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U. S. ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

REPORT NO. 1

Note by the Director, Division of Reactor Licensing

The attached report has been prepared by the Division of Reactor Licensing for consideration by the Advisory Committee on Reactor Safeguards at its April 1966 meeting.

Introduction

Consolidated Edison Company of New York, Inc. applied for a construction permit on December 6, 1965, for a 2758 Mw(t) nuclear power plant. The proposed pressurized water reactor will be the second nuclear unit to be located at the Indian Point site. The existing PWR, Indian Point Unit No. 1, has a thermal rating of 615 Mw(t). Meetings between the applicant and the Division of Reactor Licensing staff were held on January 17-18, 1966, and additional information was requested by letter dated February 28, 1966. The applicant expects to file the requested information about March 31, 1966.

This report discusses the status of the staff's review of this project with respect to: (a) site characteristics related to locating a large PWR at the Indian Point site, (b) those design features (containment and engineered safeguards) which are included to compensate for the relatively high population density, (c) the accident analysis, and (d) other related topics. The proposed facility is being considered by the Committee at this time for information only. An ACRS Subcommittee meeting at the Indian Point site is scheduled for March 30, 1966, and we anticipate a second full Committee meeting in June.

Discussion

The power level of the proposed Indian Point II reactor is considerably higher than any previously reviewed pressurized water reactor (previous high was Connecticut-Yankee at 1473 Mw(t)). The reactor is similar to the San Onofre, Connecticut-Yankee, and Brookwood facilities in most respects as regards general design and operating objectives. The reactor will be located at the Indian Point site which is in a region of fairly dense population; therefore, the applicant has proposed an improved containment design along with other engineered safeguards which it believes can adequately limit the potential off-site doses in the event of the maximum credible accident. The proposed containment (reinforced concrete with a single steel liner) is similar to Connecticut-Yankee, but will be designed to preclude leakage through all known potential leakage sources under accident conditions.

A. Site Characteristics

The Indian Point site comprises 250 acres owned by the Consolidated Edison Company of New York, Inc. and is located on the eastern shore of the Hudson River in Westchester County, New York. The site is located 2.5 miles from the center of Peekskill, N. Y. and approximately 24 miles north of New York City. The current Peekskill population is 19,000 within an anticipated growth to 30,000 by 1985. The population in the vicinity of the site is as follows:

	Cumulative Population	
Distance	1960	1980
0.5 mile	46	100
1	1,080	2,110
2	10,810	20,910
3	29,630	59,520
4	38,730	78,860
5	53,040	108,060
10	155,510	312,640
15	326,930	670,210

The minimum exclusion distance for the Indian Point site is 0.32 miles (520 meters). The nearest boundary of Peekskill, the nearest population center is 0.87 miles (1400 meters). In view of the short distances involved in this case, it is evident that the specifics of 10 CFR 100 are not too meaningful. For this reason, we have elected to evaluate off-site doses at 0.32 miles and 0.67 miles, which are the distances assumed by the applicant for exclusion distance and low population distance, respectively.

The meteorological parameters for the Indian Point site have previously been determined by on-site measurements in conjunction with the operation of Unit No. 1. This study provided the diffusion coefficients (Cy, Cz, n) for various lapse conditions, but not under inversion conditions. These measurements indicated that inversion conditions exist at the site about 42% of the time. Accordingly, in evaluation of accident consequences the applicant has assumed the parameters recommended for inversions in TID-14844, with a modification for the effects of building wake, to determine the atmospheric dispersion for these conditions.

We have received the comments from the U. S. Weather Bureau concerning the site meteorology, and it has concluded: "In summary the computed atmospheric dispersion factors for both the short term and long term accidents are realistic and somewhat conservative in light of the meteorological conditions observed for a year's period at the site meteorological tower." Accordingly, we have used the diffusion parameters presented by the applicant to compute the potential off-site doses given later in this report.

Comments have not yet been received from the staff's consultants concerning the hydrology, geology and seismology of the Indian Point site. We anticipate no problems with respect to hydrology, geology, or seismology. With respect to seismology, we understand that the USC&GS intends to recommend a maximum earthquake acceleration of 0.1g (without loss of function) which agrees with that proposed by the applicant.

