SUPPLEMENTAL
FILE COPY

FEB 19 1969

Docket No. 50-286

Consolidated Edison Company of New York, Inc.
4 Irving Place
New York, New York 10003

Attention: Mr. Donham Crawford
Administrative Vice President

Gentlemen:

We have completed our review of the information in your application for a construction permit for Indian Point Nuclear Generating Unit No. 3. The material that you have submitted does not fulfill all of our requirements for the contents of applications, as specified in 10 CFR 50 and elsewhere. Accordingly, we will need to defer continuing our review until your application has been completed.

In summary, the proposed 10 CFR 50 requires coverage as fully as available information permits of all required information. This includes a description and safety assessment of the proposed site, the preliminary design of your proposed facility, including the principal design criteria, the design bases and the relation of the design bases to the principal design criteria, information relative to materials of construction, general arrangement and approximate dimensions sufficient to provide reasonable assurance that the final design of your proposed facility will conform to the design bases with adequate margins for safety, and a preliminary safety analysis and evaluation of your proposed facility.

In addition to 10 CFR 50, our Guide for the Organization and Contents of Safety Analysis Reports summarizes the type of information that should be included in Preliminary Safety Analysis Reports. The guide provides a detailed outline of information to be provided and states that information submitted should show how the principal design criteria are met by:

1. Identifying the design <u>bases</u> and explaining the reasons therefor.

2. Describing the system or component to show <u>how</u> the design <u>bases</u> have been satisfied.

3. Showing through evaluations that design bases have been met with a reasonable margin for contingencies.

4. Providing a basis for such limits upon operation that might be appropriate in the interest of safety.

Other guidance is given in this document with respect to tests, inspection, and surveillance.

Our review has identified a number of areas in which your application is not complete. For example, amendments to your Preliminary Safety Analysis Report (PSAR) should include a discussion of the design of Indian Point Nuclear Generating Unit No. 3 with respect to the 70 General Design Criteria specified in 10 CFR 50.

As stated in your application, the design of Indian Point Unit No. 3 is essentially the same as Indian Point Unit No. 2. An adequate response to some of the information requested concerning Indian Point Unit No. 2 which reflects updated technology and the availability of more definitive information will be required. Specifically, we find that complete answers to the following questions concerning Unit 2 are not incorporated in the Unit 3 PSAR:

Letter of February 28, 1966.

Questions: 3(a), (c), (d), (e), (f), (g), (h), (i), (k), (l);
4(a), (b), (c), (d), (e), (f);
5(b), (c), (d), (e), (f);
6; 8; 10; 11; 12; 13;
14(b), (c), (d), (e), (f);
18; and 19(i)

Letter of May 11, 1966.

Questions: 4, 5, 7, 8, and 9.

In addition, the information requested on the containment structural design of Indian Point Unit No. 2 in the February 28, 1966 letter has not been adequately reflected in the Unit No. 3 PSAR.

The Advisory Committee on Reactor Safeguards in its August 16, 1966 report to the Chairman on Indian Point Unit No. 2 and in its December 20, 1967 report to the Chairman on Diablo Canyon Nuclear Power Plant made specific comments on certain aspects of the plant design which warranted careful attention. An analysis and evaluation relative to all applicable points raised by the Committee should be included.

On page 2-59 of the PSAR, some features of the proposed Indian Point Unit No. 3 have been identified with Research and Development (R&D) programs which would resolve safety questions in the design of the proposed facility. These programs are presently being carried out or are planned in the near future. The following information should be discussed regarding each of these items:

1. A delineation of the responsible organization for each R&D item.

2. A description of the program for each R&D item and an analysis of the adequacy of the program to solve the problem.

3. A proposed schedule of completion for each R&D item as related to the proposed facility construction schedule.

4. A discussion of alternatives in the event the program results do not corroborate their objectives.

A comprehensive description of the quality control procedures to be followed during the construction of Unit No. 3 was not provided in the PSAR. The description should include, as a minimum, an outline of methods, procedures, and frequency of inspection, the organization responsible for inspection, the authority of this organization, and the documentation of results.

Other areas in which your application is incomplete are outlined below:

1.0 SITE

Design criteria provided are inadequate regarding the ability to accomplish a safe and orderly plant shutdown in the event of a tornado. The design criteria for the tornado should (1) indicate the design wind loading and pressure drop considered, and the basis for their selection; (2) identify the equipment which will be designed to withstand these loadings; and (3) discuss the ability of the plant components and systems to withstand tornado-originated missiles. In this respect, our criteria are that structures important to safety should be designed for tornado winds corresponding to 300 mph tangential velocity, traverse velocity of 60 mph, and a differential pressure drop of 3 psi in 3 seconds, with stresses limited to 90% of yield stress in steel and 85% of ultimate stress in concrete. Missiles associated with tornado winds should also be considered.

