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U. S. ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Note by the Director of the Division of Reactor Licensing

The attached report has been prepared by the Division of Reactor Licensing for consideration by the ACRS at its January 1969 meeting.

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ABSTRACT

Consolidated Edison Company of New York, Inc., by application dated April 26, 1967, has requested a construction permit for a nuclear power reactor to be located on the applicant's Indian Point site on the east bank of the Hudson River in upper Westchester County, New York. This unit is designated as Indian Point Nuclear Generating Station No. 3. Site characteristics have been evaluated and found to be adequate.

The proposed unit includes a four-loop Westinghouse nuclear steam supply system with a design power rating of 3025 Mw (t). Peaking factors proposed are identical to those used in the analyses of the Diablo Canyon reactor; however, since the Indian Point Unit No. 3 average core power density is lower, the peak values of core heat flux and linear heat generation rate are lower than those approved for Diablo Canyon.

The design of the unit is essentially the same as that of the previously approved Indian Point Unit No. 2. Differences between Unit No. 2 and Unit No. 3 exist in the areas of power level, peaking factors, emergency core cooling system design, post loss-of-coolant accident protection, hydrogen control, and charcoal filter design. These differences have been identified, evaluated, and found to be adequate.

Analyses have been made of the consequences of various postulated accidents. Credit for organic iodide removal by the charcoal filters is required to meet the 10 CFR 100 guideline at the outer boundary of the low population zone for the duration of the accident. We have analyzed the experimental evidence and concluded that organic iodine removal can be accomplished by a properly designed charcoal removal system. In all accidents analyzed, the potential offsite radiological hazards are within the applicable guidelines.

We believe the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

1.0 INTRODUCTION

On April 26, 1967, the Consolidated Edison Company of New York, Inc. (Con Ed) submitted an application for a construction permit for a nuclear power unit to be located at the Indian Point site on the east bank of the Hudson River in upper Westchester County, New York. The proposed unit will employ a pressurized water reactor nuclear steam supply system designed and furnished by Westinghouse Electric Corporation, the prime contractor, under a turnkey contract. Westinghouse has engaged United Engineers and Constructors to serve as the architect-engineer.

The reactor is designed for a power output of 3025 Mwt with an ultimate capacity of 3217 Mwt. These are equivalent to net electrical ratings of 965 and 1033 Mwe respectively. Accidents and engineered safety features have been analyzed on the basis of the "stretch" capacity.

The design of the unit is essentially the same as that of the previously approved Indian Point Unit No. 2. Differences exist in the following areas: power level, emergency core cooling system, post-loss of coolant accident protection, and core peaking factors. The principal actions relating to the processing of the Indian Point Unit No. 3 application are presented in Table 1.0.

The review of this application has taken longer than that of previous applications because there was a delay in the start of our review and Con Ed delayed about six months in answering our initial request for additional information.

TABLE 1.0

Submittals	Subject	Date
Original	Preliminary Safety Analysis Report, Volumes 1, 2, and 3	April 26, 1967
Amendment 1	First Supplement to PSAR (Response to DRL comments dated February 19 and July 1, 1968)	August 30, 1968
Amendment 2	Second Supplement to PSAR (Response to DRL comments dated February 19 and and July 1, 1968)	September 16, 1968
Amendment 3	Third Supplement to PSAR (Replacement pages to PSAR)	October 18, 1968
Amendment 4	Fourth Supplement to PSAR (Response to DRL comments dated July 16, 1968)	October 31, 1968
Amendment 5	Fifth Supplement to PSAR (Response to DRL oral comments on October 11, 1968)	November 4, 1968
Amendment 6	Sixth Supplement to PSAR (Replacement pages to PSAR responding to DRL comments dated November 20, 1968)	November 25, 1968
Amendment 7	Seventh Supplement to PSAR	December 9, 1968
Amendment 8	Eighth Supplement to PSAR	December 9, 1968

2.0 SITE

2.1 Description

The proposed unit is to be located on the applicant's Indian Point site in upper Westchester County, New York, approximately 24 miles north of the New York City boundary line. Unit No. 3 will be built adjacent to and south of the presently licensed Unit No. 1. Unit No. 2, which is presently under construction, is located adjacent to and north of Unit No. 1. This site has most recently been reviewed by the Committee in connection with the construction permit review of Unit No. 2. For this reason, we have presented a summary of the important site related features below and emphasized those areas in which our current review differs from that of the construction permit review of Unit No. 2. In addition, the population distribution in the vicinity of the Indian Point site was discussed in the course of the Committee's review of the Burlington, Zion, and Consolidated Edison Units 4 and 5 sites.

2.2 Population Distribution

The population in the vicinity of the site is high. The estimated population distribution is presented below. For comparison, the Zion distribution is also presented.

CUMULATIVE POPULATION

Distance (Miles)	Indian Point 1960	Indian Point 1980	Zion 1960	Zion 1985
1	1,080	2,100	1,000	2,340
2	10,810	20,900	8,800	25,600
3	29,630	59,520	18,300	50,000
4	38,730	78,800	29,700	75,000
5	53,040	108,060	52,600	106,000
10	155,510	312,640	188,800	390,000

The minimum exclusion distance from Unit No. 3 is 350 meters (0.22 mi), and the nearest corporate boundary of Peekskill, the population center, is 1000 meters (0.63 mi) from the unit. Using these figures, a literal interpretation of 10 CFR 100, the Commission's site criteria, which states that the population center distance should be at least 1-1/3 times the low population distance, would require the outer boundary of the low population zone to be less than 750 meters from the unit. Nevertheless, Con Ed has chosen 1100 meters as the outer boundary of the low population zone because of the limited population within this distance from the plant. We conclude that this is acceptable (1) because of the limited population within the low population zone (66), and (2) because Peekskill is of a generally industrial nature in the vicinity of the unit so that resident population is low and control of the people would not be difficult.

2.3 Meteorology

The meteorology of the Indian Point site is governed by its position in a deep river valley. Consequently, wind direction generally follows a pronounced diurnal cycle with unstable (lapse) flow in the upriver direction during the daytime and stable flow in the downriver direction at night.

The meteorological model proposed by the applicant is not justified since the applicant does not include data on the specific joint frequency of stability-wind speed-direction persistence from the site, nor are any such data available from long-term measurements in the vicinity of the site. For this reason we have used our standard meteorological model for accident dose calculations. With this model we assume a 1 meter per second wind speed in the same direction under inversion conditions for a period of 8 hours; meandering of the plume centerline over a 22-1/2° sector under inversion conditions for the remainder of the first 24 hour period; and variable stability, wind direction, and wind speed for the remainder of the accident. Our consultant, Air Resources Laboratory, ESSA, concurs in these assumptions. The ESSA report will be transmitted to the Committee prior to the January meeting. We have used this model instead of that proposed by the applicant in assessing the consequences of accidental release of radioactivity. (Section 5.0 of this report).

2.4 Geology and Seismology

Unit No. 3 will be founded on a hard limestone that is well jointed but noncavernous. We have reviewed the analysis of the site geology in the PSAR and examined the boring logs, as has our geological consultant, the U. S. Geological Survey (USGS). The applicant's analysis has also been compared with the available literature. As a result of this evaluation, we have concluded that (1) the applicant's analysis presents an adequate appraisal of site geology, (2) the limestone will provide an adequate foundation for the proposed facility, and (3) there are no known active faults or other geologic structures that could be expected to localize earthquakes in the immediate vicinity of the site. The comments of the USGS which support this conclusion will be sent to the Committee prior to the January Meeting.

The seismicity of the site has been evaluated by the U. S. Coast and Geodetic Survey (USC&GS). Based on the review of the seismic history of the site and of the related geologic considerations, the USC&GS concludes that the applicant's proposal to use accelerations of 0.10g for the Operational Basis Earthquake and 0.15g for the Design Basis Earthquake is acceptable. These are the same seismic acceleration figures used in our Unit 2 evaluation. Copies of the USC&GS report have been transmitted to the Committee.

2.5 Hydrology and Flooding

Our consultant from CERC has not completed his final evaluation of site flooding elevations. This evaluation will be transmitted to the

Committee in a supplemental report when the applicant's final analysis has been evaluated. On the basis of discussions with CERC, Con Ed site flooding does not appear to be a problem in this case.

2.6 Gas Pipeline

Two gas transmission lines traverse the site. They are located 380 feet from the primary water storage tank and 660 feet from the containment. The first is a 30-inch line with a wall thickness of 0.438 inch calculated to withstand an internal pressure of 1520 psi before yield stresses are reached. It has been tested to 1390 psi. The design and operating pressure is 750 psi. The second is a 26-inch line with a wall thickness of 0.281 inch. The line is capable of withstanding a pressure of 1125 psi without yielding. It is designed to operate at 750 psi and is presently operating at 650 psi in accordance with a directive from the New York State Public Service Commission. Thus, the 26-inch line operates at 59% of yield while the 30-inch line operates at 48% of yield. Corrosion control for the lines is provided by both insulation and cathodic protection.

The pipelines are separated from important structures (e.g., service building, primary auxiliary building, and containment) by approximately 500 feet. Since these structures are designed both for tornado winds and missiles, and for earthquake loadings, the applicant has stated that adequate protection is provided against damage from concussion or flying pipe fragments in the event of a pipeline explosion.

The plant is located in such a manner that there is over 100 feet of clear space between the plant and the closest row of trees. In the event of fire from the pipeline, the applicant believes that this will serve as a firebreak to prevent a fire from spreading to the site.

If a fire did reach the plant, there is no flammable structural material in the containment or primary auxiliary building to support combustion. In addition, automatic shutoff valves at both banks of the Hudson and at Yorktown, New York, will isolate the line, thus limiting the duration of any primary fire to less than 5 minutes.

Based on the considerations discussed above, we conclude that the gas pipelines present no undue hazard to the plant.

2.7 Environmental Considerations

The Fish and Wildlife Service (F&WL) has reviewed the application relative to the consequences of release of radioactive waste materials to the environs. They have recommended that both pre- and post-operational surveys, planned in cooperation with the appropriate Federal and State agencies, be conducted. Their comments have been transmitted to the Committee. The applicant has agreed to comply with the F&WL recommendations.