B. Containment

1. Design Description

The Indian Point containment is a reinforced concrete structure having a single steel liner and is designed for a reference incident pressure (design pressure) of 47 psig. The containment penetrations have been designed to preclude leakage following the postulated maximum credible accident. In the following section, the pressures resulting from various potential energy inputs to the containment atmosphere are discussed. It should be noted that the containment stresses will not exceed 95% of yield for accident pressures up to 70 psig (1.5 times design pressure) with no earthquake, or 59 psig (1.25 times design pressure) with a simultaneous earthquake of intensity 1.25 times design acceleration of 0.1g. The free volume of the containment is 2.61×10^6 cubic feet.

The following systems have been provided to limit containment leakage:

a. Penetration Pressurization System - The containment liner penetrations are designed with double seals to permit continuous pressurization during plant operation. The space between the double seals will be continuously pressurized to approximately 50 psig to exclude outleakage under accident conditions. Pressurization will be provided by air compressors with a gas bottle system as backup if power is lost. The gas consumption of the system will be continuously monitored and recorded, and will provide continuous verification of system integrity. Individual sections can be isolated to permit the location of leaking components. The penetration pressurization system is a new containment feature proposed for this facility.

b. Isolation Valve Seal Water System - This system is designed to provide a water leg in lines penetrating the containment to eliminate possible leakage paths through pipes and valves to the atmosphere. The water leg is established using gas bottle pressurization to eliminate system dependence on electrical power. This system is not provided for closed piping systems inside the containment (ones which do not connect to the containment atmosphere or a source of radioactivity within the containment) that are designed for pressures above design pressure and are provided with missile protection. This system is similar to the one proposed for the Malibu containment.

c. Containment Liner Joint Integrity - All containment liner joints will contain testing channels to permit periodic verification of weld integrity (continuous pressurization can be provided, if required). Pressurization for this system will be provided by the penetration pressurization system.

Westinghouse has stated that the proposed Indian Point II containment is superior from a leakage viewpoint to the double containment proposed for Malibu. They believe that by operating the penetration pressurization system at a differential pressure of 50 psig, greater assurance against outleakage is provided than with the pumpback system which provided only a slight negative pressure (10 inches of water) in the popcorn concrete zone between the two containment liners. The staff is not yet in a position to evaluate the relative merits of these systems since additional information requested concerning operation of the penetration pressurization system and identification of all lines that will and will not be provided with the isolation valve seal water system has not yet been received (See question No. 19e of our letter of 2/28/66).

2. Containment Design Requirements

The applicant has presented a study of the containment pressure after an assumed double-ended rupture of the largest primary pipe for the cases listed below. The examples are listed so as to emphasize what the applicant believes falls within the range from typical to worst cases which should be considered.

a. All components of the safety injection system, containment spray, containment ventilation fan coolers operate. A 1% $Zr-H_2O$ reaction occurs resulting in a peak pressure of 40 psig.

b. No engineering safeguards operate (i.e., no safety injection, no spray, and no fan coolers). A 33% $Zr-H_2O$ reaction occurs resulting in a peak pressure of 42 psig.

c. Safeguards driven by 2 of the 3 emergency diesel generators. A 4.7% $Zr-H_2O$ reaction occurs resulting in a pressure of approximately 40 psig.

d. Delays in initiation of safety injection for arbitrary delays up to 15 minutes with and without some engineered safeguards. No mention of amount of $Zr-H_2O$ reaction; a peak pressure of approximately 45 psig.

e. Same as above but it was also assumed that hydrogen gas released by the metal-water reaction accumulated and then burned (rather than igniting spontaneously). No mention of amount of $Zr-H_2O$ reaction; a peak pressure of 56 psig.

It is obvious that many additional cases could have been examined, such as situations with the containment spray system operating, but not the ventilating fans, and situations involving a range of $Zr-H_2O$ reactions. The containment pressures resulting from these analyses would cover a wide scale of values including ones well above design. In fact, given the amount of energy, water, and zirconium available in the Indian Point II system and combining them at the appropriate time, it is not difficult to envision insurmountable containment problems - combining them at other times obviously can lessen the consequences.

