selected a system from the four systems they have under consideration.

The four systems proposed by the applicants do not include the BBRV system, which is the only wire tendon system to date which has received final staff approval for use in a containment structure. Upon the receipt of the Russellville and Rancho Seco applications, which first proposed use of a system other than BBRV, a review of the principal prestress tendon systems was initiated. The applicants for those plants subsequently amended their applications to indicate use of BBRV tendon systems and the containment structures were approved. We have continued our evaluation of other types of systems on the basis of information contained in applications and obtained informally from vendors. Our evaluation has been limited, however, since the information in formal applications has always been of a prelimi-nary nature, and the vendors indicated that needed details, such as component stress analyses and results of full-scale proof tests, could only be supplied after a contract award.

In our review of the TMI #2 application, we have had the benefit of our previous evaluation work and we feel that sufficient data have been reviewed to be able to agree in principle to the use of any of the four systems proposed. The applicants have been informed that further in-depth infor-mation will be required for evaluation of the system finally selected and that post-construction permit approval of the design prior to installation will be mandatory.

The following discussion highlights the principal differences in the systems and materials proposed from the usual 90-wire BBRV system which has been the previous "standard system", and the reasons why they are acceptable differences. Detailed tests and data will be supplied by the applicants after final selection of a single vendor, and the review will then be completed for that system. In the event that the applicants can-not establish that the selected system will provide the required safety margins, some other approved or approvable reinforcing will be required.

1. The tendon capacity ranges from 1280 to 2024 kips per tendon. This upper range of large-capacity tendons has been submitted for use on TMI #1, Russellville, Rancho Seco and Ft. St. Vrain.

Large-capacity BBRV tendons have previously been reviewed and approved for use on Russellville, with detailed information promised by the applicants after vendor selection. Rancho Seco Supplement #1 was similarly reviewed contingent upon the same data requests being fulfilled. The use of large size tendons on TMI #2 is acceptable in principle and substantiating information will be included as part of the detailed material to be submitted by the selected vendor.

2. The tendons proposed are wire strands rather than individual wires. Strands have been reviewed on Rancho Seco Supplement #1 and their use in the TMI #2 design is acceptable to us.

3. All four anchorage systems proposed are friction or friction/bearing anchorages, as opposed to the bearing anchorage installed to date (BBRV). Even though no final approval has yet been given for their use in nuclear containments, they are all extensively used throughout the general construction field. The systems proposed are:

Manufacturer	Freyssinet	Stress Steel (SEEE)	VSL Corp.	Western Concrete Structures
Designation	Wire strand	Wire strand	Wire strand	Wire strand
Ultimate Capacity	1944 kips	2024 kips	1280 kips	2000 kips
Design Capacity	1166 kips	1214 kips	768 kips	1200 kips
End Anchorage	Open end, male multiple strand fluted cone gripper bearing into female fluted conical hole	Swagged, threaded collar & rub	Split cone individual strand gripper, bearing into conical holes machined in bearing plate	Split cone individual strand gripper, bearing into conical holes machined into bearing plate

The use of friction anchorages brings up the question of the relative ductility of bearing or wedge-gripped tendons. The significance of wedge-grip systems resulting in a ductility of less than 3%, while straight bearing systems generally have greater than 4% elongation, must be viewed in relation to the actual available ductility in the overall structure. Tests made by Frick on curved 121 wire tendons with bearing anchorages showed an elongation of less tha 3% while for straight tendons results were greater than 4%. Even 1% elongation of the TMI #2 157-foot vertical cylinder walls would result in a 1-1/2 foot increase in height, an order of magnitude larger than anticipated. Based on the applicants' appraisal of anticipated deflections under 115% proof test, as summarized on page S6-C-3 of Supplement No. 6, we believe that the relatively lower ductility available with wedge anchors should not preclude the use of wedge anchor systems. In addition, if one accepts the applicants' proposal for grouting of the tendons, then the comparative importance of the tendon anchorage system is reduced, thereby tending to allay intuitive misgivings about friction anchorage systems. However, the applicants are designing the anchorage systems to withstand at least the tendon's guaranteed ultimate strength--and conducting full scale proof tests--thereby not taking any anchorage credit for the grouted tendon.

4. The applicants have specified that the material for the tendon wires be either stress relieved wire or stress relieved and stabilized wire. Stabilized wire has not yet been officially incorporated into the ASTM standards. In other respects, the wire shall conform to ASTM-A-416,

Specifications for Uncoated Seven Wire Stress Relieved Strand for Prestressed Concrete, with minimum ultimate strength of 250,000 psi and the seven wire strand shall also conform to ASTM-A-416, with minimum ultimate strength of 270,000 psi.

A review of Appendix 5K to the PSAR, a discussion on relaxation losses by Schupack & Associates, and previous contact and discussion with CF&I on their LOK-Stress (stabilized) strand has led us to the conclusion that either stress relieved or stress relieved and stabilized strands can be used for the TMI #2 containment structure prestress system. Predicted losses in the 85°F range are 14% for stress relieved and 4% for stabilized. The degree of predictability and conservatism in the stabilized strand relaxation estimate is at least equivalent to that for stress relieved strand. It is our conclusion, therefore, that if the same design conservatism will be used for stabilized strand as for stress relieved, that the use of either strand is acceptable. Moreover, stabilized strand is being used at Ft. St. Vrain and abroad.

In summa he prestressing systems mentioned above share certain characteristics. Principally, the tendon itself is stranded in each case, and meets the same material and strength specifications. Three of the anchorage assemblies are friction-wedge types, while the fourth is a friction/bearing assembly. We have agreed in principle to their use and will review the finally selected system in depth to verify that it will provide the required safety margins. It is evident that, if the system cannot meet the required standards, and cannot be modified to do so, that an acceptable substitute will have to be provided by the applicants. The applicants have been advised of our position.

5.5 Surveillance and Testing

Grouting of the stranded tendons with a cement grout has been proposed by the applicants. In the United States to date, bonded tendons with cement grout have been used only once in a containment structure. The containment for the H. B. Robinson facility used grouted vertical prestressed steel rods.

The applicants, recognizing a lack of existing data on grouting of large-capacity (approximately 2000 kips) tendons, have conducted a three-phase program of testing. Phase I investigated whether normal grouting procedures could be effective, Phase II tested the effectiveness of special grouting procedures that were developed when found to be necessary, and Phase III checked the reproducibility of the results obt-'ned with the special procedures. Horizontal and draped (around a simulated opening) tendons were test grouted. One of the reasons for conducting these tests was an opinion among many structural engineers that successful grouting of stranded tendons would prove to be much more difficult that for individual parallel wire tendons. The test program showed good grouting results, in spite of some difficulties encountered with the grouting system during part of the Phase III program. We have concluded that a tendon system can be grouted in an acceptable manner if properly qualified procedures are employed.

The use of a grouted tendon system has, in itself, several desirable features. The use of a cement grout automatically provides some bonding of the tendon to the structure with resultant increase in the margin of safety in structural strength, since there is some backup assistance for the anchorage assembly. This also tends to reduce the cyclic loading variation effects on the anchorages and assists in the control of structural concrete crack patterns. Economically, there are apparent advantages for the owner through the use of a less expensive duct filler material and in not being subjected to the expenses of future tendon surveillance. We consider the degree of corrosion protection provided by a well-greased system to be equivalent to that provided by a grouted system although the degree of inspectability to check the tendons is obviously not equivalent.

We are concerned with the fact that hundreds of tendons, amounting to several linear miles, will have to be very effectively grouted in the field under scheduling and environmental and location conditions which may differ significantly from those used in the test program. In actual practice it is virtually impossible to actually verify that a tendon duct is properly grouted. If the grouting quality should be less than that predicted by the tests, it will not be detected and corrosion protection may be lost.

Studies have been made of prestressed tendon failures (both grouted and ungrouted) which show that failures have occurred due to errors in placement, inadvertent inclusion of corrosive elements, and the presence of an unknown corrosion mechanism.

On the basis of our review, including the results of the test program, we still question whether cement grout will provide a corrosion protection environment sufficiently superior to a greased environment to warrant the loss of the inspectability inherent in the greased system. We have concluded that grouting will be acceptable only if a rigorous surveillance system is provided.

The applicants do not consider that periodic inservice surveillance of the containment tendon system is necessary, but has nevertheless presented the following program. A test pressure of 69 psig (115% of design pressure) will be applied at 2, 10 and 20 years after start of commercial operation of the plant. Visual inspection of cracks and displacement measurements with dial guages and theodolites will be used to evaluate the pressure effects. The applicants anticipate that the total deflections will be small, that the significance of differences in these small deflections will be difficult to evaluate and that only a gross difference in the structure, such as a large loss of prestress force, would be apparent from the measurements.

In our opinion the surveillance system proposed by the applicants would show only whether the structure could function as designed, or had failed. There should be a capability to detect structural deterioration, if such should take place, rather than merely being notified that the structure, at some point in time since the previous test, had lost the ability to withstand the test and design loads. We consider that an acceptable surveillance program should utilize permanently installed instrumentation to detect structural deterioration and verify the continued design capability of the tendons during the life of the structure. On this point, we have been unable to come to an agreement with the applicants.

In our opinion instruments, such as acoustical strain gauges or radio frequency cavity gauges, capable of performing satisfactorily over the life span of the structure are available. Periodic monitoring of such instruments embedded well into the concrete structure would indicate deviations in structural strains and changes in the levels of prestress. If corrosion mechanisms were present and active, the instrumentation would show decreases in structural capability before a point would be reached at which the containment could no longer withstand the design and test loads.

At Wylfa, the PCRV was instrumented by means of 426 Davall/MRE acoustical vibrating wire strain gauges placed in the structure with duplicate gauges within 12 inches of each other. At Chinon 3, St. Laurent 1 & 2, and Bugey 1 acoustical strain gauges were similarly employed. The history of these gauges in dam inspections has demonstrated their long life and independence from temperature variations and variations in the insulation of connecting leads. Other methods of monitoring the structure could also be accepted if tendon deterioration could be detected during the life of the structure and before a critical point in the structural strength were reached.

Thus, we conclude that, with proper surveillance to monitor the maintenance of structural integrity, a grouted system would be acceptable and we intend to continue our discussion with the applicants on this unresolved item and will report our progress to the Committee at the July meeting.