The applicant is conducting an environmental monitoring program which includes sampling of: atmospheric dust; waters of the Hudson River, a small lake onsite, nearby reservoirs, and the onsite well; vegetation; atmospheric gross gamma activity; and marine life in the Hudson River. This program has been in operation since 1958. We conclude it is adequate to determine the impact of the Unit No. 3 facility on the environment.

The site has also been reviewed by the Advisory Council on Historic Preservation. They have concluded that the probable effect upon the Stony Point Battlefield Reservation cannot be judged to be sufficiently adverse to warrant Council comment. The report of the Advisory Council has been sent to the Committee.

3.0 GENERAL PLANT DESCRIPTION

3.1 Nuclear Steam Supply System

The nuclear steam supply system consists of a standard Westinghouse four-loop pressurized water reactor. The core is divided into two radial regions with regard to UO_2 fuel enrichment. The fuel in the outer region is enriched to 3.2 w/o. The fuel in the inner region is arranged in a "checkerboard" array of assemblies enriched to 2.1 and 2.6 w/o. Part-length control rods and special full length rods which can be moved individually rather than in a bank (X-Y rods) are provided to control spatial neutron flux oscillations.

The proposed power level of Unit No. 3 is approximately 10% higher than that of Unit No. 2; however, is is approximately 7% lower than that of Diablo Canyon and the recent generation of four-loop Westinghouse designed plants. A comparison of Unit No. 3 with Unit No. 2 and with Diablo Canyon is presented in Table 3.1. This comparison indicates that greater margins are provided in the Unit No. 3 design than are available in recent Westinghouse four-loop PWR's because the peaking factors associated with the recent cores are used but the fuel rod specific power is not as high.

We have examined the thermal design of the core, noting the results of parametric studies of the effects of variations in inlet temperature, inlet pressure, mass flow rate, and peaking factors on minimum DNB ratio experienced. These studies have demonstrated that neither calibration errors nor small errors in the predicted peaking factors will significantly affect the thermal performance of the core. Based on the foregoing, we see no new feature in the design or proposed operating parameters of the nuclear steam supply system which would alter our conclusions made with reference to previously reviewed Westinghouse four-loop plants.

TABLE 3.1

Item	Indian Point 3	Diablo Canyon	Indian Point 2
Total Heat Generation, Mw(t)	3025	3250	2758
Maximum Specific Power, kw/ft	17.6	18.9	18.5
Maximum Heat Flux, $Btu/hr\text{-}ft^2$	543,000	583,000	570,800
Average Heat Flux, $Btu/hr\text{-}ft^2$	193,000	207,000	175,600
Average Mass Velocity, $lb/hr\text{-}ft^2$	2.53×10^6	2.54×10^6	2.56×10^6
Nominal Inlet Temperature, °F	549.7	539	543
Minimum DNBR at Nominal Conditions	1.82	1.81	1.81
Fq - Heat flux hot channel factor	2.82	2.82	3.25
F H - Enthalpy hot channel factor	1.70	1.70	1.88
Fuel Enrichments, w/o			
Region 1	2.1	2.2	2.23
Checkerboard Region	2.6	2.7	2.38
	3.2	3.3	2.68

Instrumentation is required to assure that the power distribution is adequately controlled. The applicant has stated that the four external flux monitors will be used to detect abnormal power patterns. The in-core monitors for Unit No. 3, as presently proposed, are six traveling flux probes which together may traverse any of 58 thimble locations in the core. These in-core channels are not designed to operate in the core at full power for more than a few months. The applicant believes that test programs (primarily at SENA) will adequately demonstrate the capability of the external long ion chambers to detect power patterns within the core. As discussed recently with the Committee, our position in this regard continues to be that information from in-core monitors must be provided to an operator so that the part length rods can be positioned for proper axial power shaping, unless, at some later date, experience shows that the external monitors can detect in-core anomalies with adequate sensitivity. The applicant has been informed of this position and, as in previous cases states that provision will be made for installation in in-core detectors should the Commission require it at some later date.

3.2 Sharing

Unit No. 3 is physically separated from Units No. 1 and 2. Except for the electrical interconnections noted below, which are used if the portion of the normal offsite electrical power between the Buchanan substation and Unit No. 3 is lost, Unit No. 3 will be independent of the other units on site. Separate facilities are provided at Unit No. 3 to house health physics, radiochemistry laboratories, counting rooms, maintenance shops, first aid and administrative services. A separate control room is provided. Sharing is limited to features such as parking facilities, potable water supply, fire water supply, and sanitary sewage.

The 138 kV feeder from Buchanan substation to Unit No. 2 is connected underground through two circuit breakers to the Unit No. 3 startup transformer. This feeder can be used if the normal 138 kV feeder from Buchanan to the Unit No. 3 substation is disabled. Similarly, the 6.9 kV supply, which is automatically connected on loss of the 138 kV supply, can be fed from either the 21 MVA gas turbine generator, the 13 kV underground feeder from Buchanan, or the 6.9 kV auxiliary bus of Unit No. 2 which, in turn, can be supplied from either the Unit No. 2 generator or a 138 kV feeder from Buchanan substation.

As stated above, sharing between Unit No. 3 and Units No. 1 and 2 is minimal. We can identify no shared feature which would impair the safety of any unit on site.

3.3 Auxiliary Systems

The auxiliary systems (Chemical and Volume Control System, Component Cooling Water System, Service Water System, etc.) provided for Unit No. 3 are similar in design to those provided in other pressurized water reactor plants. They represent an improvement over those provided for Unit No. 2 in that no single failure, either active or passive, can negate the ability of the component cooling water system acting in conjunction with the service water system to reject decay heat to the Hudson River, the ultimate heat sink. This capability is provided by double header arrangements.

In addition, since the service water intake structure might be demolished if one of the Liberty ships moored nearby in the Hudson River became free during the probable maximum hurricane and were driven into the structure by the storm, an alternate service water supply system will be provided which will be located such that it cannot be disabled by the storm. Details of this alternate system have not been submitted. We will review the design as a follow-up item during construction and at the operating license stage of our review.

3.4 Containment Structural Design

Consolidated Edison has engaged Westinghouse Electric Corporation to design and, as prime contractor, construct the Indian Point Unit No. 3. Westinghouse has engaged United Engineers and Constructors to provide the design of certain portions of the plant. This is identical to the organizational arrangements for Indian Point No. 2.

3.4.1 General Structural Design

The foundation material at the site from the surface down consists of a finegrained phyllite, a schist, and limestone, with bedrock lying very close to the surface. Unit No. 3 will be located on the limestone, which is fractured and jointed, making it permeable to ground water, but is hard, not cavernous, and can sustain up to 50 tons per square foot. It is therefore quite capable as a foundation material for this facility, as it is for

Units No. 1 and 2. Consolidated Edison has stated that no rock grouting is to be utilized at this site.

The Class I seismic design criteria presented by Consolidated Edison, have been accepted by us and our seismic consultants. The ground acceleration values established are: Operating Basis Earthquake - 0.1g horizontal and 0.05g vertical; Design Basis Earthquake - 0.15g horizontal and 0.10g vertical. The analytical techniques to be used by the applicant in the design of Class I structures to meet these criteria are also acceptable to us and our structural consultants. The Indian Point Unit No. 2 containment has been reviewed for the same seismic criteria and has been found by the applicant to be capable of meeting these criteria, thus providing consistency in the seismic capacities of the adjacent facilities.

Our structural consultants have recommended that safety-related Class II structures or equipment be designed for approximately one-half of the values of the provisions in the Uniform Building Code for Zone 3, in order to maintain a compatability in the design levels of critical items throughout the facility. We are continuing to review this item which has also been discussed in other recent applications, and will consider this a followup item during construction. Combined Class I and Class II structures and equipment are to be designed so that there will be no functional failure of the Class I structures or systems due to Class II failures under all the various natural phenomena or accidents which have been postulated for this facility, or such Class II failures will not be permitted under the design criteria. This criterion is acceptable to us and our consultants.

Tornado loading design criteria, including protection against tornado generated missiles, have been presented by the applicant. The design tornadic velocities are 300 mph rotational and 60 mph translational, with a resulting pressure drop of 3 psi in 3 seconds. Torsional, uniform, and non-uniform containment loadings have been analyzed by the applicant. We and our structural consultants concur in the tornado criteria and design, which are similar to recent submissions for other facilities.

The spent fuel pool design has been treated by the applicant in a fashion similar to other recent applications. The walls, floor, and water cover are to provide adequate missile protection for the fuel elements. Since present reviews are inconclusive as to the extent of water that can be removed from the spent fule pool by a tornado, the applicant has advised us that no protection will now be provided against such a loss of water, but that the pool design will enable a protective cover to be placed over the pool when and if it would be deemed advisable to do so.

The design criteria for Class I piping, equipment, vessels and reactor internals as summarized in Section 15 of the PSAR are acceptable to us and our seismic consultants.

3.4.2 Containment Structural Design

The containment structure of Indian Point Unit No. 3 is similar to the containment structure of Unit No. 2. It is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome, an internal diameter of 135 feet, a height from base to dome springline of 148 feet, 4' - 6" thick cylinder walls and 3' - 6" thick dome. The base mat is 9 feet thick, supported on rock. The containment free volume is 2,610,000 cubic feet with a design pressure of 47 psig. At 1.0, 1.25 and 1.5 times the design pressure, the respective uninsulated liner temperatures will be 247°F, 285°F, and 306°F.

The containment mat has been reviewed and accepted under the exemption request granted the applicant. The two areas which were discussed in greatest depth with the applicant at this stage involved the elasticity assumptions of the rock surface on which the mat will rest, and the shear reinforcing for the mat and cylinder walls. Both areas have been clarified to our satisfaction.