A safety analysis and the results thereof should be provided relative to the ability of the facility to accommodate the consequences of an explosion or fire in the gas pipeline which passes through the site. This discussion should indicate the location and proximity to the site of any gas pumping stations near the site.

A discussion was not included relative to the probability of significantly reducing cooling water flow from the river by blockage of the intake by debris during a flood, or as a result of collapse of the intake structure or its foundations during an earthquake.

## 2.0 REACTOR DESIGN

We understand from oral discussions with your representatives that burnable poison and part-length control rods are planned which would (1) limit the moderator temperature coefficient of reactivity to a negative value and (2) improve the spatial stability of the core to xenon oscillations. Since use of such modifications has not, as yet, been completely documented, your application should be amended to describe and analyze these modifications in terms of their effect on plant safety. This analysis should include a discussion of the effect of these modifications on the threshold for hydrodynamic instability and your consideration of the ability to determine local heat flux. In this respect, we believe that more extensive in-core instrumentation may be required and require your analysis of this necessity. Further examples of the type of information required relative to reactor design are given in Attachment A.

## 3.0 REACTOR VESSEL AND PRIMARY SYSTEM

There was a deficiency of information in the PSAR concerning the construction quality assurance and inspection programs of nuclear vessels and piping. We regard such programs as an important step in assuring safe function of nuclear plants. There was no discussion of such items as structural support design, construction, installation, and alignment, and alignment tolerance checks to assure that the reactor vessel, turbines, and steam generators will be properly installed and aligned and remain so following earthquake or other postulated accident. In developing this area, you should note the Commission's Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels dated August 23, 1967, and provide and discuss the following:

(a) Provide a tabulation of all the nuclear pressure vessels in the Class I (seismic design) systems in the facility. The tabulation should include a notation of whether the vessel design is complete, the stage of fabrication of the vessel, and the extent to which each of the vessels will comply with each of the 34 supplementary criteria.

(b) For each vessel, provide a discussion that presents the reasons why total compliance is not feasible for each criterion not met in its entirety.

The PSAR was deficient in discussing or presenting analysis or results of analysis, indicating whether or not the reactor vessel can accommodate, without failure, thermal shock incident upon or induced by operation of the emergency core cooling system at end of design life. Such analysis should have considered both ductile yielding and brittle fracture modes of analysis and the conditions or assumptions, criteria, and bases used in the analysis should have been provided.

Further examples of specific information in this and other related areas which should have been provided are given in Attachment B.

## 4.0 ENGINEERED SAFETY FEATURES

Your PSAR does not adequately discuss the effect that organic forms of iodine resulting from fuel meltdown may have on the capability of the proposed iodine removal system. Our preliminary evaluation indicates that off-site radiological consequences of accidents are sensitive to assumptions regarding organic or other forms of radio-iodine in the containment following an accident. Further information is needed to substantiate the fraction of iodine existing in the organic form assumed in your calculations. Recently published literature (BNL-11329) indicates that up to 16% of the total radio-iodine released from $UO_2$ fuel heated to temperatures of 1000°C to 1300°C in a steam-hydrogen atmosphere may be in an organic form. Means which show promise for removing organic iodine from containment atmospheres are in the early stage of development and will require continued research effort. Considering these facts in conjunction with the relatively high population density about your Indian Point site, our calculations indicate that you will be unable to meet the 10 CFR 100 guidelines for both the two-hour site boundary exposure, and the thirty-day low population zone boundary exposure for a TID-14844 fission product release, unless you can demonstrate that a significant amount of the organic iodine species can be removed from the containment atmosphere by containment sprays or other means. We anticipate that this will mean a research and development program which has progressed to the point where preliminary data is encouraging and which is outlined in sufficient detail so that we can estimate the likelihood of successful completion prior to the issuance of a construction permit.

Your PSAR did not include sufficient information on the design of the fan-cooler units to permit our evaluation of their adequacy. Specific examples of the type of information needed regarding the fan-coolers and the spray system are given in Attachment C.

## 5.0 SEISMIC DESIGN

Your PSAR should have included a discussion of the seismic design criteria, including maximum stress and deflection limits, applied to vessel internals and those components of the engineered safety features and other mechanical systems which are required for a safe shutdown of the plant. In this regard, we believe the Class I mechanical systems and components of these systems should be designed to meet the following criteria:

(a) Load combinations, including normal design loads and the design earthquake loads within normal working stress or deflection limits.