Testing

For the initial proof test, we concur in the 115% overpressurization with deformation and crack observations and strain gauge readings. We are

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discussing with the applicants the criteria to which the structural testing will be conducted and evaluated. The applicants will develop specific criteria which will establish the acceptable prestress level and they will be presented for our review and approval.

5.6 Containment Capability

The reactor building will be a Class I, steel lined, prestressed reinforced concrete building. All liner seam welds and penetration welds to the steel liner will be enclosed in welded channels to provide an additional leakage barrier and a means of leak testing. Penetrations of the liner will have means of isolation and pressurizing to about 60 psig which is higher than the peak accident pressure of 52.8 psig. The reactor building will be designed to limit the leakage rate to less than 0.2 percent by weight in 24 hours at 60 psig pressure under accident conditions.

The containment cooling systems for this plant are similar to the systems reviewed and approved for the Rancho Seco plant. There are two emergency heat removal systems: (1) a spray system containing two 1500 gpm pumps and (2) a fan-cooler system containing five fan coolers. Each of the systems is designed for a heat removal rate of 240 x 10^6 Btu/hr which is adequate to limit the containment pressure to less than 60 psig following the loss-of-coolant accident.

During normal operation cooling of the containment will be accomplished using three of the five fan coolers at a reduced cooling capacity. Cooling of the fan-cooler tubes during normal operation is with a closed loop circulating system. On receiving an ECCS signal (either 1800 psig primary system pressure or 4 psig containment pressure), two of four booster pumps start circulating 6000 gpm through the five fan-cooler tubes and the five fan-cooler fans start operating at a reduced speed to prevent overloading the fan motors. The cooling water booster pumps during the emergency fan-cooler operation take suction from the river water via the nuclear services cooling system and exhaust directly back to the river. Instrumentation installed in the fan-cooler cooling water inlet and outlet piping of each fan-cooler unit will detect leakage of unborated river water into the containment and permit isolation if required.

Three fan coolers are required for normal containment cooling and testing of the other two fan coolers can be accomplished during normal operation. The applicants will require the manufacturers of the fan-cooler components to proof test the components in an environment equivalent to the containment atmosphere following an LOCA to assure capacity and long-term operational requirements are adequate.

Testing of the containment spray will be performed periodically during the plant's life. This test will check the spray system up to, but not including, the spray nozzles. The spray nozzles will be tested using compressed air.

A containment pressure of 4 psig will initiate the isolation of all containment fluid penetrations, not serving accident-consequence-limiting systems, to reduce possibility of intolerable leakage from the containment. This system is designed with redundancy such that no single failure or malfunction of an active component can result in loss of the isolation system. The isolation valve seal water system and a containment penetration pressurization system provide a means to pressurize the containment fluid penetrations isolation valves and penetrations to a pressure greater than the containment design pressure of 60 psig. These two systems will improve the effectiveness of the containment isolation and reduce leakage from the containment. Reliable operation of the fluid seal of the isolation valve system penetration pressurization system is assured by duplication of instrument channels and provisions for periodic testing of isolation injection valve operation. Each automatic isolation valve can be tested for operability at times when the valves are not required for normal service.

The adequacy of seal water system and isolation valves will be verified during the initial leakage test of the containment building.

TMI #2 will have hydraulic actuated isolation valves on the secondary steam lines. These steam line isolation valves will be located outside the containment and will be enclosed in a structure designed to meet the aircraft strike criteria.

In evaluating the capability of the reactor building to maintain its integrity following the loss-of-coolant accident, Babcock and Wilcox has developed a model, which has been reviewed in previous plants submitted for construction permits, to describe the time variation of pressure and temperature within the containment. We performed our own calculation using the CONTEMPT Code for the 3-square-foot break which resulted in a peak pressure of 52.3 psig compared with the B&W calculated peak pressure of 51.3 psig. The peak pressure in the containment of 52.8 occurs for a 5 ft^2 break. Since design pressure is 60 psig, there is about 15% margin available to accommodate other energy sources that may result from additional metal-water reaction and secondary steam/condensate. The margin available in TMI #1 containment design was of the order of about 6%.

Table 5.6 summarizes some of the results of the Babcock and Wilcox calculations for the various conditions indicated.

The applicants have analyzed the containment capability assuming a linear metal-water reaction rate energy input to the containment concurrent with an 8.5 ft^2 hot leg rupture. This analysis indicates the containment can withstand a 100% metal-water reaction after 3520 seconds with 5 fan coolers operating or a 100% metal-water reaction after 1170 seconds with 5 fan coolers plus the 2 spray systems operating without exceeding the design pressure of 60 psig.

The above evaluation of the capability of the containment to withstand the energy releases of the complete primary coolant, one steam generator failure with secondary coolant release, and metal-water reation has been based on the assumption that the hydrogen released from the metal-water reaction is recombined as it is evolved.

TABLE 5.6

CONTAINMENT CAPABILITY ANALYSIS

INPUT PARAMETERS

Test Pressure	69 psig
Design Pressure	60 psig
Ultimate Power	2772 Mwt
Reactor Building Free Volume	$2 \times 10^6 ft^3$
Reactor Building Cooling (minimum)	240x10 BTU/hr

CALCULATED PARAMETERS FOR LOCA

	Case 1	Case 2	Case 3	Case 4
Rupture Size, ft^2	14.1	14.1	5.0	5.0
*ECCS Operation	Yes	No	Yes	Yes
Time Blowdown Ends, sec	21	21	29	29
Time Low Pressure Injection Starts, sec	28	No	27.5	37.5
Clad Reacted, %	0.1	5.5	0.1	0.1
Peak Clad Temperature, °F	2060	Melt	1868	1868
Time to Reach Peak Pressure, sec	60	250	60	60
Peak Building Pressure, psig	51.5	56.2	52.8	57.6
One Steam Generator Failure	No	No	No	Yes
Margin to Design Pressure of 60 psig, % $Zr-H_2O$	9.9	4.4	7.2	2.8
Margin to Test Pressure of 69 psig, % $Zr-H_2O$	20	15	19	11

The margins given in the above table, expressed in terms of percent zirconium-water reaction, are based upon a conservative zirconium-water reaction rate of 1 percent per second starting 60 seconds after the rupture occurs.

*ECCS (minimum) includes 500 gpm high pressure injection, 2 core flood tanks, and 3000 gpm low pressure injection.

Our previous evaluations of the dry PWR type containment ($2x10^6 ft^3$) indicated that the hydrogen flammability limit of 4.1 v/O could be reached with approximately a 30% metal-water reaction. In addition, the evaluation of the hydrogen formed by radiolysis of the core coolant during the long-term post-accident time period indicates the hydrogen generation is sufficient to reach the lower flammability limit after the accident.

The applicants were asked to analyze the effects of radiolytic hydrogen on the design and consider possible means to prevent the possibility of the hydrogen concentration within the containment ever reaching the flammability limit. Section 13.4 discusses the results of the applicants' analysis of radiolytic hydrogen effects on the plant design and indicates several approaches that are under consideration to limit the hydrogen concentration in the reactor building below the flammability level.

5.7 Conclusion

On the basis of our review, we conclude that the proposed containment design and cooling systems have the capability to withstand the blowdown energy of the primary system with sufficient margin to accommodate additional energy sources including the consequences of greater than predicted metal-water reactions. The effect of radiolytic hydrogen following a loss-of-coolant accident is being reviewed on a broad basis because of generic concern to all light water reactors.

6.0 REACTOR PRIMARY SYSTEM

6.1 General

The reactor primary coolant system is similar to the Unit 1, Rancho Seco and Russellville plants. However, for TMI #2, the applicants have proposed to design the piping for the primary coolant and emergency core cooling system in accordance with USASI B31.7, Class 1. Our review of the design of the reactor internals and the pressure vessel material surveillance program is discussed in the following sections.

6.2 Reactor Internals

The reactor internals for TMI #2 will be fabricated from SA-240 (Type 304) material. Radiographic inspection standards for welds and welder qualifications will be in accordance with the ASME Boiler and Pressure Vessel Code Sections III and IX.

For loading due to normal operation and design basis plant transients, combined with the operating basis earthquake, the primary and secondary stresses limits for the reactor internals will meet the requirements of the ASME Boiler and Pressure Vessel Code Section III. We find these limits acceptable.

The reactor internals will also be designed to withstand the concurrent blowdown and Design Basis Earthquake (DBE) loads. The method of combining these loads for analytical purposes will be to input both seismic and LOCA excitations and then vary the relative starting times until maximum structural motions, indicating maximum stresses, are obtained. Primary stresses under these combined loads will not exceed 2/3 of the stresses corresponding to the uniform strain value at operating temperature. This criterion results in allowable strains which are less than 20 percent of uniform strain.

All reactor internal components will be removable from the reactor vessel to allow inspection of the internals and the reactor vessel internal surface.

We find the general methods of analysis and the design criteria for the reactor internals to be acceptable.

The nuclear steam system supplier for this plant (B&W) is presently conducting research and development programs to more definitively characterize the blowdown forces on the reactor internals and to demonstrate the adequacy of the internal vent valve design.

In response to question 11.0-A, Supplement 3, the applicants state that information covering the pressure-time history in the reactor and primary system following an LOCA, and the resultant stresses and deflections in the reactor internals, including those resulting from the combined LOCA and DBE, will be provided for our review. This matter is of concern to all the B&W plants. As stated above, we find the limiting design criteria adequate; however, eventual determination of the adequacy of the TMI #2 plant for operation will, of course, be predicated on a realistic appraisal of the actual loadings. We will pursue this matter with the applicants in the post-construction permit period.

We find several inconsistencies in the statements made in the TMI #2 PSAR concerning the test program for the internal vent valves. Although the response to question 11.4-Ae states that the program provides for testing under simulated reactor operating conditions, paragraph 3.3.4 of the PSAR specifically exempts tests at elevated temperatures and tests to show that flow induced vibration, under actual operating conditions, will not allow bypass of significant quantities of coolant nor cause degradation of the valve assemblies or the supporting structure leading to possible systematic failure. These matters are not unique to TMI #2 design, but are of concern to the current generation of the B&W plants. We have indicated to the applicants that we will require experimental data related to the results of testing a prototype valve at fully simulated conditions to determine long-term effects. In-place functional tests of a confirmatory nature will be necessary during the cold and hot preoperation testing.

We also find that vibration studies discussed in the PSAR are limited to planned 1/6 scale model flow tests. While we concur that such tests are a valuable aid in assessing the vibration response characteristics of the internal components, these tests alone are not adequate, and therefore, we will need to review the results of more detailed tests and their applicability to the TMI #2 design.