The structural analysis for the reinforced concrete containment is similar to that of Indian Point Unit No. 2. The reinforcing in the structure will have an elastic response to all loads with limited maximum strains to ensure the integrity of the liner. The reinforcing steel will conform to ASTM Designation A432-65 with a guaranteed minimum yield point of 60,000 psi. The 14S and 18S reinforcing bars will be spliced only by Cadweld splices. The sampling frequency for test splices will be 10 of the first 50 splices, 5 of the next 50, 5 of the next 100, and 1 of each next successive 100 splices. Test splices will be production splices removed from the structure. The mean value of the ultimate strength of splices made during any time period shall be equal (as a minimum) to 75,000 psi, plus the standard deviation in strength from the mean ultimate strength. In addition, the mean value of the ultimate strength and the standard deviation shall show, by statistical analysis, that at least 99.0% of all of the splices will have an ultimate strength of 60,000 psi or greater. We concur with this approach and program.

Diagonal reinforcing will be utilized in addition to the horizontal and vertical cylinder reinforcing to handle the shears generated by earthquake or wind. This is in agreement with the position taken recently by ACI committee 349 on the Design Criteria for Nuclear Containment Vessels.

The containment liner will be carbon steel plate conforming to ASTM Designation A442-65, Grade 60. It will be 1/4-inch thick at the bottom, 1/2-inch thick in the first three courses (except 3/4-inch thick at penetrations), and 3/8 inch for the remaining portion of the cylindrical walls. The dome liner will be 1/2 inch thick. The liner nil ductility transition temperature will be 30°F lower than the minimum operating temperature of the liner material. The anchorage system for attaching the liner to the concrete consists of 1/2 inch diameter bent welding studs. They are spaced in a rectangular array 28 inch vertical by 24 inch horizontal at the 1/2-inch diameter plate, and 14 inch vertical by 24 inch horizontal at the 3/8-inch plate. The dome liner will have structural tees spaced at a maximum of 5 feet in each direction with a 1/2-inch diameter stud in the center of each 8 feet by 5 feet panel. Liner insulation will be provided at the lower portion of the containment. The insulation will by 1-1/4-inch thick polyvinyl chloride with a thin gauge stainless steel cover bolted to the liner and sealed at joints and penetrations. This system has been reviewed by us and our consultants for liner integrity and stability under various postulated accidents, fabrication tolerances and inaccuracies, and allowable erection tolerances. In general, the system has been adequately developed to ensure that potential liner buckling, with its attendant rupture hazard, can be controlled even if one of the anchors is missing or has failed. We and our consultants find that the liner and liner anchorage designs are acceptable.

The applicant's design criteria, for penetrations, including the personnel lock and equipment hatch openings, are satisfactory to us and our consultants.

Construction methods and quality assurance and quality control measures are described in the PSAR and, in general, are similar to those proposed for other recently reviewed facilities.

Pre-operational testing, consisting of a strength test, gross leak rate test and sensitive leak rate test, is satisfactory and consistent with previous applications. Post-operational testing will consist mainly of monitoring double penetrations and the liner seam weld channels.

3.4.3 Conclusion

We conclude that the containment and Class I structures and systems can be adequately designed, constructed and tested under the criteria presented by the applicant.

3.5 Instrumentation and Control

The Commission's General Design Criteria (10 CFR 50) and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (No. IEEE-279, dated August 28, 1968) have been used, where applicable, as our bases for judging the adequacy of the Instrumentation and Control systems.

The design of the instrumentation and control system has undergone numerous modifications since receipt of the original application on April 26, 1967. In effect, these modifications have rendered obsolete Section 7 of the original PSAR. As a result, all information and analyses derived for this portion of the report are based on the supplements, including Section 7 of the FSAR for Indian Point Unit No. 2 (Docket No. 50-247) which is referenced in Supplement No. 5 as a valid source of information. The reactor protection instrumentation and the instrumentation which initiates the engineered safety features are essentially identical to those evaluated during previous Westinghouse reviews. Departures from the design criteria being utilized in the recently reviewed Donald C. Cook plant, and areas wherein new information has been received since that review, are specifically addressed herein.

3.5.1 Engineered Safety Features

The following is a summary of the instrumentation channels that initiate the various engineered safety features.

1. Safety Injection:

   1. Low Pressurizer pressure in coincidence with Low Pressurizer Water Level (1/3 pair; i.e., P_1L_1 or P_2L_2 or P_3L_3).

   2. High Containment Pressure (2/3 logic).

   3. High Differential Pressure between any two steam generators (2/3 logic).

   4. High Steam Flow (2/4 logic).

(Items No. 3 and No. 4 above were not included in the Donald C. Cook plant).

2. Containment Spray:

   1. High-high containment pressure (2/3 in coincidence with 2/3).

3. Steamline Isolation:

   1. High Containment Pressure (2/3 logic).

   2. High Differential Pressure between any two steam generators.

(Item No. 2 above was not included in the Donald C. Cook plant).

4. Containment Isolation:

    1. High Containment Pressure (2/3 logic).
    2. High-high Containment Pressure (2/3 in coincidence with 2/3).
    3. Low Pressurizer Pressure in coincidence with Low Pressurizer Water Level (1/3 pair, i.e., P_1L_1 or P_2L_2 or P_3L_3).

5. Fan Cooling:

    1. High Containment Pressure (2/3 logic).
    2. Low Pressurizer Pressure in coincidence with Low Pressurizer Water Level (1/3 pair, i.e., P_1L_1 or P_2L_2 or P_3L_3).

(No fan cooling was proposed for the Donald C. Cook plant).

Instrumentation and control for the engineered safety features have been analyzed to assure that they can be built in accordance with IEEE-279. Detailed schematic diagrams of the current Westinghouse designs have not (with one exception) been made available to us. The exception is the Safety Injection logic circuitry up to and including the injection breakers. This circuitry has been studied in connection with the Ginna station review. The detailed circuit diagrams demonstrate Westinghouse's stated design objective to provide two independent logic circuits (relay matrices) in each safety feature system such that either matrix can perform the required safety feature action.

From our review of the above circuits, we conclude that Westinghouse designers are employing proper engineering practices to implement the requirements of IEEE-279. Our review of their safety feature systems is continuing. For the purpose of this review, we are satisfied with the proposed safety feature systems in that:

(a) the applicant has stated that they will conform to IEEE-279; and

(b) our review of typical designs indicates that Westinghouse is proceeding satisfactorily in accomplishing the above objective.

3.5.2 Separation of Protection and Control Instrumentation

In Section 10 of Supplement No. 5 the applicant has presented an analysis to demonstrate conformity to Paragraph 4.7 of IEEE-279 with respect to random single failures. We agree that Paragraph 4.7 has been satisfied.

We are pursuing with the applicant's instrumentation supplier (Westinghouse) the concern expressed by the ACRS in the Russellville letter with respect to systematic failures. Our objective is a suitable balance of design objectives in regard to functional and equipment diversity, interaction of protection and control functions, testing, and surveillance to achieve a protection system design that has adequate capability to cope with systematic failure modes as well as random failure modes. We anticipate that our evaluation of systematic failures will be completed in the Spring of 1969.

3.6 Emergency Power

We have used proposed General Design Criterion No. 39 as the basis for judging the adequacy of the Emergency Power System.

3.6.1 Offsite Emergency Power

The Indian Point Unit No. 3 station startup transformer (138/6.9 kV) is normally supplied from a 138 kV line from the Buchanan substation. The second independent offsite power supply to Unit No. 3 is the 6.9 kV connection to bus sections No. 5 and No. 6. This supply is automatically connected upon loss of the normal 138 kV supply, and can be fed from any one of three separate sources: (1) the station gas-turbine generator, (2) the underground 13 kV feeder from Buchanan Substation, or (3) the auxiliary bus of Unit No. 2.

The Buchanan substation itself has a tie line to the P.J.M. system, and two 345 kV lines to the Millwood switching station. Millwood, in turn, connects to the Niagara Mohawk and Connecticut Light & Power grids, and has two lines to the Buchanan 138 kV bus via a 345/138 kV autotransformer at Millwood.

Based on the foregoing we conclude that because of the multiplicity of power sources, in conjunction with the alternate 6.9 kV feeder in the event the startup transformer is lost, the offsite portion of the emergency power system is acceptable.

3.6.2 Onsite Emergency Power

There are four 480V emergency buses energized directly, when required, from the three diesel generator units. Two diesel generators are required to furnish sufficient engineered safety feature loads.

Although the diesel generators are not to be synchronized during emergency operations, our review indicates that the design, nonetheless, compromises their independence, and the independence of the respective buses. Specifically, if one generator does not start, the appropriate tie breakers are closed and the bus will be energized by one of the remaining two generators. The control system then selects the "two generator" loading sequence. Further, if any load does not then start, the system will attempt to connect a redundant counterpart (which was omitted from this loading sequence because of power limitations). If all three generators start, another loading sequence is followed which also has the provision for picking up alternate loads in the event of single failures.

We believe that the compromising of independence is not warranted by the limited increase in reliability thus obtained. The theoretical improvement gained by cross-connecting sources and loads is at most a factor of 2. When this is weighed against the potential decrease in reliability (possibly a factor of 100) arising from loss of independence, the proposed design is difficult to justify. We have expressed our concern to the applicant, and we understand that alternate designs are being considered. Accordingly, we will require that the final design of the onsite emergency power system be reviewed by the staff prior to installation.

A second area of concern is the lack of criteria relating to loading margins for the diesel generators. In response to Question 6(e), Supplement 7, the applicant implies that the generators can be safely loaded to 2250 kW, the "1/2 hour" rating. In our judgment this is not prudent in view of the sensitivity of diesel generator performance to load increases above nameplate (continuous) rating. The applicant has been advised of our concern and has assured us that this matter will be given consideration during construction. We are satisfied with the applicant's response, and the fact that the design, at present, calls for no more than 1800 kW per diesel generator.

We conclude that the five-day supply of fuel oil, onsite, is adequate in view of the immediate availability of fuel oil supplies.

The diesel generators will be housed in separate rooms in a tornado-proof structure. The structure will be provided with internal walls to provide physical isolation. The remaining components of the emergency power system are either underground or housed in tornado-proof structures. These design precautions are acceptable.

The d.c. system consists of two batteries supplying two separate independent buses. The two buses are normally separated with a non-automatic tie breaker. Essential d.c. supply circuits are redundant with feeds from each bus and all d.c. circuits separately protected by circuit breakers at their respective d.c. bus. The batteries are located in separate rooms, and can supply essential loads for two hours without assistance from their respective battery chargers. We conclude the applicant's proposed design of the d.c. system has sufficient redundancy and independence and conforms to the requirements of Criterion No. 39.