(b) Load combinations, including normal operating loads and the maximum earthquake loads within yield strength.

(c) Load combinations, including normal operating loads, applicable design-basis accident loads, and maximum earthquake loads, without loss of function of the specific structure, system, or component.

Specific examples of the type of information needed are given in Attachment D.

## 6.0 SAFETY EVALUATION

The Safety Evaluation section of your PSAR is deficient in many areas in that the bases for several assumptions made in your analyses are not presented, acceptance criteria are not defined quantitatively, and sufficient information is not provided on the loss-of-coolant accident to indicate the sensitivity of fuel temperature to variations in emergency core cooling system parameters. In addition, sufficient information was not presented to permit us to determine the capability of the containment to withstand additional sources of energy. Further, in your calculations of the off-site consequences of the design-basis accident, release of gap activity alone was assumed. We believe the fission product source assumed to be available for leakage following a design-basis accident should be that equivalent to a 100% meltdown with TID-14844 release fractions. Specific examples of the type of information needed are given in Attachment E.

In addition to the comments noted above, we will require the documentation of additional information in the areas of Electrical Power Distribution, Auxiliary Systems, Steam System, and Waste Disposal System. Specific comments in each of these areas are given in Attachment F.

The foregoing is a summary of some of the general areas in which we found your PSAR deficient. During our meetings of August 22, September 8, and September 26, 1967, we discussed these areas in considerable detail, and your reply should include not only replies to our specific comments in this letter, but should also include information on all items discussed with us to date. This reply should be submitted as an amendment to your application. Please be assured that we continue to be available to discuss and amplify the meaning of any of our comments, should you desire.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
Attachments A-F, incl.

cc: Arvin E. Upton, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1821 Jefferson Street, NW.
Washington, D. C. 20036

Distribution:
AEC Pub. Doc. Rm.
Formal Docket Cy.
Suppl.
REG Reading
DRL Reading
RPB-1 Reading
M. M. Mann
L. Kornblith, CO (2)
R. S. Boyd
S. Levine
D. R. Muller
J. Murphy
K. Kniel
N. Blunt

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | DRL:RPB-1 | DRL:RPB-1 | DRL:RT | DRL:RP | DRL | |
| SURNAME ▶ | JAMurphy/emb | DRMuller | SLevine | RSBoyd | PAMorris | |
| DATE ▶ | 2/8/68 | 2/8/68 | 2/15/68 | 2/15/68 | 2/17/68 | |

(SATURDAY)

## ATTACHMENT A

1. Discuss the "error band" or accuracy to which the following can be determined:

   a. Core inlet temperature
   b. Mass flow rate
   c. Inlet pressure
   d. Engineering hot channel factors
   e. Local heat flux
   f. Total core thermal power

   Relate each of the above to an error band on the DNB ratio and combine the effects of these variables to give an overall "error band" on the minimum DNB ratio.

2. Discuss the effect a small increase in power would have on the number of channels experiencing bulk boiling. Present in tabular or graphic form indication of number of channels experiencing bulk boiling vs. percent of full power for power levels up to 125% of full power.

3. Indicate the statistical number of fuel rods experiencing DNB assuming heat flux and enthalpy hot channel factors 110% of design value and power levels of 100, 105, 110, 112, and 125% of design value.

4. State the power levels at which (1) a minimum DNB ratio of 1.30 is reached, assuming design hot channel factors, and (2) fuel center melting is expected to occur.

5. Indicate the peak fuel burnup anticipated. Relate this exposure to fuel clad integrity considering fuel expansion, fuel fission gas release, and clad corrosion. Consider the effect that flaws or defects sufficiently small to escape inspection during fabrication would have on clad integrity. Cite applicable experience which justifies any claims made.

6. Provide the fuel pellet temperature profile and the fuel clad pressure time history used in calculating fission gas release from the fuel at the end-of-life.

7. Present and discuss the margins available in clad stress and strain and in peak fuel temperature between the limiting values and the values of these parameters when the core is operating at 112% of full power at the end-of-life.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

## ATTACHMENT B

## REACTOR COOLANT SYSTEM

Information Required

1. Provide details of the ductile yielding mode of failure analysis, including the following information:

   (1) The geometry of the plate and the cooling method assumed in the analysis.

   (2) The heat transfer coefficient used, its experimental basis, and the degree of conservatism involved.

   (3) The initial temperature of the vessel as a function of time delay in injecting the cold water.