The applicants should conduct a confirmatory program relating to measurements of the amplitude and frequency of the reactor internal structure vibrations during the cold and hot preoperational testing of this plant. These matters have been discussed with the applicants and they indicated that they will develop a program to cover these matters. We plan to follow the progress of their efforts in these areas during our review of other similar plants as well as during the post-construction review stage.

6.3 Reactor Vessel Material Surveillance Program

The program as outlined contains four surveillance capsules; the tentative DRL position requires a minimum of five. The applicants have orally informed us that they intend to participate in Babcock & Wilcox's Integrated Surveillance Program. B&W made an informal presentation to us on January 16, 1969, and from this we understand that the program will use four capsules per reactor unit. If we allow B&W to place only four capsules in each reactor in the program we conclude that each applicant should retain sufficient archive material to make up two additional capsules and to have space and brackets in the reactor vessel to allow insertion of a fifth capsule. This last provision would protect against an unfavorable conclusion upon evaluation of the Integrated Surveillance Program or if an applicant terminates his intended participation in the program. We will discuss this matter with the applicants prior to the July ACRS meeting.

7.0 OVERALL PLANT PROTECTION

7.1 Tornadic Wind

The applicants have indicated in Amendment No. 5 that the following structures will be designed to withstand the tornadic forces and missiles generated by a 300 mph tangential and 60 mph translational winds including an external pressure drop of 3 psi in 3 seconds. We conclude that the design basis in this regard is adequate.

a. Reactor Building

b. Control Building

c. Auxiliary Building

d. Fuel Handling Building

e. Intake Pump House Structure

7.2 Missile Protection

7.2.1 External Missiles

In Amendment No. 5 Table 1.6-1, the applicants present a spectrum of missiles which were considered as tornado-generated missiles. The applicants were requested to analyze the effect on Unit No. 2 of missiles which might be generated from the adjacent Unit No. 1. Two sources of missiles which are considered to be generated from Unit No. 1 are the turbine-generator and the 400-foot-high cooling towers. Aircraft missile considerations are discussed in Section 11.0.

The analysis by the applicants indicates, and we agree, that failure of the cooling towers will not affect plant shutdown capability or result in an uncontrolled release of radioactivity from Unit No. 2.

In the Unit No. 1 PSAR, missiles generated by failure of the General Electric designed turbines were analyzed. The applicants were requested to analyze the effect of these missiles in the Unit No. 2 structure. The Unit No. 1 turbine axis of rotation is oriented such that the high speed rotational components, which were considered possible missiles, cannot collide with Unit No. 2 structures unless deflected off a Unit No. 1 building column, beam or wall.

Analysis of Unit No. 1 turbine missiles presented in Supplement No. 3 of TMI #1 PSAR indicated a minimum of 13 inches of concrete was needed to absorb the energy of the most critical missile. All Unit No. 2 structures which are designed to withstand the tornadic winds and missiles will

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be completely enclosed within cast-in-place reinforced concrete structures with a minimum section thickness of about 3 feet. These wall thickness requirements are established by the aircraft protection requirement.

7 2 2 Internal Missiles

The TMI #2 design will include protection of critical components required for safe shutdown, against internal missiles generated by component failure such as valve bonnets, pump flywheels, or high pressure components.

In Amendment 10 the applicants presented a description, design evaluation, inspection and testing, and surveillance proposed for the TMI #2 primary coolant pump flywheel. The control rod drives reactor vessel, core flooding tanks and the reactor building emergency coolers are protected from potential flywheel missiles by shield walls. Thus, failure of the flywheels is not expected to damage these components; however, additional analyses of the possible failure modes including the effects of an earthquake will be necessary. We intend to discuss this matter with the applicants prior to the July ACRS meeting.

We have reviewed a preliminary design of the flywheel and concluded that the proposed inspection during fabrication is not adequate because it only considers surface conditions. We will require that the inspection efforts be increased to include radiography and ultrasonic inspection of the flywheel subassemblies The basis for requiring increased fabrication inspection is for assurance that the units are free of gross defects. Because of limited accessibility of the flywheel for inservice surveillance and inspection, assurance that preservice gross defects are not present reduces the likelihood of failure during plant operation. We will review the applicants' program for inservice inspection during our post-construction permit review as part of the overall plant inservice inspection program described in Section 18.0.

7.2.3 Conclusion on Missiles

We conclude from our review that the external and internal missile protection provided in the TMI #2 design is adequate. However, we will require that the inspection proposed for the primary pump flywheel during fabrication be increased to include sufficient volumetric inspection to assure the flywheels are free of gross defects which could result in failure during operation. We also intend to review the results of the applicants' seismic analysis and their plans for inservice inspection of the flywheels.

8.0 INSTRUMENTATION AND CONTROL

The Commission's General Design Criteria and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE No. 279), dated August 28, 1968, served, where applicable, as the bases for judging the adequacy of the Protection Instrument System and the Control System.

A comparison has been made with the previously approved Russellville plant for the purpose of avoiding repetitious analyses. Accordingly, only those design features which are new, or for which additional information has been received, are addressed herein. We have verified that only the redesigned rod control system, including the scram bus, and a slightly modified pump monitor logic circuit constitute novel features. In addition, the applicants have submitted environmental criteria (post accident, seismic, etc.) and testing information unique to this station.

8.1 Protection Instrument System

8.1.1 Scram Bus

The new design for the scram bus is shown in Figures 7-2B and 3-7? of the PSAR. As in all previously approved designs, the initiating circuitry for scram is comprised of four "two out-of-four" (2/4) relay matrices which are, in turn, actuated by the protection instrument system channels connected in a 2/4 general coincidence a angement.

Downstream of the relay matrices in the new design is a set of breakers which, when tripped, interrup. (a) the voltage source to all power supplies, (b) the control voltage to the control rod drive SCRs, and (c) the output voltage from the "hold" power supplies which are used to hold the safety groups subsequent to withdrawal.

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There are four groups of safety rods (Gr. 1-4) and four groups of control rods (Gr. 5-8). Each group is energized (moved or held) by two power supplies connected in parallel such that the loss of a power supply does not initiate a spurious trip of the associated group. In series with each control rod power supply is a gate circuit. In series with each hold supply is an independent trip breaker. Trip logic is 1/2 x 2 since the tripping of a power supply or its gate (breaker in the case of the safety rods) and the other power supply or its gate (breaker) trips the respective rod group.

With respect to single failures, our analysis indicates that, because of redundant breaker logic (1/2 x 2), a failed breaker cannot prevent a scram. Consequently, it appears that the only areas of vulnerability remaining to be considered are those which arise from possible short-to-line faults downstream of the breakers which, by their nature, could not be terminated by breaker action.

Any such fault which occurs immediately downstream of a breaker is equivalent to breaker failure, which has been considered above. Potential failures downstream of the power supplies and gate circuits should next be considered. An unsafe failure of any one (or combination) of the gate circuits to a power supply will be overriden by the breaker which interrupts the voltage to the supply. An unsafe failure, such as a short-to-line, downstream of the power supplies would affect only one group; the remaining seven groups would not be prevented from scramming.

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In terms of testability, it appears that all of the faults postulated above are inherently detectable. For example, short circuits between the various portions of trip circuits would be detectable inasmuch as each path of the collective circuitry can be independently tripped. Thus, the presence of a voltage downstream of a tripped component would indicate the presence of a short circuit.

Based on the foregoing analysis, we conclude that the design of the scram bus satisfied all present criteria with respect to redundancy, independence, and testability and is acceptable.

8.1.2 Pump Monitor Logic

The pump monitor logic has been changed to permit the loss of two pumps without scramming provided (a) the two pumps are in different loops and (b) reactor power is less than 50% FP.

We have concluded that this modification does not compromise plant safety. Our conclusion is based upon the applicants' response to Question 9.3 in Volume 4 which states that the pump monitors are essential for safety only if all four pumps are simultaneously lost when the reactor is operating above 109% FP. Since (even with the new logic) any two monitors would trip the reactor under these conditions, the modification has not compromised the single failure criterion. Accordingly, we conclude that the modification is acceptable.

8.1.3 Environmental Testing

The only instruments located within containment which initiate safety feature action are the reactor coolant pressure detectors. The applicants have stated that these instruments and associated wiring will be qualification tested for the conditions which would exist after an accident. These tests will simulate the post accident environment as closely as possible.

While we agree with the applicants' criteria with respect to the testing of instruments and wiring within containment, we have observed that vital valves have not been included in the test program. We will pursue this matter with the applicants and will be prepared to report orally to the to the Committee.

8.1.4 Seismic Design

With respect to seismic design, the applicants have stated that the components of the protection system, including the station batteries and racks, will be designed to withstand simultaneous accelerations of 0.08g vertical and 0.12g horizontal. We understand that the acceleration values represent ground motion and do not necessarily apply at the equipment location. We will discuss the matter with the applicants and will be prepared to report orally to the Committee.

Assuming satisfactory resolution of these items, we conclude that the applicants' criteria with respect to the functional capability of vital components under seismic conditions are acceptable.

8.2 Control System

The rod control system has been redesigned to accommodate the new drives which have replaced the previously approved rack and pinion concept.

The system is described and evaluated in Section 3.2.4.3 of the PSAR. A block diagram is shown in Figure 3-70.

In addition, the applicants and B&W presented supplementary information to the staff during a meeting at Bethesda arranged for this purpose.

Each of the 69 rods is powered by a stepping motor whose stator and rotor are physically separated by the pressure boundary. Stator windings are star-connected and are sequentially energized to produce rotor motion. For example, when two adjacent stator windings are energized, the rotor will align with the resultant magnetic vector; i.e., it will align midway between the windings. Energizing the adjacent winding while retaining power at the first two will yield a resultant along the central (of the three) stator windings. Thus, by sequentially energi[illegible] two and three windings, a rotor will be displaced 30 degrees per step. Further, in addition to displacing a rotor, the magnetic field serves to clamp the roller nut to the lead screw. Scram is accomplished by de-energizing the the stators.