3.6.3 Cable Routing and Loading

In response to staff questions (Ref. Supplement No. 7, 6(g) and (h)), the applicant has discussed his criteria relating to the internal routing of instrument and power cables, cable tray loading, and overcurrent protection. Instrument and power cables are to be separated by channels with either a minimum separation of 1 foot between cables, or protection within rigid steel conduit. Fire barriers will be placed beneath trays carrying protection circuits if such trays are located above trays carrying power cables. Connecting tubing between pressure sensing locations and transmitters will by physically protected and separated to prevent common failures resulting from missiles. The transmitters will be located in structural steel racks such that they are separated by a steel plate barrier.

All electrical overload protection, with the exception of that provided for some small motors, is furnished by 3-phase power circuit breakers.

Electrical loading and heat dissipation of cables on ladder trays throughout will be carefully studied and controlled to ensure no excess heating. IPCEA standards and manufacturers' recommendations will be followed.

Control room instrumentation is installed on a channelized basis with redundant instrumentation in separate cabinets or racks. Physical protection is afforded by space (aisles), metal barriers, or other equipment.

We have reviewed the applicant's criteria and conclude that, if followed, adequate precautions will have been taken against fire and/or other common failure modes.

3.7 Leakage and Fuel Failure Detection

Leakage from the primary system to the containment will be detected by the containment air particulate monitor. This system is quite sensitive to small leakage rates if background is low and there is sufficient activity in the coolant. It is largely ineffective during startup since there is little corrosion product activity in the primary water. As a backup, the containment radiogas monitor will also detect leakage, but with a lower sensitivity. It is also ineffective at startup when leakage detection is of great importance. The humidity detector in the containment also provides an overall method for measuring leakage from all water and steam systems inside containment. It is less sensitive than the air particulate monitor and also serves as a backup. However, since it does not depend on activity, its sensitivity is not reduced early in plant life. Proper use requires adjustment of the base line dewpoint temperature as cooling water temperature varies.

In addition, a system is provided which collects and measures the moisture condensed from the containment atmosphere by the cooling coils of the recirculation units. Since the cooling coils provide the only surfaces in the containment significantly below the dewpoint temperature, condensate flow should provide a good estimate of any leakage rate into the containment. However, since some primary system leakage passes directly to leakoff connections, e.g., pump seals, pressurizer relief valves, this system can measure only unanticipated leaks from lines or components.

Leakage from the primary system to the secondary system in the steam generators is determined measuring the sodium-24 activity of the steam generator blowdown water by gamma spectrometry and comparing the activity with that in the primary. The applicant has stated that, based on the experience at Indian Point Unit No. 1, it is anticipated that a primary-to-secondary leakage rate of 1 lb/hr (0.002 gpm) represents the lowest leakage rate measurable by this method.

Large leakage rates could be detected by an increase in makeup water flow rate required to maintain pressurizer level, and by an increase in the containment sump level.

The leakage detection methods proposed reflect the present state of the art. Details of the actual instrumentation to be employed for each method have not been reported by the applicant and the capability of each of the systems has not been established. Further, the applicant has not proposed any pre-operational tests which would determine these capabilities. Also, the proposed systems do not provide any capability for locating leaks.

We have concluded, therefore, that although a number of different techniques are used to detect leakage, each method has certain shortcomings, and the applicant should continue to consider alternate or additional designs or procedures prior to completion of construction. The proposed leakage detection systems will be analyzed and evaluated in depth at the operating license stage of our review.

As presently conceived, the plant will rely on the letdown monitor to detect fuel failure. However, a research and development program is presently underway at the Saxton reactor which is considering the following:

1. Delayed neutron monitor.

2. Coolant gamma activity monitor.

3. Gross gamma monitor along a main coolant line.

4. Letdown monitor.

The evaluation of the performance of these devices at Saxton will be available by late 1969. Thus, we conclude that a scheme which optimizes the current technology in reliability, sensitivity, and response time can be installed in Unit No. 3 prior to operation.

4.0 IMPORTANT SAFETY CONSIDERATIONS

4.1 Tornado Considerations

The control building, diesel generator building, primary auxiliary building, containment, and all connecting ducting for essential cabling and piping are designed to withstand tornado wind loadings corresponding to 300 mph tangential velocities, transverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function. The stress criterion used for tornado loading requires that there be no gross yield of the structure with the yield stress limits revised by the capacity reduction factors, Ø, of 0.95 for tension members, 0.90 for flexure, and 0.85 for diagonal tension, bond and anchorage. These structures are also designed to withstand the following tornado-generated missiles:

1. 4" x 12" x 12' plank at 300 mph
2. 4000 lb passenger car at 50 mph not exceeding 25 ft above the ground.

The foregoing criteria are consistent with similar criteria found acceptable for previously licensed plants. In addition to the foregoing criteria, the following general criteria have been adopted by the applicant relative to tornado considerations:

1. A tornado will not cause a loss-of-coolant accident.
2. A tornado will not impair the ability to safely shutdown the plant.
3. A tornado following a loss-of-coolant accident will not impair the long-term safety of the plant.

The tornado protection criteria outlined above are met by the protection provided by the facility structures with three exceptions. These exceptions, summarized below, rely on redundancy rather than structural protection:

1. Emergency feedwater for the steam generators is supplied from redundant water supplies. The feedwater pumps are housed in a protected structure. The normal source of feedwater is the secondary feed circuit which requires operation of the main condenser, air ejector, and service water system. For an alternate water supply, the feedwater pumps can take suction on the condensate storage tanks, the city water storage tanks, or can be connected directly to the city water supply. Thus, even if the normal feedwater train is disabled, three additonal feedwater sources are available to provide water to the steam generator feedwater pumps to permit dissipation of decay heat.
2. The makeup water for the primary system requires the availability of either the primary storage tank or the refueling water storage tank. In addition, limited makeup can be achieved using the volume control tank, boric acid tanks, and the monitor tanks.
3. Service water supply relies on the redundancy provided by the two supply lines, four screens and six pumps. Two pumps, one screen, and one supply line are required for prolonged shutdown.

The effect of tornadoes on the spent fuel pit is being evaluated by the applicant. He has stated the pit will be designed such that a cover can be added later if it cannot be demonstrated that a tornado has an insignificant effect on the fuel in the pit.

We have examined the structural design criteria and the general criteria for the plant proposed by the applicant and consider them to be acceptable. Since considerable time is available for action if one or more of the redundant components are lost, and in view of the physical location of redundant features, we conclude that the redundancy supplied in lieu of structural protection is acceptable.

4.2 Engineered Safety Features

4.2.1 Emergency Core Cooling System

4.2.1.1 ECCS Design

The ECCS for this plant closely resembles that of Indian Point No. 2. This ECCS consists of (1) one high pressure coolant injection and recirculation subsystem (HPS), (2) one low pressure coolant injection and recirculation subsystem (RHRS), (3) one low pressure coolant recirculation subsystem (LPS) located entirely within containment and (4) one accumulator subsystem. The inclusion of an internal recirculation system is the major difference between the ECCS of this Unit and those of recent four-loop PWRs.

The three pumps of the HPS are normally aligned to a common suction header which is fed by the refueling water storage tank. In addition, the suction of all high head pumps can be remotely realigned to the discharge of the low head subsystems. The three high-head pumps discharge to a header which feeds injection lines to the hot legs of reactor coolant loops 1 and 3, and injection lines to the cold legs of reactor coolant loops 2 and 4. Two high-head pumps have sufficient capacity to accommodate spillage from one of these four injection lines.

The two RHRS pumps take suction from the refueling water storage tank for short-term coolant injection and from the containment sump for long-term coolant recirculation. These pumps discharge through a common line to the two residual heat exchangers and then to the primary system by four cold leg injection lines.

The LPS contains two low-head recirculation pumps which are located within containment. They take suction from a recirculation sump and discharge to either of the two residual heat exchangers. One of four low pressure pumps (two each in the RHRS and the LPS) is capable of supplying the required post-accident recirculation flow to the core. The four accumulators discharge through the low pressure, cold leg injection lines, and the accumulators are sized on the basis that one of the four spills through a break.

We have done a failure mode analysis of the proposed ECCS and have concluded that it is designed to provide coolant injection at both high and low vessel pressure even if any single active component fails to operate. For breaks larger than about 6 inches in diameter, the accumulator subsystem is the only subsystem which can reflood the core in time to adequately limit clad temperature, oxidation, and deformation. This single subsystem is acceptable for coolant makeup because (1) it stores the energy required for operation, (2) it requires no external controls or signals for operation, and (3) it has sufficient capacity to accommodate anticipated spillage and core flow bypass. Our failure mode analysis has also shown that the two completely independent flow paths from the containment sumps, through the ECCS, to the core, provide the capability for long-term core cooling by recirculation of coolant to the top or to the bottom of the core even if any single component, active or passive, fails to operate.

For cold leg breaks in a PWR there is a possibility that some "steam binding" will occur due to the pressure drop for steam flowing from the core through the primary loop to the cold leg break. For the Westinghouse vessel and loop configuration, a sufficient head of water is developed in the downcomer to drive the steam generated in the core through the loop and out the cold leg break, thus allowing the water level to rise above the core.

The ECCS design makes it possible to establish a recirculation flow path through the core for either hot or cold leg breaks. Thus, boiling in the core can be terminated in about 4 days when the recirculating coolant has been sufficiently subcooled by the containment heat removal systems. The nonboiling mode of long term core cooling should reduce the rate of hydrogen production by radiolysis in the core.

Redundancy of the low-head pumps is sufficient to allow maintenance of one of the four pumps during normal reactor operation without requiring plant shutdown.

The performance capability of the accumulator subsystem was analyzed by the applicant assuming one accumulator spills through the break and either two or three accumulators deliver coolant to the vessel. For the case of two accumulators delivering, the peak clad temperature is predicted to be about 2600°F. For the case of three accumulators delivering, the peak clad temperature is predicted to be 2225°F. Nitrogen gas from the accumulators will pass through the core, temporarily disrupting core cooling. However, this will occur after the temperature rise has been terminated and will not significantly affect the core temperature transient.