   (4) The effect of axial temperature gradient in the vessel, during filling with cold water, on the total stress intensity and the distortion of the vessel.

   (5) The temperature profiles and the calculated thermal stress profiles through the thickness of the plate for several times during the cold water injection transient.

   (6) The magnitude of the axial dead load stresses in the vessel.

   (7) The magnitude of the stresses in the vessel shell due to potential simultaneous seismic loading.

   (8) The value of the yield stress used as the failure criterion in the ductile yielding analysis.

2. Provide details of the brittle fracture analysis, including the specific information listed below:

   (1) The critical stress intensity factor ($K_{IC}$) assumed, and the basis for its selection.

   (2) The assumed time-integrated neutron flux (nvt) at the reactor vessel inner diameter.

   (3) The value of residual stresses assumed in the base metal and the weld areas.

   (4) The initial crack geometry and size assumed in the analysis.

   (5) Equations used to correlate crack size with the calculated stress intensity factor ($K_I$).

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

3. Provide information which will allow an evaluation of safety margins, as follows:

   (1) An estimate of the maximum acceptable initial temperature of the vessel that could be tolerated without failure of the vessel.

   (2) An estimate of the maximum neutron flux exposure (nvt) of the vessel that could be tolerated without vessel failure.

   (3) An estimate of the maximum allowable pressure stress, when combined with other stresses present in the vessel, which could be tolerated without failure.

4. Evaluate the capability of the piping, safety injection nozzles, and vessel nozzles to withstand the transient.

5. Evaluate the effects of this transient on the core barrel and other internals with regard to assuring that distortion would not restrict the flow path of the emergency core coolant.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

ATTACHMENT C

1.0 In order that we may assess the ability of the fan-cooler units to function as proposed, the following information is requested:

1.1 A preliminary design of the heat exchanger surface and fan assembly including configuration and dimensions.

1.2 Heat transfer performance for the unit for the spectrum of accident conditions including flow rates, temperatures, pressures, and compositions for both steam-air flow and cooling water.

1.3 An outline of the analytical procedures used in designing the heat transfer surface and in determining its performance and the basis for these analytical procedures.

1.4 A discussion of experimental verification of the heat exchanger design and further experimental verification which may be required.

2.0 The following information is requested relative to the heat removal and iodine scavanging functions of the containment spray:

2.1 Since nozzle performance with respect to particle size is more critical with respect to iodine removal than with respect to steam condensation, provide nozzle performance curves of drop size vs. pressure drop and relate to optimum drop size for iodine removal. Indicate your criterion regarding size distribution for any given pressure drop.

2.2 Discuss the considerations that are required in header (or nozzle) design to assure proper drop size distribution from each nozzle, and the effect debris from the sump might have on spray performance.

2.3 In view of the importance of demonstrated function of engineered safety systems simulating, as closely as reasonable, accident conditions, discuss why you consider that a functional test of the completed Indian Point spray system is not required. Describe your plans for an engineering scale test of a typical system which could qualify the system design as having been proof tested, and evaluate the applicability of the test to the installed system.

3.0 Describe the screens provided in the sump, indicating arrangement and mesh sizes. Can the largest credible particle passing lengthwise through the screens result in a clogged nozzle or in pump damage?

4.0 Design of the sodium thiosulfate iodine removal spray system is based on the work of Griffiths. This paper presents an analytical approach for

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

determining the potential performance of sprays in removal of iodine from the containment atmosphere following a loss-of-coolant accident. There are some questions regarding the application of the work of Griffiths to the system as proposed for the Indian Point plant.

4.1 The removal coefficient for iodine is directly proportional to the diffusivity of iodine through air. This diffusivity has been shown to vary inversely with pressure and directly with temperature. It appears that the diffusivities used by Griffiths are for a pressure of one atmosphere. Discuss the corrections made to the Griffiths' model to account for the pressure and temperature conditions in the containment following blowdown.

4.2 The containment spray system is planned to be used for both iodine removal and steam condensation following loss-of-coolant accidents. The condensation process will involve the deposition of a significant film thickness of water (condensate) on each drop for the drop sizes proposed in this system. Discuss the effect of this liquid undoped film on iodine removal considering iodine to be in elemental, ionic, and organic forms. Perform a sensitivity study illustrating the variation of iodine removal coefficient with containment atmosphere relative humidity. Describe the model and absorption mechanism assumed for each of the three iodine forms noted above.