The rods are divided into eight groups (four control and four safety). Each control group has its own pair of power supplies. A fifth pair, the Auxiliary Power Supplies, are used to withdraw the safety groups one at a time. A sixth pair, the Hold Supplies, hold the safety rods in place once they have been withdrawn. The elements of a pair operate in parallel in order to prevent loss of control, or a trip, in the event of one power supply failure

As shown in Figure 3-70, each power supply (e.g., Group 5 Supply A) is energized from a three-phase source. Within a supply, the three-phase is transduced to six-phase, star connected. Within each supply there are six sets of six SCR's. The cathodes of each set are connected in common and to one of six output points. The anodes of each set are respectively connected to one of the six a.c. phases described above. Thus, each set of

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SCR's becomes a six-phase half-wave rectifier. The six output points (plus a common return) are respectively connected to the stator windings of a rod motor, or motors, when the stators are connected in parallel. The energizing of two and three windings is then accomplished by the gate circuits shown in Figure 3-70 as a single SCR symbol. Each symbol represents six gating circuits (one for each of the six rectifiers). The gating circuits are themselves sequenced from photo-cell circuits which receive light impulses from a coded disc rotated by the programmer motors. When in motion the programmer motors are synchronized to the vital bus which is supplied from an inverter. Thus, each rod is synchronized to its programmer motor.

Rod position indication is by means of 69 individual indicators (scale and pointer). Rod position is sensed by reed switches whose outputs are analog signals proportional to rod motion. A deviation alarm compares the analog position for each rod to its group demand position and annunciates any deviation. The group demand is derived from the programmer output.

Our analysis of the control system considered potential malfunctions which could permit, or cause, spurious reactivity insertions. With respect to malfunctions which could cause more than the intended group to be withdrawn (at design speed) we agree with the applicants' analysis that, should a multiple withdrawal of any number of rods be initiated at any power level, it would be successfully terminated by the protection system.

With respect to withdrawals at greater than design speed, we believe that the applicants have not considered the possibility of excessive withdrawal

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speeds resulting from a runaway programmer. Since each programmer is a synchronous motor supplied from an inverter, it follows that a spurious frequency change at the inverter would result in a corresponding change in rod speed. At this stage of the review, however, we will accept the applicants' criterion that the drive controls, or mechanism and motor combination, shall have an inherent speed limiting feature (Ref. PSAR, Page 3 93). We will study the final design during the POL review in order to confirm that it satisfies the criterion.

We conclude that the control rod system, including the rod position indication, is adequate and is consistent with recently approved systems of similar design.

8.3 Emergency Power

8.3.1 Offsite Power

Offsite electrical power for the two engineered safety feature (essential) buses is furnished by two auxiliary transformers, each connected to a separate bus at the 230 kV substation. Each essential bus is normally supplied from a separate transformer, with provisions for automatic transfer to the other transformer. In addition, a tap is made from each Unit No. 1 auxiliary transformer so that either may be used to provide another source of power if required.

The breaker and a-half switching arrangement in the 230 kV substation allows any of six lines or either of the unit generators to be connected to either of two full-capacity main buses that supply the two auxiliary transformers of each unit. If the substation is separated from its ties to the 230 kV system, neither the reactor nor the turbine should be tripped, and either Unit No. 1 or Unit No. 2 itself can supply the auxiliary demand through the substation and the auxiliary transformers.

Engineered safety feature loads will be duplicated in two independent systems with each system being fed from one of the 4160 volt essential buses. A manually operated bus tie is provided for emergency use.

The applicants have performed analyses to show that the sudden loss of Unit No. 2, or the simultaneous sudden loss of both units, will not cause instabilities within the external grid. We agree with the results.

The offsite system is arranged with sufficient redundancy and automatic switching so that any single failure should leave one of the essential buses in service to meet minimum requirements. We, therefore, consider it adequate to meet the requirements of the criterion.

8.3.2 Onsite Power

Onsite electrical power for the two engineered safety feature buses is furnished by two diesel generators. Each emergency generator will feed one of the two engineered safety feature 4160 volt buses (split bus arrangement), and only one is needed to supply the necessary load. The 4160 volt buses and their associated 480 volt buses will be physically separated from each other. The essential buses, the diesel generators, and the fuel supplies will be of Class I seismic design. Enclosures housing the diesel generators are of all-concrete construction with eighteen-inch interior walls and three-foot exterior walls for earthquake, fire, and tornado protection. The underground fuel oil storage tanks will supply one generator at full power for seven days.

Each diesel generator will start upon loss of its respective bus and will energize that bus when it has achieved rated speed and voltage. In addition, both diesel generators start upon occurrence of (1) Initiation

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of Safety Injection, (2) Over-pressure in the Reactor Building, or (3) loss of either transformer source. Equipment for shutdown is then manually connected to the bus. If there is a requirement for engineered safety feature system operation, the respective equipment is automatically started sequentially as soon as each generator connects to its bus.

Each diesel generator has a continuous rating of 3000 kw. Maximum emergency load is 2749 kw. Thus, assuming failure of one diesel, there is a 9% margin with respect to the continuous rating.

The diesel generators can be separately tested (started and loaded) during reactor operation. Further, the split bus design allows the independent testing of each safety feature sequencing system.

The 250/125 volt, 3-wire d-c system will consist of two buses, each supplied by battery and static rectifier charger. The two buses and loads will be arranged as a split bus and will be designed to satisfy the single failure criterion in terms of connecting ESF loads and shedding unnecessary loads in the event of an accident. Each battery is sized to carry the necessary load for a period of two hours. The two station batteries will be located in separate rooms of a Class I structure.

We conclude that the design of the onsite emergency power system fulfills the requirements of the General Design Criterion #39 relating to independence, redundancy, testability and functional capability and is acceptable.

8.4 Cable Separation

The applicants' criteria relating to cable tray loading and separation have been submitted in Section 8 of the PSAR and in Supplement #2. These criteria apply to the protection system and the onsite power system and are summarized below.

A. Metal enclosed bus ducts will be used for all major bus runs where large blocks of power are to be carried. The routing of the ducts will be such as to minimize their exposure to mechanical, fire or water damage.

B. Wire and cable related to engineered safety feature and reactor protection systems will be routed and installed in such a way as to maintain the integrity of their respective redundant channels and protect them from physical damage. Power and control cables for duplicate auxiliaries or services will be run by different routes to lessen the probability of an accident's disabling both pieces of identical equipment.

C. Cables will be selected with respect to their current carrying capacities, insulation properties and mechanical construction.

D. Power circuits of 480 volts and less shall be installed with only one layer in a tray.

E. Insulation used in power cables shall be rated for 90°C conductor temperature, but the sizing shall be based upon a conductor temperature of 75°C.

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We have reviewed the criteria as presented in the PSAR and concluded that, if properly implemented, they reduce the probability of cable fires, and provide protection against random and systematic failures. Our conclusion with respect to the low probability of cable fires is based on (a) the limited loading of the trays carrying the 480 volt power circuits, (b) the derating to 75°C, and (c) the proper selection of all cables in terms of current carrying capacities.

With respect to systematic and random failures, we conclude that the diverse routing of redundant cables can, if properly installed, provide adequate protection against the propagation of a fire, and against any lesser single event occurring within a tray.

We have observed that there are no criteria with respect to fusing. We will resolve this prior to the meeting with the Committee.

Apart from this one reservation, we are in agreement with the applicants' cable tray loading and separation criteria.

9.0 ENGINEERED SAFETY FEATURES

9.1 Emergency Core Cooling System (ECCS)

The TMI#2 ECCS design is essentially identical to that proposed for the TMI#1, Crystal River, Rancho Seco, and Russellville nuclear plants. Single failures of active ECCS components and single passive failures during long-term cooling can be tolerated without jeopardizing the plant safety. All piping for the primary coolant system and emergency core cooling system will be designed in accordance with the USASI Code B31.7, Class I.

The ECCS for the TMI#2 will consist of the following systems:

1. High pressure injection system - This system normally operates as part of primary system makeup and purification. Two independent and redundant systems with three high pressure pumps inject a minimum of 500 gpm into primary inlet piping. The system is initiated by a signal derived from primary system pressure (1800 psig) or from the containment pressure (4 psig).
2. Core flooding system - There are two independent and redundant storage tanks containing 1880 ft^3 of borated coolant. The coolant is discharged automatically via two separate 14-inch diameter nozzles into the primary pressure vessel when primary system pressure drops below 600 psig.
3. Low pressure injection system - This system normally operates for shutdown cooling as part of th decay heat removal system. It consists of two independent and redundant systems. Each system provides 3000 gpm coolant injection via the same 14-inch diameter nozzles used for injection of core flooding tank coolant into the pressure vessel when primary system pressure drops to 200 psig. The initiation signals for the system are the same as described in 1 above.

The source of coolant for the ECCS high pressure injection and low pressure injection is a 450,000 gallon borated storage tank. All emergency [illegible] coolant will be maintained at a minimum concentration of 2270 ppm of boron. Sizing of the ECCS design was based on limiting the maximum fuel clad temperature to less than 2300°F for any size primary pipe rupture up to the double-ended rupture of the 36-inch diameter outlet piping and operation at the ultimate power level of 2772 Mwt.

The applicants have performed analyses for a primary pipe rupture spectrum from 0.4 ft^2 to 14.1 ft^2 to determine the time delay to initiate the ECCS signal based upon an 1800 psig primary system pressure and the 4 psig containment pressure ECCS signal backup.

The results of the analyses indicate the 1800 psig ECCS signal has a time delay of about 1 second before ECCS operation for primary pipe ruptures in the range from 0.4 ft^2 to 14.1 ft^2. Delay time in activating the ECCS by the 4 psig containment pressure is a maximum of 20 seconds for the 0.4ft^2 primary piping rupture.

Based on 1800 psig primary ECCS signal, high pressure injection valve operation starts at about 15 seconds and the valve is fully open at 25 seconds. Based on the 4 psig containment pressure ECCS signal, the high pressure injection valve would start opening at 20 seconds and be fully open at 30 seconds. The effect of differences in ECCS actuation time is about a 50-gallon loss of high pressure injected coolant. This water loss would have no effect on the consequences of LOCA.

For primary pipe ruptures less than 0.4 ft^2 the applicants indicate the backup ECCS signal (4 psig) may not be an adequate backup ECCS signal; however, the applicants conclude that the 1800 psig primary system pressure ECCS signal based on the redundant two-out-of-three actuation signal provides sufficient reliability for ECCS actuation. Although we agree on the

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reliability of the 1600 psig primary system pressure ECCS signal, we conclude that further evaluation is necessary with regard to ECCS signal diversity. We have informed the applicants of our requirements for diversity and indicated we will follow up during the post-construction permit review stage.