All but one of the ECCS lines which penetrate containment are equipped with remote operating valves inside and outside containment which can be used to isolate an ECC subsystem. The exception is the sump suction line in the RHR subsystem. This line is equipped with an exterior (to containment) isolation valve and a concentric guard pipe which extends from the sump out to a leak-tight chamber which encloses the isolation valve.

It is planned to use the LPS for long term core cooling. This subsystem circulates coolant from the sump, through the residual heat exchangers, and back to the reactor without leaving the containment. In the event that recircualtion at high pressure is required, the RHRS and the HPS will be used and the coolant will leave containment.

All of the ECC subsystems can accomplish their functions when operating on emergency (onsite) power. If one of the three diesels fails to start, a minimum of two low-head and two high-head pumps would be available for operation. The diesel loads and the ECCS starting sequence are arranged so that the ECCS will be pumping at minimum acceptable capacity, assuming no further component failures following the diesel failure, within about 30 seconds following a LOCA.

The RHRS pumps, heat exchangers, and injection lines are shared with the shutdown cooling system. When the reactor is at power the ECCS function of these components is ensured by two closed isolation valves between the shut-down cooling system and the RHRS.

4.2.1.2 Performance

The applicant has presented performance analyses based on computer codes developed by Westinghouse. These codes are: FLASH-R which is used to calculate rate of coolant blowdown through a break, rate of coolant influx from the ECCS, core and loop pressure drop and flow, energy influx from the core, and energy efflux via the steam generators; CHIC-KIN which is the reactor kinetics code used to calculate the fuel energy input to the coolant during blowdown and to calculate the fuel energy input to the coolant during blowdown and to calculate the void shutdown for large breaks; LOCTA-R2 which is used to calculate the extent of clad-water reaction; and SLAP which is used for small break blowdown calculations.

In using these codes to determine the performance capability of the ECCS, the applicant has made conservative assumptions with regard to the more significant parameters, as follows: break opening time (all breaks assumed to occur instantaneously), reactor coolant pumps trip (loss of AC power coincident with shutdown), reactor shutdown (minimum void formation model for the void shut-down calculation), blowdown heat transfer (no credit taken for transition boiling and DNB assumed at 0.5 sec for all breaks), vessel water level (no credit taken for boiling froth height), accumulator spillage (one of four assumed to spill for all cold leg breaks), high head subsystem spillage (one of four injection lines assumed to spill through the break), and core heat transfer during reflooding (uniform coefficient of 25 Btu/hr-ft^2-°F). Based on our present understanding of the blowdown and core heatup phenomena, we conclude that the codes have been used conservatively to predict the course of the loss-of-coolant accidents. However, we cannot conclude that the models employed in these codes completely simulate the complicated blowdown heat transfer process or account for all of the blowdown mechanisms that might occur. AEC safety research programs in the areas of blowdown heat transfer and emergency core cooling (e.g., LOFT semiscale and FLECAT) should, in the next several years, provide adequate confirmation of the conservatism in the Westinghouse blowdown and core heatup models.

The applicant presented the results of blowdown and core heatup analyses for the double-ended, 6 ft^2, 3 ft^2, and 0.5 ft^2, breaks in the cold leg and in the hot leg of one of the reactor coolant loops. The cold leg breaks result in higher peak clad temperatures than hot leg breaks of corresponding size because of core flow reversals during blowdown, steam binding above the core during accumulator injection, and spillage of one accumulator; all of these effects were considered in the applicant's analyses. The performance of the minimum ECCS (i.e., 3 or 4 accumulators for a cold leg break or 4 of 4 for a hot leg break, 1 of 2 high head injection pumps, 1 of 2 low head injection pumps, and emergency power) is summarized in the following table.

Break Size, ft^2	Maximum Clad Temperature, °F	Total Percent Rod Perforations	Total Percent Clad-Water Reaction
8.2 cold leg (Double Ended)	2225	90	negligible
6.0 cold leg	1990	85	negligible
3.0 cold leg	1700	77	0
0.5 cold leg	2020	89	negligible
9.2 hot leg (Double Ended)	2110	-- (a)	negligible
6.0 hot leg	1840	-- (a)	negligible
3.0 hot leg	1510	-- (a)	0
0.5 hot leg	1350	-- (a)	0

(a) Not yet evaluated

The peak clad temperatures conservatively calculated for these breaks are well below the Zircaloy melting temperature. The peak temperatures for some breaks are above the Zircaloy-water reaction threshold (1800°F), but they are generally below the accelerated reaction temperature range (2200°F). The total clad-water reaction calculated for each of the breaks is much less than 1 percent of the total fuel clad mass. Furthermore, the clad temperature calculations reported by the applicant show that the clad hot spot is above 1800°F for only about 50 seconds and that only about 2.0 percent of the total clad volume exceeds a temperature of 1800°F for the double-ended cold leg break. On the basis of data from ANL which indicate that longer periods at higher temperatures are required to cause Zircaloy clad embrittlement by oxidation, our preliminary conclusion is that the clad heat transfer geometry will not be significantly altered by thermal shock upon quenching. We will continue our evaluation of the effect of quenching as further data become available.

The Westinghouse calculation predicts large numbers (~90%) of clad perforations for some of the intermediate size breaks; e.g., the 0.5 ft^2 cold leg break. Westinghouse has made conservative assumptions with regard to blowdown heat transfer and water level for the intermediate size breaks and we conclude that the resulting calculated temperatures are conservatively high. To demonstrate this the applicant has analyzed breaks of less than 0.5 ft^2 for the following two cases: (1) using their standard conservative assumption that DNB occurs at 0.5 seconds after all breaks, regardless of size, on all the rods in the core, and (2) using a three-dimensional thermal and hydraulic code (THINC) which calculates flow redistribution during blowdown and which predicts the time of DNB. The results of the second more realistic calculation show that the number of clad perforations decreases with break size to about 30 percent and 20 percent for the 0.5 ft^2 and the 0.3 ft^2 cold leg breaks, respectively.

Westinghouse is currently doing R&D work on rod perforations to improve their calculational model and to establish that the core heat transfer geomtry is maintained after a large number of perforations. We will continue to examine this area.

The applicant has also presented results of blowdown analyses for small breaks in the cold leg of a reactor coolant loop. In this analysis the flow out the break was defined by the Moody correlation for two phase critical discharge and the water levels are quiet levels: i.e., no credit is taken for the actual froth level that would occur due to void formation in the core. The following table summarizes the results of these analyses for the case of 1 of 2 high head pumps operating at 3/4 flow (1 of 4 injection legs assumed to spill).

Break Diameter, in.	Minimum Water Level, ft From Top of Core
1 (0.005 ft^2)	Slight decrease in normal level
2 (0.022 ft^2)	Above core
3 (0.049 ft^2)	-2.0
4 (0.087 ft^2)	-5.0
6 (0.196 ft^2)	-10.0

As indicated by this table, the core hot spot, which is located at about the axial midplane of the 12 ft core, will remain covered for break sizes up to 4 inches in diameter.

The applicant has also presented core heatup analyses using the THINC code for the 6-inch diameter break. The code predicts nucleate boiling in the core throughout the blowdown transient for the 6-inch break and the resulting peak clad temperature is about 725°F. The clad heatup process for a break of less than 4 inches in diameter, i.e., a break which does not uncover the core hot spot, is described as follows. The reactor will scram on low pressurizer level or pressure and the core heat will decrease to less than 4 percent of full power by the time the top of the core uncovers. Core coolant flow will decrease during the blowdown as the reactor coolant pumps coast down following the asusmed loss of offsite power; natural circulation of coolant will follow pump coastdown. Boiling heat transfer will occur in the core throughout the blowdown because the heat flux is low (20,000 Btu/hr-ft^2-°F) at the time of minimum water level (100 seconds) and DNB is not predicted to occur. Even if DNB were to occur, it is likely that the transient film boiling which should ensue will provide sufficient cooling to prevent clad burnout. It appears, therefore, that the ECCS should be able to prevent all clad damage for breaks of less than 4 inches in diameter.

For those breaks between 6 inches and 0.5 ft^2 (9.5 inches) the hot spot is uncovered for only a brief period and the clad temperature transients should be less severe than those calculated for the intermediate size breaks where the core uncovers more rapidly. The hot spot will be cooled by the two-phase mixture of coolant rising from the quiet water levels as steam is formed by depressurization and boiloff. The accumulators will refill the vessel for these breaks and terminate the core temperature transient.

We have done calculations to verify that breaks with an equivalent diameter of less than 3/4 inch will cause loss of coolant at a rate which can be accommodated by the reactor charging pumps (no ECCS action required). These pumps will maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown. Since instrument taps and sample connections are of less than 3/4-inch diameter, protection of the core following the rupture of these lines is afforded by the charging pumps.

4.2.1.3 Conclusions

We conclude that the design of the proposed ECCS (1) limits the peak clad temperature to well below the clad melting temperatures, (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.

However, further research and development concerning clad perforations and their effect on core cooling is needed as well as experimental confirmation of ECCS design conservatism. These items are research and development programs and are discussed in Section 6.0 of this report.

4.2.2 Post Loss-of-Coolant Accident Protection (PLOCAP)

Provisions will be made in the design and layout of Unit No. 3 to enable the installation of additional equipment to mitigate the consequences of a post-lost-of-coolant-accident reactor vessel failure, if further analysis of the thermal shock experienced by the vessel during safety injection indicates that such protection should be required.

The proposed PLOCAP system would direct the low head injection flow and the subsequent recirculation flow to the hot legs of the coolant loops to provide top injection for the core. In addition, a fast-acting cavity flooding system would be provided. Valves, which open upon receipt of signals from both safety injection initiation and accumulator low pressure, would permit the cavity flood tanks to drain to the cavity, raising the level of water to just below the bottom of the reactor vessel. This level will be specified to prevent damage to the pressure vessel in the event of inadvertent opening of the cavity flood tank valves.

The combination of the safety injection and low accumulator pressure signals would also open valves in the discharge of the recirculation pumps to permit the cavity filling to be completed. These pumps take suction on the sump and, therefore, cannot increase the level in the cavity unless a loss-of-coolant has occured.