4.3 Outline any laboratory scale or engineering scale tests that are being planned to support the claimed performance of the thiosulfate spray iodine removal system. Include a discussion of the chemical stability of the thiosulfate solution when exposed to the radiation fields and temperatures which will exist following a loss-of-coolant accident. Consider both hydrogen produced by radiolysis of the water as well as thiosulfate destruction and subsequent release of absorbed iodine. Indicate the sensitivity of the stability of the system to pH.

5.0 A system used to inject sodium thiosulfate into the containment spray water is presented in the PSAR and the systems' operation is briefly described. A complete evaluation of the adequacy of the system as proposed requires additional information as follows:

5.1 Present a more comprehensive description of this system and its operation, indicating the advantage of the system proposed over other concepts, such as aspiration from the thiosulfate tank or the application of controlled gas pressure to the tank.

5.2 Provide the preliminary design of the sodium thiosulfate storage tank with regard to the inlet system and diffuser which will enable the

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

solution to be forced out of the tank with little or no mixing.

5.3 Describe the method of system calibration used to achieve the desired injection rate.

5.4 Describe the engineering tests that have been performed on this system or similar systems to assure proper function of this concept.

5.5 State the shelf life of the sodium thiosulfate as stored in this tank. What periodic testing is contemplated to check chemical stability?

6.0 Submicron particulate fission products, ($<$ 0.1 micron) including iodine adsorbed on submicron particles, may not be efficiently removed by the spray system or the absolute filters in the fan-cooler system. State and justify by referencing suitable experimental information the proportion of fission products that are assumed to be present in the containment atmosphere in submicron particulate form following a loss-of-coolant accident.

7.0 Discuss the chemical stability of the chemical additive following the design-basis accident. Consider the effects of temperature, radiation, and possible chemical reactions with materials inside containment. Identify the products of thiosulfate reaction or degradation and estimate the concentration of these products.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

ATTACHMENT D

## SEISMIC DESIGN CRITERIA FOR REACTOR INTERNALS

### Information Requested

1.0 For reactor vessel internals, and for each type of component of the reactor coolant system and the engineered safety features, provide the following:

a. The proposed stress or deformation limits for primary tensile or membrane loads.

b. The proposed stress or deformation limits for combined primary loads (tension and bending).

c. The margin of safety between the limits in b above and the expected collapse or failure condition.

Consider the following loadings:

a. Load combinations, including normal design loads and the design earthquake loads.

b. Load combinations, including normal operating loads and the maximum earthquake loads.

c. Load combinations, including normal operating loads, maximum earthquake loads, and applicable design-basis accident loads.

2.0 Identify specific reactor internals which must maintain their functional performance capabilities to assure safe shutdown of the reactor. Provide calculated (or estimated) maximum limits of deformation or stress, at which inability to function occurs, for each component identified. Also, supply the calculated (or estimated) maximum design limit value, and the expected deformation or stress. In all cases, identify the applicable loading combination and state the proposed margin of safety.

3.0 In cases where limit load analysis is to be employed, describe the method in detail. If strain hardening effects will be considered in the analysis, supply the actual stress-strain curves for the principal materials of the Class I components involved. Provide also a realistic estimate of the maximum allowable strain based on appropriate material properties at the applicable temperature in order that an estimate of the margin of safety can be made.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

Provide also information that will permit evaluation of the effect of irradiation on the material properties and of the effect of irradiation, welds, and material imperfections on the deformation limits proposed.

4.0 Supply criteria or specific information on the interaction forces, deformation and stresses connected with the relative motions between the reactor vessel, steam generators, or other large components. Indicate how these relative motions will be controlled by snubbers or other means, and what reaction forces (and corresponding stresses) will be transmitted to the pipes.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |



U.S. GOVERNMENT PRINTING OFFICE : 1966—O-214-629

## ATTACHMENT (E)

### 1.0 REACTIVITY CONTROL SYSTEM MALFUNCTIONS

1.1 Provide an analysis of the following "startup accident" for a representative spectrum of initial power levels: Simultaneous withdrawal of all rods not already in the full-out position, assuming the excursion is terminated only by action of the "High Nuclear Flux" channels and the inherent negative feedback characteristics of the reactor itself.

1.2 State your criterion regarding the minimum acceptable DNB ratio for a transient resulting from an uncontrolled rod withdrawal.

### 2.0 TRANSIENTS INITIATED BY CHANGES IN LOAD

2.1 Indicate the basis for the assumed pressurizer water level prior to the initiation of a turbine trip. What margin will be provided to assure that a "solid" hydraulic primary system will not occur? Evaluate this accident assuming one pressurizer safety valve does not open.