In the event of a loss-of-coolant accident due to a rupture of the hot leg primary piping, the nitrogen used to pressure the core flooding tanks could expand and pass through the core. We requested the applicants to determine the effect on core cooling and resultant fuel clad temperatures of the nitrogen blowdown. In Supplement No. 5 of the PSAR (subject #7) the results of the analysis indicates, and we agree, that sufficient vent paths exist between the inlet downcomer and the plenum above the core th[illegible] the expansion of the nitrogen would result in nitrogen passing through the core for only several seconds. The analysis indicates maximum hot spot clad temperature as a function of time would not significantly be changed from those presented in Section 14 of the PSAR.

The applicants have performed the TMI#2 accident analysis based upon the ultimate power level of 2772 Mwt. In analyzing the loss-of-coolant accident, the assumption is made that the minimum emergency coolant injection of two core flooding tanks plus 3500 gpm (one high pressure injection pump and one low pressure injection pump) provide coolant to the core. Delivery of the 3500 gpm coolant injection is assumed not to start until the primary system has depressurized to 100 psi or after 25 seconds, whichever occurs later. The maximum fuel clad temperatures for a hot leg and cold leg break spectrum are shown in Table 9.1. The maximum clad temperatures shown are for the power levels of 2568 Mwt and 2772 Mwt, respectively. The method of analysis is the same as has been used on previously reviewed B&W reactors.

We have concluded that the design of the proposed ECCS (1) limits the peak clad temperature to well below the clad melting temperature, (2) limits the fuel clad-water reaction to less than one percent of the total clad mass, (3) terminates the temperature transient before the core geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, and (4) reduces the core temperature and removes core heat until the core will remain covered without recirculation and replenishment of coolant. The ECCS is designed to provide this protection for all sizes and locations of pipe breaks up to and including the instantaneous double-ended rupture of the largest reactor coolant pipe. We plan to continue our review of ECCS initiation signal diversity with the applicants during the post-construction permit review stage.

TABLE 9.1

MAXIMUM CLAD TEMPERATURES RESULTING

FROM A LOSS-OF-COOLANT ACCIDENT

Hot Leg Ruptures

Rupture Size (ft^2)	Full Power Seconds	Minimum Water Level Below Core (ft)	Hot Spot Max. Temperature (°F) 2568 Mwt	2772 Mwt
14.1	1.55	-6.8	1,950	2,060
8.5	2.65	-5.1	1,916	2,040
5.0	4.12	-4.0	-----	1,655
3.0	3.35	-2.2	1,235	1,477
1.0	6.25	+4.7	1,075	1,151
0.4	6.50	+12.0	1,015	1,181

Cold Leg Ruptures

Rupture Size (ft^2)	Full Power Seconds	Minimum Water Level Below Core (ft)	Hot Spot Max. Temperature (°F) 2568 Mwt	2772 Mwt
8.5	0.60	-6.7	1,785	1,854
5.0	0.40	-6.0	----	1,868
3.0	1.00	-4.8	1,575	1,599
1.0	5.38	+3.6	1,250	1,354
0.4	5.50	+7.0	1,090	1,176

Additional calculations were performed to determine the effects on the maximum fuel pin clad temperature by a parametric study of the 14.1 ft^2 break. The following is a summary of these calculations.

Parameter		Change in Maximum Clad Temperature (°F)
DNB	(a) reached at 0 time	-15
	(b) reached at 4 seconds	+260
Heat Transfer Coefficient	decrease of 20%	+120
Moderator Coefficient	increase from -0.5×10^{-4} to -1.2×10^{-4} $\Delta k/k/°F$	+240

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The B&W reactor, like other PWR's, is subject to potential steam binding of core coolant flow in the post-LOCA period. In the B&W single pass heat exchanger and primary loop design, a loop seal which remains filled with water following blowdown from a cold leg break would hinder steam venting from the primary system. The B&W reactor design therefore includes core barrel vent valves which act to equalize steam pressure across the core support shield and hence to preclude depression of the core water level for the steam production (flow) rates caused by boiloff from the shutdown core and from the hot vessel and internals. However, if steam generator tubes were to break during or soon after a loss of primary coolant, steam from the secondary system could flow rapidly into the reactor vessel upper plenum and severely increase the steam binding. Like the Westinghouse and Combustion plants, there is some relatively small number of tube failures in the B&W secondary system which would eventually result in core coolant level depression. We have not specifically addressed this problem in the review of TMI#2, and therefore the analyses to determine the effects of concurrent LOCA and steam generator tube failures have not been performed for the B&W design. We intend to study this PWR problem further on a generic basis.

9.2 Iodine Removal System

The TMI#2 spray system is similar to the TMI#1 and Russellville Nuclear Unit No. 1. The TMI#2 design will employ an alkaline sodium thiosulfate solution mixed into the two independent 1500 gpm containment spray systems to increase the iodine removal capability following the loss-of-coolant (LOCA) accident.

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In assessing the effectiveness of the spray system, the applicants use a spray removal fa. r of 12 hr^{-1}. The determination of this removal factor is based upon; (1) a uniform spray drop size of 1000 microns, (2) a containment temperature of 286°F and a pressure of 60 psig, (3) a flow rate of 1500 gpm (one of two sub-systems operational), (4) the use of a calculational method for the gas-film resistance model developed by Griffith (ORNL), (5) assumption that 5% of the iodine released to the containment is non-removable, (6) a containment leak rate of 0.2%/day, and (7) alkaline sodium thiosulfate additive to spray solution. Using the above seven conditions the applicants calculated the exclusion radius (2-hour) and low population zone (30-day) thyroid doses to be 125 rem and 90 rem, respectively.

Our evaluation of the proposed spray system effectiveness resulted in an iodine removal factor of 3.1 hr^{-1}. We have used a calculational method similar to that used by the applicants described above with the following modifications: (1) inclusion of a liquid film mass transfer resistance in addition to the gas-film resistance (2) use of an experimentally derived correction factor for a standard deviation of 1.5 from the mean geometric drop diameter (3) assumption that 15% of the iodine released to the containment is non removable (10% organic + 5% particulate), and (4) reactor building leak rate of 0.2%/day. Using these modifications which give a total reduction factor of about 3.6 (2-hour) and 6.1 (30-day), we calculated the exclusion radius (2-hour) and low population zone (30-day) thyroid doses to be 450 rem and 290 rem, respectively.

Our calculations of the potential doses, without any iodine removal system, indicate that iodine reduction factors of approximately 5.3 for the exclusion radius (2-hour) dose and of 5.9 for the low population zone (30-day) dose are required in order to meet the guideline values specified in 10 CFR Part 100.

Since the research and development for the alkaline sodium thiosulfate spray additive has not been completed, the applicants provided alternatives which would be used to reduce the iodine thyroid doses below the 10 CFR 100 guideline values using our method of evaluation. The applicants have indicated that alternate action could be used in the event the research and development for the alkaline $Na_2S_2O_3$ additive proves to be inadequate or unacceptable. These include a reduction of the allowable containment leak rate, the addition of particulate filters (HEPA) and containment recirculation filter systems with charcoal filters, or a combination of these actions. Although not specified by the applicants, an increase in the spray flow rate would also effect a reduction in the doses.

The following table is presented to illustrate the degree to which the doses may be reduced. We have used our standard calculational methods and assumptions with the exception that we allowed credit for spray effectiveness in removing organic iodine in cases (2) through (5).

TABLE 9.2

DOSE REDUCTION EFFECTIVENESS

Case No.		Iodine Removal	2-Hour Thyroid Dose (REM) With HEPA	Without HEPA
1.	Spray only (1500 gpm)	a. nonorganic $\lambda = 3.1hr^{-1}$ b. organic $= 0$	370	450
2.	Spray only (1500 gpm)	a. nonorganic $\lambda = 3.1hr^{-1}$ b. organic $\lambda = 0.35hr^{-1}$	330	400
3.	Spray only (1800 gpm)	a. nonorganic $\lambda = 3.6hr^{-1}$ b. organic $\lambda = 0.35hr^{-1}$	290	370
4.	Case (1) plus Rancho Seco filter system	a. nonorganic $\lambda = 3.6hr^{-1}$ b. organic $\lambda = 0.35hr^{-1}$	290	-
5.	Case (2) and reduce leak rate ~~from~~ 0.2 to 0.1%/day	a. nonorganic $\lambda = 3.1hr^{-1}$ b. organic $\lambda = 0.35hr^{-1}$	160	200

If the B&W research and development program results do not confirm the applicants' assumptions on spray effectiveness, then we can require the installation of HEPA filters and a combination of increased spray flow and decreased allowable containment leak rate. On this basis we can conclude that, based on the results of an R&D program, an iodine removal system for TMI#2 can be developed that would limit the radiological consequences of a loss-of-coolant accident to doses significantly below the 10 CFR 100 guideline values.

10.0 CONTROL ROD DRIVE MECHANISM

The control rod drive mechanism to be used in the TMI#2 reactor was changed in Amendment No. 4 from the original rack-and-pinion type drive mechanism to a roller-nut type drive mechanism. A description of the proposed roller nut drive and its components is given in Section 3.0 of the PSAR.

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The roller-nut drive mechanism is based upon extensive analytical, developmental, design, test, and manufacturing experience obtained over the years from the Shippingport and Naval Nuclear Programs.

The control rod drive system includes drive mechanisms which actuate control rod assemblies (CRA), drive controls, power supplies, position indicators, operating panels and indicators, safety devices, enclosures, housings, and mountings. Criteria applicable to the drive mechanism design cover (a) single failure (b) uncontrolled withdrawal, (c) equipment removal (d) position indication, (e) system monitoring, (f) drive speed, (g) mechanical stops, (h) CRA positioning, (i) CRA trip and (j) CRA withdrawal. We have reviewed the design criteria and conclude they are essentially the same criteria used and previously reviewed for the rack-and-pinion drive mechanism.

In Supplement No. 5 the applicants have addressed areas in which we have expressed concern regarding the change to the roller-nut control rod drive mechanism.

The roller-nut drive mechanism will have the capability of applying an insertion force of approximately 400 pounds in the event a control rod should fail to drop to a fully inserted position following a scram. This rod drive-down option is not automatic but requires operator action to initiate.

The following table contains a comparison of control rod drive mechanisms for various PWR plants including the proposed roller-nut drive mechanism.