In addition, because of the use of computer codes by the applicant which employ various analytical models (not necessarily verified by experimentation) and which incorporate various assumptions at different points in the analysis, it is difficult to determine the conservatism or relative importance of the various parameters involved in the cases that are examined. In view of the preceding comments, the staff is attempting to develop an approach based on energy considerations which it believes may allow a better perspective in which to judge the maximum containment pressures that could arise and the need, effect, and margin of safety provided by certain engineering safeguards. In line with this approach, we have asked the applicant to (See question No. 4):

(1) Indicate the total energy available from all primary and secondary sources including decay heat and 100% metal-water and hydrogen recombination reactions.

(2) Show the fraction of each of the component heat sources in the containment atmosphere as a function of time and show the energy the containment and structures can absorb as a heat sink as a function of time.

To indicate how we will use the type of information we expect to obtain, the staff has calculated the initial containment pressure following a loss of coolant accident utilizing all energy sources available before the blowdown.

Energy Source	Energy-BTU	% of Total Available	Total Cumulative Pressure-Psig
Primary Coolant	3.09×10^8	78	38.6
Internal Core Energy Available for Transfer	0.20×10^8	5	41.8
Excursion Due to Positive Moderator Coefficient ($1 Insertion)	0.027×10^8	1	42.2
Secondary Coolant (one loop)	0.62×10^8	16	49.6

The staff has also calculated the energy available after the blowdown.

Energy Source	Energy-BTU	% of Total
Primary System	3.09×10^8	48.8
$Zr-H_2O$ (100%)	1.23×10^8	19.4
H_2 Recombination	1.0×10^8	15.9
Decay Heat (30 min.)	1.0×10^8	15.9
(10 min.)	0.4×10^8	--

Based on the preceding results we chose to calculate the increase in containment pressure which would result from an initial blowdown energy of 41 psig (approximately what the applicant calculates) followed by varying amounts of $Zr-H_2O$ reaction resulting in hydrogen recombination. The results are based on the assumption of no heat losses to the containment or other structures and uses the energy available from the hydrogen recombination to superheat the containment atmosphere (the $Zr-H_2O$ heat as well as decay heat energy remains in the reactor vessel).

% H_2 Recombination	Energy to Containment-BTU	Containment Pressure-psig
10	0.1×10^8	48.5
20	0.2×10^8	56
33	0.33×10^8	66

The applicant in contrast to the above assumption of no heat losses to the containment and structures has examined a 33% H_2 recombination case assuming only heat losses to the containment and structures (see page 6, Case b). For this condition the pressure remained relatively constant for the first hour indicating the energy abosrbed in the containment as a heat sink equaled the energy available from the H_2 source. This example appears to indicate that (1) the assumptions used in calculating the energy absorbed by the heat sinks for this case, or engineering safeguards in other situations, are critical in determining the containment pressure, and more importantly

(2) the amount of $Zr-H_2O$ reaction occurring as a function of time is of prime significance since the heat sink capability may not be sufficient for short reaction times. There is sufficient decay heat, given the proper mechanism, to raise all the clad to the $Zr-H_2O$ reaction temperature in 7 minutes. This being the case, the $Zr-H_2O$ reaction assumed to occur should properly be used in a parametric study with the time for the given amount of reaction being varied. We expect to review the answers to our questions with the aforementioned comments in mind, and with the thought of requiring the applicant to prepare containment capability curves which would show the containment pressures resulting from a given metal-water reaction occurring over various time intervals with the energy absorbed in the heat sinks and removed by the safeguards as an additional parameter.

C. Accident Analysis

With the assumption that leakage from a containment building is to be expected at the penetrations, the Indian Point II containment has been designed to have essentially negligible leakage following the assumed double-ended failure of the largest primary pipe. If this can be demonstrated, the only engineered safeguards that would be required are those necessary to prevent overpressurization and subsequent failure of the containment. Safety injection, containment spray and air recirculation systems have been provided to perform this function (similar to Brookwood and Connecticut-Yankee). The design criteria of these systems are that containment integrity shall be maintained under all credible accident conditions.