We have considered the possibility that operation of this system might aggravate the thermal shock problem for small breaks. Signals indicating low accumulator pressure as well as safety injection system initiation must be received before the cavity flood tank valves are opened. Thus, the system cannot be activated unless primary system pressure is below 600 psi. In addition, the primary system pressure is rapidly decreasing with time when cavity flooding is started. For illustration, a tabulation of the time increment for the pressure to decrease from 600 psi to 100 psi for various break sizes is presented below. For the rupture of the 1- to 6-inch lines, it is assumed that two of the three high-head safety injection pumps deliver through three lines. The coolant in the fourth line is assumed to spill to simulate a break near the injection location.

Break Size	Time to Depressurize from 600 to 100 psi (sec.)
1 inch line	335
2 inch line	360
3 inch line	335
4 inch line	260
6 inch line (0.196 ft^2)	115
0.5 ft^2 (cold leg)	12
0.5 ft^2 (hot leg)	15
3 ft^2 (cold leg)	5
3 ft^2 (hot leg)	8
6 ft^2 (cold leg)	4.5
6 ft^2 (hot leg)	4.5
8.2 ft^2 (cold leg)	3.5
9.2 ft^2 (hot leg)	3.5

Even for the small breaks, system pressure drops to below 100 psi within six minutes of the time the cavity flood system is initiated.

To permit installation of the PLOCAP system at a later date, the following provisions will be incorporated into the design:

1. A standpipe will be installed over the incore instrumentation passageway to permit the retention of water in the cavity to the level of the core without flooding the floor of the containment.

2. Nozzles will be installed on each hot leg pipe to permit installation of a hot leg injection system.

3. A second containment sump line will be installed which will enable achievement of the high recirculation flow rates required to rapidly raise the cavity liquid level above that resulting from drainage of the cavity flood tanks.
4. Space will be reserved in the primary auxiliary building for increased heat exchange and pumping capability.
5. Provisions will be made to ensure that the cavity flood tanks and associated piping can be installed.
6. Detailed pipe layouts and plant arrangements will be developed considering the extra pipework and containment penetrations required by PLOCAP.

We have evaluated the provisions for the proposed system and have concluded that these provisions are adequate to give reasonable assurance that a reliable system can be installed to mitigate the consequences of a reactor vessel failure following a loss-of-coolant accident, should subsequent evaluation show that such a system would be required. The foregoing is substantially similar to the PLOCAP design described to the Committee in connection with the Zion review.

4.2.3 Iodine Removal

4.2.3.1 Spray Removal

We have calculated the iodine reduction factor for the chemical additive spray system using a modified form of the equation developed by Griffiths to determine the theoretical efficiency of the system. This differs from the model used by the applicant in that factors of conservatism are included to account for possible liquid film mass transfer resistance and drop coalescence. The value of the calculated removal constant for elemental iodine is 4.9 hr^{-1}.

A value of 10% of the iodine in the containment atmosphere (2-1/2% of the core inventory based on TID-14844) has been adopted by DRL as reasonable upper limit for the fraction in the form of organic iodides. This is based on an extensive literature examination of available data and on a theoretical evaluation of all applicable formation mechanisms. Since experiments have shown that the removal of organic iodides by an alkaline spray solution is negligible, no reduction of the organic iodides is assumed in this analysis. On this basis the two-hour overall iodine dose is reduced by 5.2 and the thirty-day overall iodine dose is reduced by an upper limit of 10.

4.2.3.2 Removal of Organic Iodides by Impregnated Charcoal Filters

The applicant has proposed a filter system similar to but smaller than that proposed for Unit No. 2. In this system, a portion of the discharge from each of the fan-cooler units is passed through activated charcoal impregnated with 2-5% inorganic iodide. The air entering these filters has a relative humidity of approximately 100%.

Charcoal in this form has demonstrated the ability to remove organic iodides from a moving air stream. At low relative humidities (less than 70%) the removal efficiencies are measured to be in excess of 95%. Above 90-95% relative humidity there is considerable uncertainty on the degree of removal effectiveness for organic iodides.

Research performed at ORNL in a small-scale apparatus, using impregnated charcoals of various manufacture, indicates that at 90% relative humidity the removal efficiency for organic iodine has decreased to about 90%. When the relative humidity is above 90% , there is a rapid decrease in removal efficiency. The published curves show essentially zero effectiveness at 100% relative humidity. The principal investigator (R. Adams) has stated that the charcoal probably was partly "waterlogged" in at least some instances under these severe conditions. However, many experiments at high relative humidities have yielded low organic iodide retention and yet have shown no indication of waterlogging.

Westinghouse has performed experiments with a full-scale prototype charcoal filter unit in a loop (The Connecticut Yankee Tests). Temperatures were of the order of 270°F (maximum expected post-accident conditions) and a steam-air environment was maintained. Relative humidities were measured by a wet bulb-dry bulb arrangement, and relative humidities of 100% were claimed. In most cases, organic iodide removal capabilities in excess of 70% were reported. On the basis of these tests, the Westinghouse position is that the impregnated charcoal filters will remove organic iodides even at high relative humidity.

At present there is conflicting evidence regarding the capability of impregnated charcoals to effectively remove organic iodides from a moving air stream held at a relative humidity near 100%. This uncertainty extends both to the capability of the charcoal and to the question of whether it is possible to prevent intergranular condensation of water under these conditions. Since doubt exists as to the performance characteristics of impregnated charcoals under these severe conditions, we have held further meetings with Con Ed and Westinghouse and they have agreed either (1) to conduct further R&D in the area of organic iodide removal by impregnated charcoal at high relative humidity, and to reserve space in the containment for dehumidification equipment to be installed if the R&D program indicates it is necessary; or (2) to provide a design for a system to dehumidify the air entering the charcoal filters. The applicant will indicate his decision in this regard to us just prior to the Subcommittee meeting and we will report orally at that time.

The details of this proposal will be formally submitted to us prior to the January meeting. Provided that this submittal includes an adequate description of the scope and nature of the proposed research and development program or an adequate preliminary design of dehumidification equipment, we conclude that the applicant can provide a filter system with a removal efficiency for organic iodides of greater than 5%. As indicated in Section 5.5 of this report, a filter system with this efficiency will reduce the offsite doses to below the 10 CFR 100 guidelines.

4.2.4 Hydrogen Production and Recombination

Hydrogen production following a loss-of-coolant accident has been estimated by Westinghouse from the following sources:

1. Hydrogen from 2% metal-water reaction.
2. Corrosion of exposed aluminum surfaces at 1,000 mg/dm^2/day.
3. Radiolysis of water in the core by absorbed gamma radiation at 0.44 molecules/100 ev.
4. Radiolysis of water not in the core by beta and gamma radiation from an assumed 50% of the core halogens entrained in the water at 0.30 molecules/100 ev.

These sources and assumptions are the same as those used recently in our evaluation of the D. C. Cook plant. Based on the limited information available at this time, these appear to be realistic estimates. However, as discussed in Section 6.0, research and development effort is being directed to eliminating uncertainties in the prediction of hydrogen production. The results of these programs will be reflected in the final design. Further, although the radiolytic decomposition of water is a reversible reaction, no credit is given for reduction of the hydrogen production rate by the back reaction. On this basis, we conclude that the parameters chosen are reasonable bases for the preliminary design of the proposed recombiner system. Using these assumptions, the following table of hydrogen concentration vs. time is obtained:

Days Post Accident	Rate of hydrogen formation (scfm)	Hydrogen in Steam Free Air (v/o)
2.7	2.9	1.0
9.8	1.74	2.0
51	1.1	4.1
$\geq$100	--	10.0

(As a comparison, use of the same assumptions on the D. C. Cook plant which utilizes the ice-condenser concept with smaller containment volume yields a 2 v/o hydrogen concentration in 2.6 days.)

Our position with respect to the production of hydrogen by radiolysis has not changed from that discussed with the ACRS during review of the D. C. Cook plant but is restated for the Committee's convenience. The applicant has proposed research and development programs designed to determine the rate of radiolytic decomposition of the spray and core cooling water within the containment. Parameters which will be examined include flow, temperature, and chemical factors. In addition, we believe that equilbrium hydrogen concentrations (taking into account the reverse reaction) to be expected in the accident environment should be determined.

We will work with the applicant to assure ourselves that his research and development program meets our requirements and specifically includes the following areas:

1. The effects of flow, boiling, temperature, and chemical composition on the radiolysis rate and on equilibrium hydrogen concentrations.
2. The amount of gamma and beta energy absorbed in the water in both core area and in the sump and its effect on the radiolysis rate. Radiolysis from both an external radiation source (simulating the core) and a radiation source mixed in the water (simulating fission products in the containment sump) must be explored.
3. The effect of air-to-water volume ratios and the influence of surface areas at the liquid-gas interface on the radiolysis rate and equilibrium concentrations.

To eliminate the potential for rapid hydrogen oxidation, the applicant has proposed the use of a flame combustor using the containment atmosphere as a primary oxidant and supplemental hydrogen as fuel. Two flame combustors will be located inside containment, one serving as a spare. Each consists of a blower to circulate containment air to the combustion chamber, the combustion chamber, two ignitors (one required) consisting of a capacitance system with surface gap plugs designed to operate in a wet environment, and a dilution chamber downstream to reduce exit temperature to below 300°F. Hydrogen is supplied to the combustor from tanks outside containment through two normally closed valves located outside containment, and a check valve located inside containment. Each combustor contains two thermocouples. To ensure presence of an oxidant, oxygen is bled into the containment through a separate penetration. This inlet line is located to ensure mixing by the containment ventilation system before introduction to the combustor. Oxygen flow is proportional to hydrogen flow to maintain stoichiometry.

The hydrogen supply lines will be purged with nitrogen before introducing hydrogen. A block and bleed system is provided to prevent either hydrogen or oxygen inleakage when the system is not in use. Further, we have been orally informed that large quantities of hydrogen will not normally be located at the site. Hydrogen for the recombiner will be brought to the site following an accident. In view of the time available, we believe this provision is adequate. Alarms are provided to alert the operator to low combustor temperature, and to low manifold pressure for both the hydrogen and the oxygen.