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TABLE 10.0

COMPARISON OF CHARACTERISTICS OF PWR CONTROL ROD DRIVE DESIGNS

	Babcock & Wilcox		Westinghouse		Combustion Engineering	
Type of Drive	Rack-&-Pinion	Roller-Nut	Magnetic Jack	Roller-Nut	Rack-&-Pinion	Magnetic Jack
No. of Drives/reactor	69	69	52	8	45	65
No. of Rods/drive	16	16	20	20	1 (Cruciform)	5 or 10
Control Rod Poison	Ag In Cd	Ag In Cd	Ag In Cd	Ag In Cd	B_4C	B_4C
Total Rod Worth (Δk)	10%	10%	7%	--	9%	9%
Stroke Length; in.	139	139	144	144	132	132
Drive Speed; in/min	25	30	45	15	46	23
Type Scram	Gravity & Auto Drive-Down	Gravity	Gravity	No Scram (Partial length rods)	Gravity & Auto Drive-Down	Gravity
Position Indication	Gear-pot & Inductance Coil	Magnetic Reed Switches Motor-pot	Differential Transformer	Differential Transformer	Gear-pot & Magnetic Reed Switches	Differential Transformer
Trip Time	1.4 sec. for 2/3 insertion	1.4 sec. for 2/3 insertion	2.0 sec. full insertion	---	2.5 sec. 90% insertion	< 2 sec.
Design pressure; psig	2500	2500	2485	2485	2500	2500
Design pressure; °F	650	650	650	650	650	650
R&D Program	Yes	Yes	No	Yes	Yes	Yes
Approved for CP	Yes (all B&W)	No	Yes	Yes (all Westinghouse Reactors)	Yes (Palisades)	Yes (Maine Yankee)
No. of partial length rod drives	8	8	None	8	4	---
Designer	B&W	B&W	West.	West.	Comb. Engr.	Comb. Engr.
Manufacturer	B&W	B&W and Royal Ind.	West.	West.	Comb. Engr.	Comb. Engr.

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A roller-nut drive mechanism research and development program, essentially the same as the R&D program for the rack-and-pinion drive mechanism, has been completed by Babcock & Wilcox. The results of this R&D program will be submitted as a topical report in 1969.

We have reviewed the information regarding the drive mechanism presented in Supplement No. 5, Section 3, and questions 11.0 of Supplement No. 3. Based upon our review of the applicants' R&D program, we have reasonable assurance that the roller-nut control rod drive mechanism will provide an acceptable design for use in the TMI#2. We will review the results of the program prior to plant operation.

We conclude that the proposed control rod drive mechanism is acceptable for TMI#2.

11.0 AIRCRAFT PROTECTION

Because of the proximity of the Olmstead State Airport to the Three Mile Island site, we required that the TMI#1 plant design include features to protect against an aircraft strike and accompanying fire hazard. A statistical analysis of the air traffic, including frequency and type of aircraft landing and taking off from the Olmstead State Airport integrated with the fatal accidents for landings and takeoffs of U.S. air carriers over the past 10 years is presented in Supplement No. 4. The applicants' criteria for protection against the effects of aircraft crash for the Three Mile Island site require that all components and buildings, which contain accident-mitigating components, equipment required for safe shutdown of the plant, or components whose failure would result

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in an uncontrolled release of radioactivity will be protected to withstand without failure the impact of a 200,000-lb aircraft (B-720 Class traveling at a velocity of 200 knots and any accompanying fire which might result. These criteria are consistent with those given in the Proposed Criteria for Siting Power Reactor Facilities Near Airports.

Our consultants have reviewed the TMI#2 design criteria with regard to fire protection, missile characteristics and structural design requirements. Detailed design for aircraft protection is currently being established for the TMI#1 which is under construction. The applicants have indicated the TMI#1 design results will be applied to TMI#2 design.

The following list shows the buildings and components of TMI#2 which will be designed to withstand the above aircraft strike.

Aircraft Protected Buildings

a. Reactor building
b. Control building
c. Auxiliary building
d. Fuel handling building
e. Intake structure-service water pump bay

Aircraft Protected Equipment

a. Main steam lines between steam generators and the main steam isolation and second [illegible] heat steam isolation valves.
b. All feedwater lines between steam generators and the feedwater control valves.
c. Main steam pressure relief valves.
d. Main and startup feedwater control valves.

The emergency diesels will be located in a separate building adjacent to the fuel handling building, on the opposite side from the switch yard. Temporary loss of the emergency diesels due to an aircraft strike fire, which might result from a deficiency of oxygen, would not impair the safety of the plant.

Protection of the plant and components from the effects of the fire accompanying the aircraft strike will be provided by including the following criteria.

Fire Protection Criteria

a. Ventilation openings will be located to take advantage of natural shielding provided by other plant structures.

b. Ventilation inlet openings to critical areas will be provided with passages, curbs, fire dampers and sensing devices for automatically closing dampers to isolate these areas.

c. Self-contained breathing apparatus will be located in critical areas.

d. Design will provide drainage of liquids away from the protected areas.

e. Piping and conduits passing through protected areas will be designed with seals and curbs to prevent liquid leakage into protected areas.

The TMI-2 will supply air to the control building and auxiliary building through a tunnel with a remote intake structure located 125 feet from the plant. The intake structure will contain vapor detectors and fire

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detectors to initiate isolation of the control building air intake dampers and start operation of a deluge water spray system in the air intake tunnel to control any fire hazard.

Gilbert Associates, Incorporated, the architect-engineer on TMI#2 will provide consultant services for aircraft protection requirements for the TMI#2 plant design.

We and our consultants, Mr. T[illegible] Pinkel (fire protection), Mr. James Proctor (missile characteristics), and Dr. Nathan M. Newmark (structural design) have reviewed the TMI#2 design and conclude the aircraft protection provided and criteria established are adequate to meet the requirements for this site. We shall review the final design for aircraft protection during post-construction meetings and during the operating license review to assure all criteria have been incorporated into the final design of the TMI#2 facility.

12.0 FUEL STORAGE POOL

The fuel storage pools will be located within a Class I reinforced concrete building. Blowout panels will provide pressure relief consistent with the external pressure drop of 3 psi in 3 seconds. A minimum wall thickness of about three feet will provide protection against missiles generated external to the fuel handling building including an aircraft strike. The fuel storage pools walls will be reinforced concrete lined with stainless steel.

A separate steel lined pool will be provided to accommodate the spent fuel shipping cask during fuel transfer operation. This pool will be separated from the fuel storage pools by a wall with a removable gate to permit underwater transfer of spent fuel to the shipping cask. The applicants were requested to analyze the possibility of the shipping cask being dropped into the fuel storage pool. In Amendment No. 5 an analysis is presented which indicates that the shipping cask, if dropped, could result in shear failure of the bottom slab of the shipping cask pool. The applicants have incorporated an administrative key-control interlock to limit the travel of the fuel handling crane during transfer of the shipping cask. This crane travel limit is required to prevent the shipping cask from falling into the fuel storage pool if the crane or lifting cables were to fail during shipping cask transfer operations. In addition, the area below the concrete floor of the cask pool will be backfilled to prevent failure of the floor if the cask is dropped into this separate pool.

The applicants state, and we agree, that inclusion of the above crane travel interlock and proper administrative control will provide adequate protection against the shipping cask falling into the fuel storage pool due to a failure of the crane or lifting cables.

13.0 ACCIDENT ANALYSIS

We have reviewed the potential accidents analyzed by the applicant and performed independent calculations to determine the potential doses associated with these accidents. A comparison of calculated doses for several potential accidents is shown in Table 13.0.

Our Review considered a spectrum of accidents which we currently consider for a pressurized reactor facility. However, the following discussion is limited to those accidents which result in potential doses approaching the Part 100 guidelines. These accidents are: the loss-of-coolant (LOCA), the steam generator tube failure, and the fuel handling accidents.

13.1 Loss-of-Coolant Accident

The design basis loss-of-coolant accident for the TMI#2 plant is similar to that used for other PWR's, i.e., a double-ended break in the largest pipe (36 in. ID) in the primary coolant system. This potential loss-of-coolant accident has been analyzed to determine the capability of the engineered safety features to limit the consequences associated with the accident and to establish the acceptability of certain of the plant design features. In the analysis, consideration has been given to fuel clad temperatures, extent of metal-water reaction, hydrogen accumulation in the containment, and effects of blowdown forces on the reactor vessel and internals.

The applicants' analysis of this accident indicates: (1) no fuel melting, (2) low clad temperature (< 2100°F), (3) acceptable containment pressures (< 53 psig), (4) a total 2-hour thyroid dose at exclusion radius of 2.5 rem, (5) less than 1% metal-water reaction based on minimum engineered safety feature operation.

We and the applicants have analyzed the loss-of-coolant accident assuming the TID-14844 fission product release model. The calculated dose for this accident is shown in Table 13.0. The major difference between our

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calculated dose of 450 rem and the applicants' value of 125 rem results from the iodine removal constants used for the containment spray system and the percent iodine assumed to be in non-removable form. Our calculated dose for this accident is in excess of the 10 CFR 100 guideline value. Based on our analysis, the applicants must either provide additional support of the capacity of the iodine removal system inside the containment or else make necessary modifications. This matter was discussed in Section 9.2.

13.2 Fuel Handling Accident

The applicants have presented an analysis based on the failure of 56 fuel rods occurring during a fuel handling operation. The fission product release is assumed to occur under 10 feet of water with only the fuel rod gap activity released. It was assumed by the applicants that the accident occurred following a 24-hour shutdown period. In addition, they assumed the failure to occur in an element that had been operated at a power level of twice the average. The applicants use a reduction factor of 0.01 for iodine retention by the water and a reduction factor of 0.1 for the charcoal filters in the ventilation system. The doses calculated using these assumptions are given in Table 13.0.

Our evaluation of the fuel handling accident assumes the failure of all [illegible] fuel rods of a single fuel assembly. The fission product iodine released from the fuel rods is 10% of the iodine contained in the fuel pins. We assume the iodine released from the fuel rods is reduced by a factor of 0.1 by water retention, and a factor of 0.1 by the ventilation system filters. Using the applicants' other assumptions regarding fuel

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peaking factors and decay time, we calculated a thyroid dose of 295 rem. We have discussed our results with the applicants. They indicated that the 24 hour period for refueling was not really possible and that more like a 72-hour period would be expected. The result of this proposed change in assumptions is to reduce the thyroid dose to 195 rem as in Table 13.0. Our calculated doses to the thyroid and whole body resulting from the fuel handling accident are below the 10 CFR 100 guideline values.