The verification of negligible containment leakage under accident conditions may be difficult to demonstrate experimentally because of the limitations of current leakage measuring techniques. As a result, the applicant has provided two iodine removal systems to limit the potential off-site doses following the MCA. Calculations of the effectiveness of these systems assume a containment leakage rate of 0.1%/day at 47 psig. These systems are the internal air filtration system (impregnated charcoal filters) and the containment spray system. A chemical reagent (sodium thiosulphate) will be added to the containment spray water to aid removal of elemental forms of iodine. The chemical additive is also a new feature for this facility and should improve the effectiveness of the spray system. The system is designed to provide iodine removal equivalent to that provided by the internal air filtration system.

The potential off-site doses following the MCA have been calculated by the staff to indicate the degree of iodine removal required (filter efficiency) to meet the siting criteria suggested in 10 CFR 100 under the following assumptions:

Power level - 2758 Mw(t)
Equivalent I^{131} available for leakage - 3.2×10^7 curies
(25% of total inventory)
Unfilterable iodine - 5%
Fan capacity - 65,000 CFM each, 4 of 5 operating
Recirculation rate - 6 containment volumes per hour
Filter efficiency - as indicated
Containment leakage rate - (ground release) 0.1%/day for
first day, 0.045%/day for next 30 days
Flow bypassing filters - 10%
Credit for building wake effects

The following is a comparative tabulation of the potential off-site doses using various combinations of the calculational model and filter efficiencies. To account for building dilution, the model suggested by Gifford & Fuquay has been used.

Condition	Filter Efficiency (%)	Integrated Thyroid Dose (rem) Site Boundary (520 meters) 2 hr.	Low Population Distance (1100 meters) 30 day
TID-14844 assumptions on meteorology and constant 0.1% leakage rate (no credit for building wake)	0	2,390	39,000
Assumptions as stated above	0	870	3,250
	30	300	*
	45	*	300
	90	130	230

* Not calculated

The above table indicates that a filter efficiency of 45% is required to limit the low population (30 day) and exclusion distance (2 hour) thyroid doses to 300 rem or less. These doses would be considerably lowered by functioning of the containment spray system and by functioning of the containment pressurization system. The calculated doses are a direct function of the percentage of iodine assumed to be unfilterable. For example, by increasing the unfilterable fraction from 5% to 10%, the 30 day thyroid dose at 1100 meters (assuming a filter efficiency of 90%) is increased from 230 to 380 rem. We believe that the assumption of 5% unfilterable iodine is acceptable for calculational purposes for Indian Point II (it was also assumed for Brookwood), but have requested that the applicant provide an analysis of the available experimental data in this area. (See Question No. 6)

We recognize that the containment vessel may not be the only source of leakage of radioactive material under MCA conditions. Accordingly, we intend to evaluate the off-site doses from all potential sources of release of radioactive material as our review progresses. (See Question No. 9)

D. Related Topics

The following topics are areas in which the staff believes that the Indian Point II facility represents a significant extrapolation from the Brookwood facility design or are worthy of special consideration due to the size and location of the reactor.

1. Thermal Analysis - The Indian Point II reactor, besides having the highest power rating of any proposed or currently licensed reactor, has many thermal and hydraulic parameters which are closer to design limits than for previous cases. Table I compares these parameters for the Indian Point II, Brookwood, and San Onofre reactors.

TABLE I

COMPARISON OF THERMAL & HYDRAULIC PARAMETERS

	Indian Point II	Brookwood	San Onofre
1. Maximum Specific Power, Kw/ft	18.5	16.7	15.0
2. Maximum Heat Flux BTU/hr-ft^2	571,000	518,000	463,000
3. Average Heat Flux BTU/hr-ft^2	175,000	152,000	143,000
4. Average Mass Velocity lb/hr-ft^2	2.6×10^6	2.43×10^6	2.02×10^6
5. Core ΔT, °F	57	54	49
6. Number of Fuel Rods	40,000	21,000	28,000
7. Core Equivalent Diameter, inches	134	96	111
8. DNBR at nominal conditions	1.81	1.90	2.07

		Indian Point II	Brookwood	San Onofre
9.	Hot Channel Factors	3.25	3.41	3.23
10.	Maximum Overpower, %	112	112	118

In addition, the W-3 correlation used to predict burnout represents a "best-fit" of experimental data at pressures, geometries, and lengths not found in the Indian Point reactor. Based on the applicant's statistical analysis of the data points, at a DNBR of 1.3 there is a 5% probability of DNB occurring; at a DNBR of 1.49 there is a 1% probability; and at 1.77 there is a 0.1% probability.