The combustor is designed to process 331 scfm. It will normally be started when the containment reaches a hydrogen concentration of 2 $^{v}/o$. Thus, the combustor will remove 6.6 scfm of hydrogen from containment. Hydrogen generation at 9.8 days (the time at which a concentration of 2 $^{v}/o$ is predicted) is 1.74 scfm. Thus ample margin is provided.

To estimate the ability of the recombiner to accomodate additional metal-water reaction, we have used the curve presented on p. 2(2-4)-3 of the First Supplement to estimate the hydrogen generation rate as a function of time. Graphical differentiation of this curve gives radiolytic production rates of 13.3 scfm and 10.8 scfm at 10 and 20 hours after the accident respectively. Aluminum corrosion is stated to generate 1000 scf over the first 7 days, a rate of 0.1 scfm. Thus, we estimate the hydrogen generation rate to be 13.4 and 10.9 scfm at 10 and 20 hours, respectively. The recombiner is designed to process 331 scfm. If operating at a 4$^{v}/o$ hydrogen concentration, the system would remove 13.2 scfm hydrogen. Linear interpolation of the rates presented above indicates this rate could be met 10.8 hours after the accident. Considering the hydrogen produced by both radiolysis and corrosion in this period, it is calculated that a 25% metal-water reaction could occur without reaching a combustible concentration averaged over the entire containment.

A testing program will be established which will generate the following information relative to design and performance of the recombiner.

1. Performance of the combustor at light-off and under operation with the fuel supply rate varied to provide combustion zone outlet temperatures in the range from 300°F to 1800°F.
2. The lower limit of oxygen concentration for flame stability.
3. Efficiency of combustion by operating at design conditions and determining outlet hydrogen concentration.
4. The stability range of the burner by varying air and fuel flow.
5. The effect of steam and entrained water on burner lightoff and operation.

We have reviewed the recombiner design and test program as described above. On the basis of our review and of discussions with experts in this field, we conclude that the flame recombiner may be a feasible solution to the hydrogen problem; however, many aspects of the design must be examined more closely before we can conclude without reservation that the design is acceptable. For example, it will be necessary to determine the performance limits of the recombiner including limits on pressure, moisture and hydrogen concentration to demonstrate substantial margin with respect to variation in the expected post-accident conditions; means must be provided to preclude inadvertent introduction of hydrogen or oxygen into the containment at any time; definitive criteria for recombiner design and operation must be agreed upon; and the applicant should not only explore the capabilities of the proposed flame recombiner, but also investigate alternate means of recombining the hydrogen, including catalytic recombiners, cryogenic separation, chemical absorption, and processing of the containment gases external to the containment structure.

The capability to start the recombiner at the proper time is dependent upon the testing technique used to sample containment atmosphere. The applicant proposed to batch-sample the atmosphere and analyze it by gas chromatographic means. A sensistivity of 0.02 $^{v}/_{o}$ and a reproducibility of $\pm$ 5% of the measured value is claimed. In our judgment, these figures appear to be reasonable; however, the details of the means of handling batches of radioactive gas from the containment have not yet been developed. This area will also require further careful review.

On the basis of our review of the preliminary recombiner design and the research and development and testing programs proposed by the applicant in this area, we conclude that there is reasonable assurance that the safety problems associated with the radiolytic production and recombination of hydrogen will be resolved prior to the operation of Indian Point Nuclear Generating Unit No. 3. We will continue to study radiolysis as a general problem and will continue to review the requirements for a recombiner on this plant as a followup item during construction. In evaluating the radiolysis problem and any proposed recombination or cleanup devices, we are seeking assistance from recognized experts in the appropriate fields.

4.3 Stress Analyses

4.3.1 Reactor Internals

The reactor internals will be designed to meet stress limit criteria as established in Section III of the ASME Boiler and Pressure Vessel Code for the normal design loads of mechanical, hydraulic, and thermal plus the operational basis earthquake load.

The reactor internals will also be designed to withstand the concurrent blowdown and design basis earthquake loads, as indicated by the applicant in Supplement 1, Section 15. Primary tensile stresses under such load combination will not exceed stresses corresponding to 20% of the uniform strain at temperature, while the allowable deflection limits will be about 50% of the loss-of-function deflections for the specific components. We consider these stress and deformation limits to provide adequate margins of safety, since they are basically the same as the criteria recently accepted for other PWR plants.

4.3.2 Reactor Coolant System

Section III of the ASME Pressure Vessel Code will be used to design the reactor vessel, pressurizer, coolant pump casings, and the steam generator. To provide access for inspection, the vessel and its internals will be constructed so as to permit removal of the internals during plant life. The reactor coolant piping design will be analyzed in accordance with the requirements of USA S.I. B31.1 Code for Pressure Piping. A complete stress analysis which reflects consideration of all design loadings detailed in the design specification will be prepared by the manufacturer to assure compliance with the stress limits of Section III for the reactor vessel, steam generator, pressurizor, and pump casing. Westinghouse will independently review these stress analyses. A similar analysis of the piping will be prepared by or for Westinghouse by a qualified piping analysis contractor.

The reactor coolant system, and all other Class I (seismic) mechanical systems, will be designed to withstand normal design loads of mechanical, hydraulic, and thermal origin plus operational basis earthquake loads within normal code allowable stresses. In addition, as stated in Amendment 1, Class I systems and components will be designed to withstand the concurrent blowdown and design basis earthquake loads. Primary membrane stresses under such load combinations will not exceed stresses corresponding to 20% of the uniform stress at temperature.

We conclude, on the basis of our evaluation, that the design criteria proposed for the reactor coolant systems provide adequate margins of safety.

4.3.3 Reactor Vessel Thermal Shock

Our general review of the thermal shock problem is continuing. We are still uncertain that presently available experimental data, and the analytical techniques of elastic fracture mechanics, can clearly demonstrate that the reactor vessel will maintain its integrity under the thermal shock conditions experienced as the relatively cold emergency core cooling system water is added to the vessel following a loss-of-coolant accident.

The uncertainties in the analysis of the thermal shock effects on the reactor vessel are of three origins:

a. The heat transfer calculations leading to temperature and stress distributions through the vessel wall, as a function of time,

b. The experimental data on fracture toughness, and

c. The analytical techniques of elastic fracture mechanics.

There is general agreement among the reactor manufacturers, based on results of fracture mechanics analyses, that an initial small crack, which could be present at the vessel beltline, would propagate under the thermal shock stress conditions. As of October 1968, the extent of crack propagation, assuming an initial circumferential crack and cooling water temperature of about 70°F, has been calculated as follows:

B&W	55% penetration at	600 sec
CE	40% penetration at	1000 sec
W	60-80% penetration at	1000 sec

We have recently received an additional written submittal from Westinghouse based on new fracture toughness data. The conclusion of this report is summarized in the Fifth Supplement to the PSAR and states that considering the conservative lower fracture toughness band, any crack propagation is expected to be less than 32 percent penetration and, therefore, the integrity of the reactor vessel will be maintained throughout the life of the plant. We are presently reviewing that report and anticipate that our review will be completed and the acceptability of the predicted penetration determined during the first quarter of 1969.

At the present time, we conclude that there should be no danger of vessel failure until several years of vessel irradiation. The Heavy Section Steel Technology Program at Oak Ridge National Laboratory, due for completion by 1973, will provide additional data on material properties. Westinghouse is also participating in Euratom-funded fracture mechanics program to obtain irradiated fracture toughness properties. Furthermore, even if it should be shown that the vessel might crack, there appear to be suitable engineering solutions that could be employed if needed.

5.0 ACCIDENT ANALYSES

5.1 Steam Line Rupture

5.1.1 Rupture Inside Containment

The detailed analyses of the transients resulting from a rupture of a main steam line (4.6 ft^2 break area) have been conservatively estimated for Unit No. 3. These analyses indicate that for a steam line rupture at the steam generator discharge with failure of a single RCC element to scram, the rapid cooldown of the moderator would cause the core to return to critical in 25 seconds after the rupture and result in a maximum heat flux of 40% of the full power average heat flux. Injection of borated water would cause the core to go subcritical at 87 seconds. The applicant estimates, and we agree, that 10% fuel failures could result from this transient.

The offsite consequences of this accident which were calculated using assumptions tabulated in Appendix 1 are given in Table 5.1.

5.1.2 Rupture Outside Containment

The consequences of a steam line break outside containment have also been analyzed. The resulting transients are less severe than those for the rupture inside containment since the break occurs downstream of the flow nozzles. This limits the break area to 1.5 ft^2 and results in a slower depressurization of the secondary and less temperature decrease in the primary. For this case return to criticality occurs 42 seconds after the rupture. The core is made subcritical by the addition of borated water at 107 seconds. In this case the maximum heat flux would be 25% of the full power average heat flux. The applicant has stated, and we agree, that no DNB or resulting fuel failure will occur.

The offsite consequences calculated using the assumptions tabulated in Appendix 1 are given in Table 5.1.

5.2 Steam Generator Tube Rupture

The double-ended rupture of a steam generator tube is well within the capability of the core cooling system. Thus, core damage is not assumed to occur. If offsite power is not lost, the air ejector effluent is diverted to the containment upon receipt of a high radiation signal. Thus, the release of radioactivity to the environs occurs only during the one minute span between the time of the tube rupture and the time diversion is completed. Further, since the release of activity is through the condenser, a large water to-air iodine partition factor is available. The applicant has determined the whole-body dose at the site boundary to be 18 millirem for this case.

If the air ejector effluent is not diverted to the containment, the release of activity to the atmosphere continues until the steam generator can be isolated in 30 minutes, i.e., when the primary system pressure drops below the secondary system safety valve setting. For this case, the applicant calculated the whole body dose at the site boundary to be 0.5 rem.

To indicate the upper limit of consequences we have calculated the offsite doses assuming (1) the tube rupture occurs concurrent with a loss of offsite power resulting in loss of condenser flow, thus preventing diversion of the air ejector effluent to the containment, and (2) the operator does not isolate the affected steam generator. For these reasons, the doses are relatively high. Other assumptions are tabulated in Appendix 1. The offsite doses are given in Table 5.1.