### 3.0 STEAM GENERATOR AND SECONDARY SYSTEM MALFUNCTIONS

3.1 Tabulate by isotope the primary system fission product inventory assumed to be present with 1% failed fuel. Describe the model used in deriving this inventory including fuel temperature, diffusion coefficients, size of perforation, and the mode of operation of the demineralizers, and rate of boron dilution.

3.2 Indicate the minimum number of simultaneous steam generator tube ruptures which would (1) result in operation of the steam system safety valves, (2) cause fuel cladding damage, and (3) uncover the core hot-spot. Relate these to core water level. Plot minimum DNB ratios, reactor water level, equilibrium primary system pressure and peak secondary system pressure vs. number of tubes ruptured to indicate the sensitivity of the number of tubes rupturing to core conditions. Assume double-ended tube ruptures at the peak of the U-bend.

3.3 Justify the iodine separation factor assumed in your analysis of steam generator tube rupture. If it is based on experimental data, relate experimental conditions to those present during a tube rupture.

3.4 It appears that rupture of the steam bypass line upstream of the steam bypass valve could result in simultaneous blowdown of all four steam generators. Analyze the consequences of this rupture indicating maximum fuel clad temperature and extent of fuel damage.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

3.5 In the event of a steam-line rupture coincident with the failure of an RCC assembly to scram, return to criticality and core damage may occur. State your criterion regarding the extent of core damage which is considered acceptable and plot power level and effective multiplication factor vs. time. Assuming a primary system fission product inventory equivalent to that resulting from this limiting core damage plus an additional 1% failed fuel, calculate the dose to the environs considering the maximum primary-to-secondary steam generator tube leakage at which the loop will be allowed to operate. Perform a similar analysis considering the consequences of rupture of the steam bypass line.

3.6 State the maximum stresses experienced in the tubes and tube sheet during a steam-line rupture transient, including thermal stresses.

3.7 State the basis for the initial pressurizer level assumed in your analysis of a steam-line rupture. Indicate the margin in pressurizer volume which remains after injection of sufficient boron to render the core subcritical.

3.8 In the PSAR it is stated that in the event of a steam-line rupture accompanied by a stuck control rod and loss of off-site power, there will be ". . . no consequential damage to the primary system and the core will remain in place and intact." Define this criterion in terms of primary system pressure and stress levels; clad stress, strain, and maximum temperature; number of rods undergoing DNB; and metal-water reaction.

4.0 <u>MISCELLANEOUS INCIDENTS</u>

4.1 In the PSAR it is stated that in the event of loss of electrical power to all reactor coolant pumps, the plant will be designed in such a manner that the resulting flow coastdown and reactor trip will ". . . prevent fuel failure and reactor coolant system overpressure." Define this criterion in terms of maximum primary system pressure and minimum DNB ratio experienced.

4.2 Indicate the basis for the fission product inventory assumed in the volume control tank and the gas decay tank. List all assumptions made in determining the quantity of radionuclides released to the atmosphere from rupture of these vessels, including air temperature and relative humidity, liquid temperature, halogen vapor pressure, radioactive decay, etc. Discuss the effect that a load-following mode of operation would have on the inventories in these tanks.

| OFFICE ▶ | | | | | | |
|---|---|---|---|---|---|---|
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |



U.S. GOVERNMENT PRINTING OFFICE : 1966—O-214-629

4.3 Justify the assumption that only one row of fuel rods could be damaged in a fuel-handling incident. Consider the effect of dropping one element on another during refueling. Relate your assumption to fuel damage which has occurred during handling at operating reactors.

4.4 State the basis for the gap activity assumed to be present in the damaged fuel rods during a fuel handling incident and for the halogen partition factor assumed.

4.5 To illustrate the safety margin which exists due to the inherent design of the facility, identify the maximum hypothetical missile originating from the turbine which would not penetrate the containment vessel. To indicate the sensitivity of casing energy absorption on your analysis, present this information, assuming casing energy absorptions of zero, 25, 50, and 100 percent of your best estimate of absorption.

5.0 LOSS-OF-COOLANT ACCIDENT

5.1 Discuss the effect of the loss-of-coolant blowdown on steam generator tube and tube sheet integrity and state the maximum leakage rate from secondary-to-primary which could occur following blowdown. Indicate the stresses experienced due to the differential pressure following blowdown and the thermal stresses resulting from blowdown. Consider the plant to be approaching the end of life with the maximum primary-to-secondary leakage with which it would be operated.