We believe the applicant's presentation of thermal analysis results (i.e., statement of nominal and transient DNB ratios) does not allow an assessment of the conservatism of the proposed design. We have asked the applicant a series of questions (See Question No. 5) which are expected to result in information to allow us to ascertain more clearly the condition of the core during nominal and overpower conditions, and the required power level increase before significant fuel damage occurs. In particular, we have asked for a distribution curve showing the fraction of the core operating above various power levels with their corresponding DNB ratios for the design and overpower conditions, and an indication of how many channels require only 5% additional power to cause bulk boiling. We have asked the applicant to estimate whether any fuel rods approach design limits (e.g., DNB or center fuel melting) at a hypothetical 125% overpower condition. Finally, we have requested the applicant to arbitrarily raise the hot channel factors in heat flux and enthalpy by 10% and report the results of its thermal analysis for assumed conditions of 100, 110, and 125% of nominal power. With this information we will be able to assess the conservatism of the thermal design.

2. Reactivity Accidents - The staff has reviewed the applicant's safety evaluation of the effects of plant abnormalities and transients, and because of the lack of any specific information we have requested evaluations to support PSAR results for the startup accident, steam line rupture, refueling, and control rod cluster ejection accident (See Question No. 11). The staff believes the applicant has not fully evaluated the consequences or possible mechanisms of a massive reactivity excursion. In particular, we believe the applicant should evaluate (1) the type of damage which could result from energy releases within the reactor vessel (i.e., rupture of the primary system boundary or massive fuel rod failures), (2) the required energy to initiate such failure, (3) the reactivity insertion (ramp or step) which would produce the required energy, and (4) the possible mechanisms for inserting the reactivity.

The applicant's assumption that no reactivity insertion considered causes a potential safety hazard does not indicate how close their calculated situation is to a severe accident condition.

In line with this, we have requested the applicant to consider cases of hypothetical reactivity insertions considerably greater than that which it has assumed for the ejection of a single control cluster, and to consider the generation of curves that relate reactor period to (a) integrated excursion energy and (b) average fuel temperature (See Question No. 11). We have also asked the applicant to state and justify the energy required to initiate failure of the primary system boundary and to state possible mechanisms to provide this amount of energy by reactivity insertion (See Question No. 3L). We expect the applicant to indicate, for the situations examined, the reactivity inserted and the reactivity worth of the shutdown mechanisms (e.g., Doppler and void) showing at least some sort of parametric survey to account for uncertainties in these shutdown mechanisms (e.g., moderator coefficients which could be positive or negative and scram delay times).

3. Xenon Oscillations - Spatial instability due to xenon oscillations is a function of the uniformity of the core power distribution. The applicant has stated that the core will be stable early in life but as burnup progresses and axial flux peaking is reduced, calculations indicate that a xenon oscillation may occur. The applicant believes the in-core and out-of-core nuclear detectors will detect this oscillation and rod insertion can be used to maintain the core within safe limits. We believe the applicant has not fully presented the possible magnitude or error in determination of the xenon oscillation, or demonstrated that he has factored this into his safety analysis by using appropriate axial flux shapes, for example, in his thermal analysis calculations. The applicant has been requested to provide experimental evidence to indicate the sensitivity of the external ion chambers to changes in the axial and radial flux distribution (See Question No. 14F).

4. Plant Interaction - The proposed facility will be the second PWR to be located at the Indian Point site. The staff recognizes that some interaction between the two reactors may exist, and has requested the applicant to discuss the relation to nuclear safety of any systems or equipment that will be shared by both facilities (See Question No. 17).

Conclusion

From our review thus far, it is evident that the Indian Point site and size of Unit No. 2 pose special safety considerations relating to containment and engineered safeguards for the former and to thermal and accident analyses for the latter. However, based on the information thus far presented in the application and considering the understanding developed in meetings with the applicant, we do not foresee any insurmountable problems with the design and construction of Unit No. 2. We expect to be in a position, following receipt and evaluation of the requested information, to be able to fully discuss, in Report No. 2, the overall acceptability of the applicant's proposal to design and construct the Indian Point II reactor.