5.3 Refueling Accident

A refueling accident can result if a fuel assembly is dropped or otherwise damaged during transit from the reactor vessel to the spent fuel pit. The applicant has stated that the maximum damage which can occur would result in release of gap activity from one row of fuel rods in a fuel assembly. We are continuing to evaluate the consequences of this accident as it relates to similar accident evaluations of General Electric, Combustion Engineering, and Babcock and Wilcox reactors.

We have calculated the radiological consequences of this accident given in Table 5.1 using the assumptions tabulated in Appendix 1.

5.4 Rod Ejection Accident

The detailed rod ejection physics calculations have not been completed by the applicant. The applicant states that the reactivity worth of a single inserted rod will be less than 0.5% k. The applicant has stated that any rod which drops into the core at full power will be promptly noticed. As noted in Section 3.1, we conclude that information from in-core instrumentation must be provided to an operator to enable him to detect anomalous power patterns in the core. On this basis, we have not considered the increase in rod worth resulting from xenon decay in the poisoned region. The detailed analysis of this transient at the operating license stage of our review will consider this effect if it is not evident that a dropped rod can be detected.

We recognize that the 1% failed fuel elements which might be present in the core might affect the ability to cool the core following a rod ejection accident. We are continuing to review this problem and will require a detailed analysis of the consequences of clad rupture in the context of other regulatory reviews.

Because we assume loss of offsite power concurrent with this accident, and because we assume primary-to-secondary leakage, radioactivity in the primary system would leak to the secondary system and be released via the secondary relief valves until the primary system depressurizes sufficiently to allow the residual heat removal system to function in a manner identical to that assumed by the staff in our evaluation of a steam line rupture inside containment. Since the fuel failures associated with these accidents are the same, the model used in assessing the radiological consequences is identical to that used in determining the consequences of a steam line rupture inside containment as stated in Appendix 1. The resulting calculated doses are identical.

5.5 Design Basis Accident (DBA)

The ability of the emergency core cooling system to cope with a major loss-of-coolant accident is presented in Section 4.2.1. We have calculated the consequences of the DBA assuming a TID-14844 fission product release and considering the effect of the spray system in reducing the iodine source in the containment. The assumptions are tabulated in Appendix 1. The consequences are given in Table 5.1 for three cases: (1) No iodine removal, (This is presented to indicate the total dose reduction achieved by the dose limiting safety features), (2) Iodine removal by sprays only, and (3) Iodine removal by sprays and charcoal ($\geq$5% removal efficiency for organic iodides). The offsite dose at the outer boundary of the low population zone is seen to be within the 10 CFR 100 guidelines if the removal efficiency of the filters for organic iodides is at least 5%. As discussed in Section 4.2.3 of this report, we conclude that such efficiencies can be achieved.

5.6 Summary of Radiological Consequences

The following is a summary of the staff's estimate of the doses resulting from the accidents analyzed:

Accident	Two-Hour Dose at Site Boundary		Course of Accident Dose at LPZ Outer Boundary	
	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)
Steam Line Rupture Inside Containment	18	200	12	200
Steam Line Rupture Outside Containment	0.28	140	0.23	62
Steam Generator Tube Rupture	12	73	5	31
Refueling Accident	0.26	105	0.11	44
Rod Ejection Accident	18	200	12	200
DBA without iodine removal	9.4	1410	13	3350
DBA with iodine removal by sprays	5.8	272	7.6	383
DBA with iodine removal by sprays and charcoal	< 5.8	< 272	< 7.6	< 300

6.0 RESEARCH AND DEVELOPMENT

This section will be submitted as a supplemental report.

As added background regarding the Westinghouse sponsored R&D, we have sent to the Committee, a report titled "Westinghouse Electric Corporation Development Programs." This report was submitted to the Atomic Safety and Licensing Board in support of the Public Service Electric and Gas Company's Salem application.

7.0 TECHNICAL QUALIFICATIONS AND CONDUCT OF OPERATIONS

7.1 Technical Qualifications

We have reviewed the application with respect to the adequacy of the technical qualifications of Con Ed and its contractors. The execution of the Indian Point Nuclear Generating Unit No. 3 project is the sole responsibility of the Consolidated Edison Company of New York, Inc. They have previous nuclear experience through their operation of Indian Point Unit No. 1 and have a twenty-man Nuclear Division associated with their Mechanical Engineering Department.

Con Ed has engaged Westinghouse Electric Corporation as the prime contractor under a turnkey contract. Westinghouse has engaged United Engineers and Constructors to serve as the architect engineer. These contractors as well as Con Ed, are recognized to be competent in their areas of specialization. On the basis of our previous and current evaluations of plants designed and constructed by the contractors and the applicant's experience in the operation of Unit No. 1, Consolidated Edison Company of New York, Inc. and its contractors are technically qualified to design and build Indian Point Unit No. 3.

7.2 Conduct of Operations

The applicant proposes a Unit No. 3 station staff of approximately 42 people. Shift operations will be directed by the Unit Watch Foreman reporting directly to the Station Shift Supervisor. The latter will report to the Station Generation Superintendent. Overall station responsibility resides with the Station General Superintendent.

The applicant proposed a 4-man shift crew. We have noted in our review of the Russellville and D. C. Cook applications that we consider 4-man operating shift to be marginal. We continue to have this concern in this case and believe that a minimum Unit No. 3 shift operating crew of five men is required, at least during initial station power operation. We will review the operating philosophy and justification of nuclear station staffing in sufficient detail and in time to assure an adequate operating shift crew size prior to initial plant startup.

Both staff and operating personnel will be drawn largely from the ranks of the Unit No. 1 and 2 organization. Specific training for Unit No. 3 will be provided for the AEC licensed operator candidates approximately one year prior to fuel loading and will include both Westinghouse-administered training and onsite Con Ed training. Although the Unit No. 3 staff will be developed from within the existing organization, we have received assurance from the applicant that competency of the Indian Point Station staff will not be diluted or downgraded.

On the basis of our review of the information presented, we conclude that the applicant has a training program which is satisfactory and will provide a qualified group in terms of competence and generally an adequate number of people.

8.0 QUALITY ASSURANCE AND QUALITY CONTROL

This section will be submitted as a supplemental report.

9.0 CONCLUSION

We have identified and reviewed the safety issues associated with the proposed design and forthcoming operation of Indian Point Nuclear Generating Unit No. 3. With the exception of the emergency power system where we do not consider there to be adequate independence of the diesel generators, and the information yet to be submitted on organic iodide removal, we believe there are no unresolved safety considerations that are not general to all pressurized water reactors. We will further review and resolve the question of diesel independence prior to installation. We anticipate that the organic iodide removal question will be resolved prior to the January ACRS meeting. Accordingly, we believe the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

APPENDIX 1

ASSUMPTIONS USED BY THE STAFF IN THE ACCIDENT ANALYSES

1. Steam Line Rupture Inside Containment

(1) Prior to the accident, the plant is operating with 1% failed fuel and 10 gpm primary-to-secondary leakage. The applicant has indicated these values will be proposed as technical specifications.

(2) 10% fuel failures result from the transient. Equilibrium secondary activity calculated does not consider normal steam generator blowdown and assumes all iodines which leak prior to the accident are retained in the steam generator until they decay or the accident occurs.

(3) Leakage from the primary system to the secondary system continues in the intact steam generators following the accident. It is assumed that secondary pressure drops instantaneously to atmospheric pressure while the primary system pressure decreases linearly to 350 psia in 8 hours, corresponding to a cooldown rate of 50°F/hr. Primary-to-secondary leakage is assumed to vary as the square root of the P. At 350 psia, the residual heat removal equipment can be operated and the steam generator isolated, thus eliminating leakage.

(4) Loss of offsite power requires heat rejection by boiloff to atmosphere in the intact steam generators for 8 hours.

(5) Boiloff in the steam generators results in release of equilibrium secondary activity and the activity in the primary system leakage to atmosphere with an iodine water-to-air partition factor of 50.

(6) Standard ground release meteorology and dose conversion factors.

2. Steam Line Break Outside Containment

(1) Prior to the accident the plant is operating with 1% failed fuel and a 10 gpm primary-to-secondary leakage.

(2) Secondary equilibrium activity calculated assuming all iodine remains in the liquid phase with no credit given for normal steam generator blowdown.

(3) Integrated leakage from primary-to-secondary following the accident calculating with the same assumptions as that assumed in 1(3) above.

(4) Equilibrium activity in the affected steam generator and all activity in the primary system leakage released to the atmosphere without partition since all liquid in the steam generator is assumed to flash.

(5) Standard ground release meteorology and dose conversion factors.

3. Steam Generator Tube Rupture

(1) Prior to the accident the plant is operating with 1% failed fuel and 10 gpm primary-to-secondary leakage.

(2) Secondary system equilibrium activity determined as in 1(2) above.

(3) The affected steam generator is not isolated, resulting in blowdown of 50% of the primary system volume to the secondary system.

(4) Loss of offsite power occurs requiring operation of the steam line relief valves and reliance upon steam generator boiloff to dissipate decay heat.

(5) Resulting flashing to atmosphere releases all noble gases in the primary-to-secondary blowdown to atmosphere. Equilibrium iodine in the secondary system and in blowdown from the primary system released with a water to steam partition factor of 10.

(6) Standard ground release meteorology and dose conversion factors.

4. Refueling Accident

(1) Perforation of 15 fuel rods (one row of rods in an assembly).

(2) Gap activity in the rods is released. This is assumed to be 20% of the noble gases and 10% of the iodine in the rods.

(3) The accident occurs 100 hours after shutdown. This represents a reasonable estimate of the time required to cooldown, remove the pressure vessel head and the upper internal package, and begin the refueling operation.

(4) 90% of the released iodine is retained in the water of the spent fuel pit or canal.

(5) Standard ground release meteorology and dose conversion factors.

(6) No credit given for spent fuel building confinement.

5. Design Basis Accident

(1) TID-14844 releases (100% noble gases, 25% iodines, and 1% solids).

(2) Design containment leakage rate, 0.1% per day, for first day, and 0.045% per day thereafter.

(3) Spray removal constant for non-organic iodines of 4.0 hours^{-1}.

(4) 10% organic iodide fraction.

(5) Standard ground release meteorology and standard dose conversion factors.