5.2 Provide plots of water volume in the pressure vessel, clad hot spot temperature, percent metal-water reaction, and percent of clad experiencing perforations vs. time for cold leg and hot leg breaks. Assume break areas of 0.005, 0.1, 0.5, and 3 square feet and a double-ended break of the main coolant line. Assume that (1) the emergency core cooling system does not function, (2) minimum engineered safety features operate with two accumulators discharging to the vessel, (3) minimum engineered safety features operate with three accumulators discharging to the vessel, and (4) the system functions at 100% of capacity. For reference, include the adiabatic (from time of rupture) clad temperature transient on the temperature plots.

5.3 Provide a plot of time to recover the lower half of the core vs. break area assuming minimum engineered safety features and time to reach 1% clad metal-water reaction and time to reach incipient clad melt vs. break area assuming the emergency core cooling system does not function. Consider a range of break areas from 0.005 square feet to that equivalent to a double-ended cold leg rupture.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

5.4 On one graph for each break size, plot hot-spot heat transfer coefficient, average vessel pressure, core flow rate, and energy input to the hot element vs. time. Provide this information of hot and cold leg breaks of 0.005, 0.1, 0.5, and 3 square feet and also for double-ended ruptures of the main coolant line.

5.5 It appears that if a steam bubble were formed in the pressure vessel head following a loss-of-coolant accident due to establishment of a water seal between the main coolant pump and the steam generator, sufficient back-pressure might result to limit the amount of cooling water entering the core and thus delay core cooldown. Describe the method whereby this is considered. Present a curve of maximum clad temperature vs. time assuming (1) two accumulators discharge to the reactor vessel, and (2) three accumulators discharge to the vessel. Continue these curves until peak clad temperature is below the saturation temperature associated with containment pressure. In deriving these curves, assume minimum engineered safety features are operable.

6.0 <u>LOSS-OF-COOLANT ACCIDENT CONTAINMENT PRESSURE RESPONSE</u>

6.1 For the analysis performed of the pressure response of the containment to a loss-of-coolant accident during blowdown, provide the following:

a. Assumptions of flow conditions during both subcooled and two-phase blowdown.

b. Basis for the flow coefficients used.

c. Basis for the heat transfer coefficients employed in transferring core sensible heat during blowdown.

d. Determination and method of addition of energy generated by fission within the core and decay.

e. Assumption regarding relative humidity and basis for the value employed.

6.2 For the period after the initiation of blowdown, provide the following:

a. A listing of the effective surface areas and thickness of the static sinks and a justification for the heat transfer coefficients applied.

b. A description of the thermal resistance assumed between the liner and containment concrete and a justification of the amount of heat assumed to be absorbed by the concrete during blowdown.

c. A discussion of the adequacy of the heat transfer coefficients assumed for transfer of heat from hot metal.

6.3 Provide a plot of peak containment pressure vs. rupture diameter to permit determination of the most severe loss-of-coolant accident.

6.4 In order that we might understand the magnitude and accuracy of the individual energy sources and sinks, provide the curves of the following parameters as a function of time: (Assume the rupture area associated with the peak of the curve in 6.3 above and that minimum safeguards operate.) Include error limits indicating maximum and minimum values of these parameters on these curves.

a. Energy sources.

(1) Energy transferred to containment from core sensible heat.

(2) Energy transferred to containment from hot metal.

(3) Decay heat energy transferred to containment.

(4) Energy of zirconium-water reaction and associated hydrogen oxidation.

b. The derivative with respect to time of the parameters in "a."

c. Energy sinks.

(1) Energy absorbed by static sinks.

(2) Energy absorbed by fan-cooler units.

(3) Energy absorbed by spray water.

(4) Energy absorbed by containment liner.

(5) Energy absorbed by containment concrete.

(6) Energy absorbed by the miscellaneous metal within the containment.

(7) Energy absorbed by internal concrete structures.

d. The derivative with respect to time of the parameters in "c."

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

6.5 Present curves of containment pressure vs. time for the following cases, assuming the safety injection system, including its associated accumulator subsystem, does not operate, metal-water reaction occurs as determined by core heatup and steam availability, and the rupture area is that associated with the peak of the curve in 6.3 above:

(1) Minimum containment engineered safety features operate.

(2) One residual heat removal pump and three containment fans operate, containment spray system inoperative.

6.6 In order to evaluate the margin provided in the containment design for metal-water reaction, provide a plot of peak containment pressure vs. percent metal-water reaction. Assume the rupture area associated with the peak of the curve requested in 6.3 above, minimum engineered safety features operate, and linear metal-water reaction rates of 0.05%/sec. and 0.10%/sec.