13.3 Steam Generator Tube Failure

The applicants have analyzed the accident consequences of a double-ended steam generator tube failure (435 gpm leak primary to secondary) with the reactor operating with 1% of failed fuel. The radioactivity released is the fission product inventory contained in 1.6×10^8 cc of primary coolant which leaks to the secondary condenser and out the plant vent.

The applicants used a partition factor of 10^{-4} for iodine in the condenser. Based on the above assumptions, the thyroid dose at the 610-meter exclusion radius is 0.018 rem.

Our evaluation of this accident assumes the same amount of primary coolant leaks to the secondary system; however, due to uncertainty as to how the secondary release of fission products occurs, we assume no partitioning factor for the gaseous fission products. Even with this conservative model the calculated dose is within the 10 CFR 100 guideline values.

TABLE 13.0

ACCIDENT DOSES CALCULATED BY STAFF & APPLICANT

Incident	Exclusion Radius (610 meters) Staff		Exclusion Radius (610 meters) Applicant		Low Population Zone (2 miles) Staff	
	Thyroid	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body
Loss of Coolant (rem)(c)	450	7	125	7	290	3.0
Steam Line Break (rem)	75	-	1.0	1.0	(a)	(a)
Gas Decay Tank Failure (rem)	---	4	1.0	3.0	(a)	(a)
Steam Generator Tube Failure (rem)	270	3	1.0	2	(a)	(a)
Fuel Handling Accident (rem)(b)	195	9	1.0	1	(a)	(a)

(a) For these short duration accidents, the low population zone dose would be a fraction of those calculated for the exclusion radius and well below Part 100 guidelines.

(b) The staff analysis assumed 208 failed fuel pins (one fuel bundle). The applicants' analysis assumes 56 failed pins. The remaining differences between calculated doses are due to water retention and iodine released values used.

(c) Major difference between calculated doses is due to credit for iodine removal by the containment spray system. This accident is referred by the applicants as the maximum hypothetical accident (MHA) using TID-14844 fission product release model. See Section 9.2 for evaluation of iodine removal capability.

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13.- <u>Radiolytic Hydrogen Generation</u>

In Supplement No. 3, question 2.2A, the applicants presented an evaluation of the potential hydrogen concentration which could occur in the reactor building following an LOCA. The evaluation considers the source of hydrogen generation to be (a) radiolytic and (b) chemical. We have also discussed the results of the evaluation at various meetings with B&W and the applicants. B&W presented the results of its evaluation of the hydrogen concentration within the reactor building (1) following an LOCA and (2) following the Maximum Hypothetical Accident (MHA). The resultant offsite doses from either of these cases were indicated to be less than the 10 CFR 100 guideline values. For these analyses, containment purging was maintained at a rate to limit the hydrogen concentration below 3.5 v/o within the containment. A B&W topical report is expected to be issued this year on the proposed means to cope with the effects of radiolysis.

The applicants have indicated in Supplement No. 3, question 2.2A, that in the event that purging is not permitted as a means of controlling the hydrogen concentration within the containment, that other means such as recombiners would be considered to control the hydrogen concentration.

Although the acceptable methods for controlling the hydrogen concentration within the reactor containment have not been established, we conclude that the applicants' commitment to provide other means of controlling the hydrogen if venting is not permitted is adequate for the construction permit review.

We shall continue to follow the radiolytic hydrogen problem and its resolution during the TMI#2 operating license review when sufficient definition and methods of controlling the hydrogen concentration in the reactor containment have been established.

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1.0 RESEARCH AND DEVELOPMENT

During the development of the Babcock and Wilcox pressurized water reactor, certain areas of the design have required a research and development program for additional information to meet required design criteria. The following areas have been included in a research and development program:

a. xenon oscillations

b. core thermal and hydraulic tests

c. fuel rod clad failure

d. high burnup fuel tests

e. internal vent valves

f. control rod drive test

g. once-through steam generator

h. self-powered detector test

i. blowdown forces on reactor internals

j. chemical spray system

k. effects of radiolysis

The applicants provided information in Amendment 5, Supplement No. 3 for each of the above research and development areas relating to program objectives and scheduled completion dates except for the effects of radiolysis. We have included this item in the above list of research and development areas.

In Supplement No. 6, Item A, the applicants presented a discussion on all asterisked items identified in ACRS letters as areas of concern for large pressurized water reactor designs. Each of the items is discussed as to action status. We have reviewed the information furnished by the applicants and conclude that the program commitments are adequate for the construction permit stage of review.

We are reviewing the applicants' analysis regarding the effects of radiolysis. Venting has been considered as a means to cope with this matter. However, other means can be effective in the event that venting is not acceptable as the only method. Such means include the use of recombiners.

15.0 GENERAL DESIGN CRITERIA

The principal design criteria for the Unit No. 2 were developed based upon meeting requirements set forth in the "General Design Criteria for Nuclear Power Plant Construction Permits," of 10 CFR Part 50. The applicants have responded to each of the 70 criteria indicating how the principal architectural and engineering criteria of the plant will meet these criteria. We have reviewed the TMI#2 design criteria and find they meet the intent of the Commission's 70 General Design Criteria.

16.0 ADDITIONAL REVIEW MATTERS

16.1 Technical Qualifications

The Jersey Central Power & Light Company (JCPL) and Metropolitan Edison Company (Met Ed) as co-owners, have arranged for the purchase of equipment, consulting, engineering and construction services for the construction of the Three Mile Island Nuclear Power Station Unit No. 2. As co-owner, Met-Ed is responsible for the design, construction, and operation of the unit. JCPL and Met-Ed are owned by the General Public Utilities Corporation (GPU), a holding company which also owns two additional utility companies, New Jersey Power and Light Company (NJPL) and Pennsylvania Electric Company (Penelec). The GPU current nuclear electric generating program consists of Oyster Creek Unit No. 1 (JCPL to be operated in 1969, Three Mile Island Unit No. 2 (Met-Ed being reviewed for construction permit, Three Mile Island (Met-Ed) Unit No. 1 under construction, and the Saxton research and experimental nuclear unit.

The engineering and management responsibility for the design and construction of the TMI#2 rests with the Project Director, who is vice president and chief engineer in the Met-Ed Company.

GPU has formed a Nuclear Power Activities Group (NPAG) to provide direct technical assistance and guidance to the nuclear project managers of its subsidiary companies during construction and operation of nuclear electric generating stations.

In addition to the support from NPAG, the TMI#2 project manager has available on a consulting basis two additional companies, Pickard & Lowe Associates and Burns and Roe, Incorporated. The following organizations and responsibilities have been identified by the applicants.

TABLE 16.1

TMI#2 PROJECT ORGANIZATION

ORGANIZATION	FUNCTION
Metropolitan Edison Company (Met-Ed) Jersey Central Power & Light Company (JCPL)	co-owners
Burns & Roe, Inc. (B&R)	architect-engineer
Metropolitan Edison Company	responsible for design construction and operation
United Engineers and Constructors	construction manager
Pickard & Lowe Associates (P&L)	design consultant
Babcock & Wilcox (B&W)	nuclear steam supplier
MPR Associates (MPR)	provide quality assurance assistance
GPU Nuclear Power Activities Group (NPAG)	technical assistance
Schupack & Associates	structural consultant
Gilbert & Associates, Inc.	architect-engineer cooling towers and swithyard and aircraft design consultants

In Appendix 1B, Figure 1B1 of the PSAR, the applicants presents an organization chart showing the relationship and line of authority proposed for the quality assurance program to assure the design and construction of TMI#2.

Met-Ed will station a full-time resident quality control representative at the construction site to maintain surveillance over the construction and installation work. Work at the site or in any subcontractor's shop can be stopped on discovering any deviation from applicable codes, specifications or quality control requirements by the Met-Ed Project Director Met-Ed resident Quality Control Representative, or the site construction manager.

Prime contractors associated with TMI#2 are competent in their areas of specialization and are considered to be technically qualified to design and build TMI#2. The proposed schedules for the #1 and #2 plants is as follows:

	Unit #1	Unit #2
Start Construction	2/68	7/69
Load Fuel	12/70	9/72
Commercial Operation	5/71	5/73

This schedule promotes continuity of personnel during construction, and it allows five months more for construction, and three months for testing and startup, on Unit #2 than on Unit #1. We consider the proposed schedule for Unit #2 to be realistic.

In general, the quality assurance program for TMI#1 has been satisfactory to date. The principal problem reported has been a few cubic feet of voids, owing to debris on the bottom of the forms, in the concrete ceiling of the tendon gallery. We consider that the review and evaluation, and corrective action by the applicants and their contractors in this instance are satisfactory

16.2 Conduct of Operations

The two plants will be adjacent to each other, and will share certain facilities such as new fuel storage and the electric switchyard but each will have its own control room. Technical support and maintenance groups will service both units, and are considered to be satisfactory. There will be a single Supervisor of Operations, who reports to the Plant Superintendent. On each shift for the two units, there will be 1 Shift Supervisor, 1 Shift Foreman, 4 Control Operators and 4 Auxiliary Operators. We consider this to be satisfactory with one reservation. The applicants proposed that only the Shift Supervisor will be a licensed senior reactor operator. The Shift Foreman and four control operators will be licensed reactor operators. However, in our judgment, for a two-unit station, there should be two senior reactor operators on duty at all times. We have informed the applicants that this matter could be resolved at the operating licensing review stage but that they should make provisions in the preoperational training program to accommodate such a requirement.

The applicants plan to use the emergency plan developed for Unit 1 with special provisions to include possible emergencies which could occur while Unit 2 is under construction and Unit 1 is operational.

We have reviewed the applicants' submittal for training, administrative controls, review and audit, and their emergency plan. All are similar to their plans for TMI#1 which we have previously found to be satisfactory. A more comprehensive in-depth review of the applicants' final plans in these areas will be made at the operating license stage.

On the basis of our review of the above organizational scheme, we have concluded that the Met-Ed organization is acceptable with regard to the technical personnel and nuclear capability. The relationship between Met-Ed, GPU, B&R (architect-engineer), UEC (construction manager), B&W and other consultants is such that the design and construction organization is adequate. We have also considered the experience gained from Oyster Creek Unit No. 1 construction and operation and construction of Three Mile Island Plant which will further increase and broaden the applicants' technical competence and nuclear capability.

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.6.3 Industrial Security

We have discussed with the applicants the proposed industrial security measures which will be employed during construction and operation of TMI#2. Access to the Three Mile Island site can be by either the north or south bridge or by boat. The applicants have indicated the access by the north and south bridges will have control stations to control access to the site during construction and operation of TMI#1 and #2. The applicants also indicated that during construction all inspectors and supervisory personnel would be instructed to watch for any signs of potential sabotage.

It is probable that many acts could be easily committed that would affect the availability of the station; however, we consider it highly unlikely for a large scale act of industrial sabotage at the station to occur that would affect the health and safety of the public. We intend to continue our review of the industrial sabotage potential on a general basis throughout the post-construction review and operating license review of the facility.

17.0 QUALITY ASSURANCE

For our evaluation of the quality assurance program for the Three Mile Island No. 2 facility, we used the guidelines based upon the proposed amendment to 10 CFR 50, Appendix B Quality Assurance Criteria for Nuclear Power Plants dated April 17, 1969.

For the Three Mile Island #2 project, the applicants are establishing a comprehensive quality assurance program under their own quality assurance organization. The quality assurance program for the TMI#2 will be applied to the nuclear portions of the plant and is described in Appendix 1B of the PSAR. In particular, it will be applied to the reactor coolant system and directly associated auxiliary systems, the containment system, the engineered safety features, the fuel handling system and the radioactive waste disposal system.

MPR Associates, Inc. has been retained by the applicants as an additional quality assurance organization. Met-Ed/JCPL and MPR quality assurance organization is intended to provide unification, overall direction, guidance, and surveillance over the quality assurance efforts of the Architect-Engineer Burns & Roe (B&R), the nuclear steam supplier Babcock & Wilcox (B&W), and the constructor United Engineers & Constructors (UE&C). UE&C as construction manager and principal constructor has responsibility for assuring adequacy in all construction activities. B&R has responsibility for the design portions of the plant other than the Nuclear Steam Supply system which will be furnished by B&W.

A three-level check and balance system has been set up within the quality assurance program to provide the organization and line of authority to assure that components, systems, and equipment are designed, manufactured, installed and tested to meet all code and specification requirements. The line of authority in the three-level quality assurance program is shown in figure 1B-1 of the PSAR. Separation of the personnel performing a surveillance function appears to be adequate to isolate these functions from the pressure of schedules.

The applicants have incorporated into the quality assurance program a general type specification which will be contractually enforced. This general specification establishes the quality control requirements for the fabrication installation, testing, and construction of reactor coolant system, containment system, engineered safety features, fuel handling system, and the radioactive waste handling system. The generic specification presented in Section 4 of Appendix 1B was developed by the applicants.

Jersey Central Power & Light Company and Metropolitan Edison have indicated that they are in the process of writing a Quality Assurance Program with assistance from GPU and MPR. This written quality assurance program will be used as a procedure manual throughout construction of TMI#2 by the applicants to assure the quality assurance program presented in Appendix 1B is carried out. JCPL and Met-Ed have indicated they will have monthly project progress meetings in which all contractors and their quality control personnel will meet to review status and follow-up of quality assurance matters.

Both B&R and B&W will have quality assurance plans, programs, and organizations, as required by the generic functional specifications for a QA program. The applicants appointed a GPU quality assurance engineer responsible for directing and coordinating the entire program, including MPR efforts. Met-Ed/JCPL, as co-owners of the TMI#2 facility, intend to use the GPU engineering and specialist groups in extensive support of the applicants' quality assurance team. The applicants (Met-Ed/JCPL) intend to review all specifications pertaining to the nuclear portion of the plant and have retained final responsibility for establishing and enforcing adequate quality assurance requirements and procedures.

In most areas we found the program plan to be excellent. These areas include:

1) The applicants' intent to establish their own quality assurance organization using qualified quality assurance personnel at the site;

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(2) The provision, on the part of the applicants, that duties of their quality assurance personnel will be solely QA duties, i.e., they will have no other detailed technical or line organizational duties;

(3) The applicants' decision to have MPR Associates, Inc., report directly to the applicants and to aid in the promulgation and execution of specific, systematic QA plans and surveillance activities;

(4) The intent of the applicants in conjunction with GPU to use their engineering staff to support the Met-Ed/JCPL and MPR organization in the review of specifications, attendance at meetings, surveillance and audit functions, etc., including the intent of applicants to assure that independent checking of designs and specifications will be carried out between Burns & Roe, UE&C and B&W (between these and other important organizations) at important interfaces, to the degree deemed necessary to assure compatibility;

(5) The steps being taken to assure close association and interchange of information including a written QA program plan plus the commitment in the PSAR that adequate written QA procedures will be provided; and

(6) A commitment on the part of the applicants to establish and implement written procedures to cover each of the "AEC's elements of quality" and to require important major contractors to do the same.

Based on our review of the quality assurance program proposed by the applicants, we conclude that it will provide adequate quality assurance and control of the design, construction and erection of the TMI#2 facility. However, we are concerned with the degree of participation of the utility in the quality assurance program. During the post-construction meetings with the applicants, we shall continue to review the quality assurance program to determine its effectiveness in providing the functions intended to assure that critical systems of the plant meet the design specifications. The applicants' generic specifications which will include requirements for written procedures, will assist the Division of Compliance in inspection of the plant throughout its construction.

18.0 INSERVICE INSPECTION

In Item 8, Supplement No. 6 of the TMI#2 PSAR, the applicants discussed their contemplated inservice inspection for surveillance of the reactor coolant system. The N-45 Committee Code for Inservice Inspection has been considered by the applicants to determine its effect on the reactor plant design regarding inspection access capability. The design of the TMI#2 incorporates a reasonable degree of accessibility to facilitate the performance of inservice inspection of the primary coolant system.

The following items will be pursued by the applicants to develop an inservice inspection program for the TMI#2.

1. Continue to review the facility design development to ensure suitable access is provided around all nuclear steam supply system components and piping to permit performance of the inservice inspection program.
2. Continue the program of review and surveillance of remote inspection devices and development program to permit making any minor adjustments in the facility design which may be necessary to utilize these devices for inservice inspection, if appropriate, when they become available

3. Develop a schedule defining the scope, type, and frequency of inservice inspection. The schedule would be incorporated into the Technical Specifications for the facility. Inspection points selected will be evaluated on the basis of irradiation effects, configuration, materials, operating conditions, stress levels, transient effects and other items which could lead to crack initiation or propagation. The program developed will be one which assures the continued integrity of the reactor coolant system consistent with minimum downtime and radiation exposure for personnel.

On the basis of our review of the applicants' program concerning inservice inspection, we conclude that it represents a reasonable approach to development of an adequate inservice inspection program. We will follow up on the applicants' action during the post-construction permit phase and therefore intend to require the applicants to submit a status report of their design provisions for inservice inspection including their plans for inservice inspection of the primary coolant pump flywheels for our review.

Detection of leakage from the reactor coolant system in the TMI#2 plant can be made by one or a combination of the following:

(1) Monitoring of primary system coolant inventory
(2) An increase in airborne activity in the containment atmosphere.
(3) An increase in activity in steam generator secondary coolant or in condenser vacuum system.
(4) Periodic visual inspection during shutdown.
(5) Monitoring of sump inventories

We conclude that the above methods of detecting primary coolant leakage are adequate for the construction permit review. However, during the post-construction permit stage, we intend to review the overall capability of the system considering sensitivity, response time and limits for inclusion in the Technical Specifications.

19.0 CONCLUSION

On the basis of our review of TMI#2 several matters were identified that require continuing review after issuance of the construction permit. This action is acceptable to us because in many cases the carryover items are of a generic concern to the B&W plant design and light water reactors whereas in other instances, the additional design details that will be necessary to resolve these matters will be made available prior to construction. Based on our review and discussions of these matters with the applicants, we conclude that they can be resolved adequately within about a year following issuance of the construction permit.

The matters of general concern to this class of light water reactor plants that will require a continuing review effort are:

(1) combined (LOCA/Seismic) loading conditions on internals, testing aspects on the vent valves, and confirmatory vibrational test program for core internals (Section 6.1)

(2) pressure vessel surveillance program (Section 6.2)

(3) ECCS initiation signal diversity (Section 9.1)

(4) iodine removal effectiveness (Section 9.2)

(5) effects of radiolysis (Section 13.4)

(6) steam generator tube failure coincident with an LOCA (Section 13.[illegible])

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The matters of particular concern to TMI#2 that will require resolution as the necessary information becomes available are:

(7) flooding protection requirements (Section 2.5)

(8) environmental monitoring program (Section 2.6)

(9) tendon anchorage design and surveillance program (Section 5.5)

(10) analysis and inspection of flywheels (Section 7.2.2)

(11) inservice inspection provisions (Section 18.0)

(12) leakage detection (Section 18.0)

In view of the actions proposed to resolve the above items, we conclude that the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safey of the public.

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APPENDIX A

TMI #2 CHRONOLOGY OF ACTION

Application filed with three volumes Preliminary Safety Analysis Report (PSAR)	April 29, 1968
Amendment No. 1 filed (clarification of Unit 2 core power level and net electrical output)	May 24, 1968
Amendment No. 2 filed (Errata plus Grouted Tendon Test Program Results)	July 1, 1968
Amendment No. 3 filed (response to staff's request of June 25, 1968 concerning reanalysis of Probable Maximum Hurricane flood height)	September 3, 1968
Amendment No. 4 filed (response to staff's request of September 19, 1968)	November 4, 1968
Amendment No. 5 filed (response to staff's request of October 18, 1968)	November 29, 1968
Amendment No. 6 filed (response to staff's request of October 18, 1968 plus complete PSAR revision changing plant site to Three Mile Island site)	March 10, 1969
Amendment No. 7 filed (response to several areas discussed in meetings with the applicants)	March 17, 1969
Amendment No. 8 filed (revision and additional information regarding diesel load and size and pressure test surveillance)	April 16, 1969
Amendment No. 9 filed (change of responsibility for design and construction to Met-Ed)	May 7, 1969
Amendment No. 10 filed (final up-dating of PSAR including additional flywheel write-up)	June 2, 1969
Request for exemption to permit installation of the tendon access gallery	May 1, 1969
Supplemental information relating to public need for exemption request	May 22, 1969

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