6.7 Discuss the possibility of a localized pressure pulse originating within the containment which could damage structural members if pipe rupture were to occur in closed areas (e.g., the space between the nozzles and the shield wall, the primary pump and steam generator compartments, etc.).

7.0 DOSE CALCULATIONS

7.1 State your criterion regarding acceptable thyroid and whole body doses to operators in the control room following a fission release equivalent to a 100% meltdown with TID-14844 release fractions. Assume no credit is taken for operation of the isolation valve seal water system; i.e., assume a containment leakage rate of 0.1%/day.

7.2 Considering the relatively high population density in the vicinity of the proposed Indian Point site, and in order that we might assess the sensitivity of the off-site consequences of the design-basis accident to assumptions made concerning the chemical form of the iodines and the removal coefficient ($\lambda$) of the reagent spray, you are requested to submit plots of the following parametric studies:

(1) Average iodine reduction factor as a function of removal coefficient ($\lambda$) for time periods of 0-2, 0-12, and 0-24 hours and for 0-30 days. Assume organic iodine content of 0, 5, 10, 15, and 20% of the total radio-iodine.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

(2) Two-hour and 30-day thyroid and whole body doses vs. distance, assuming 0, 5, 10, 15, and 20% organic iodine and assuming (1) no iodine removal by the sprays and (2) spray efficiency as predicted by your calculations.

(3) Number of people exposed vs. whole body and thyroid dose received in 2 hours, 12 hours, and 30 days, assuming organic iodine fractions of 0, 5, 10, 15, and 20%. Present this for each of the two 22-1/2° sectors having the highest population.

Assume a TID-14844 fission product release with a containment leak rate of 0.1%/day for the first 24 hours and 0.045%/day thereafter.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

## APPENDIX F

1. Discuss the operation of the emergency diesel power supply system under accident conditions with no normal power sources available. Indicate the sequence of equipment that is automatically started during the injection phase (including designation and horsepower of each), and the loads (designation and horsepower) on each diesel for the recirculation phase. Give basis for the rating of the diesel proposed to furnish these loads. Confirm that after any single fault or failure, including the failures of any one diesel to start, sufficient power is available for engineered safety features. Describe equipment used for the automatic sequence loading and consider a failure in one automatic sequence. Confirm that all equipments, including diesels, fuel, auxiliaries, controls, wiring, etc., have the physical separation required to prevent a single accident (including fire) from disabling more than one diesel power supply.

2. Confirm that a fault on any bus will lock out all possible sources of power to that bus until the fault is cleared, and that lockouts will be provided for bus ties to prevent any two power sources from being tied together. Confirm that circuit breakers connecting Emergency Generators 1, 2, and 3 to Buses 5A, 2A, 3A, and 6A will not close if there is a voltage on the bus from any source that is not synchronized, and that Emergency Generators that are synchronized with power from Station Auxiliary Transformer supply are left connected to that power only as long as necessary for load testing purposes.

3. Evaluate the ability of the system to supply power to safety loads under accident conditions with a loss of outside power, and with any single fault or failure in the d.c. system. Additional information is needed, including diagrams, of the 125V d.c. system and the 120V a.c. instrument supply system. This information should include assurance that buses, batteries, and inverter sets are physically separated so that a single accident could not take out both sources of supply to controls, instruments, and other important loads.

4. Evaluate the ability to supply electric power from the incoming power lines to the engineered safety features. Include, as a minimum, the effect of sudden trip of the unit, fault on the incoming lines, fault or equipment failure in the Buchanan substation, or fault or equipment failure within the plant. Supply pertinent statistics showing the ability of the system to withstand the tripping of large units.

5. Evaluate the ability of all electrical components required for safety to withstand the accident environment. Include an identification of the equipment, and an estimation of the length of time that each piece of equipment must function. Discuss these special design provisions which enable the motors, valves, wiring, and any other components to function in the accident environment.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |

6. We believe the component cooling water and the service water systems should be designed to accommodate a single failure, either active or passive, as a means of insuring long-term function following a loss-of-coolant accident. Describe and discuss the modifications required to meet this goal.

7. Provide a detailed diagram of the steam supply system line which leads from the main steam header to the auxiliary feedwater pumps. Evaluate the consequences of a single failure in the steam supply to these pumps.

8. Describe the pressure relief protection provided for the gas decay tanks associated with the waste disposal system.

9. Describe the ventilation provisions provided for the areas near the waste storage tanks and the engineered safety feature equipment.

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | | | | | | |
| SURNAME ▶ | | | | | | |
| DATE ▶ | | | | | | |