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. 2

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 May, 1966 meeting.

Introduction

The proposed Indian Point Unit No. 2 pressurized water reactor is being considered for information only by the Committee at its May meeting. The Committee also considered this project for information at its April meeting, and we anticipate a third full committee meeting in June. A subcommittee meeting was held at the Indian Point site on March 30, 1966, and a second subcommittee meeting is scheduled for May 3, 1966.

The Consolidated Edison Company of New York filed the First Supplement to its application on March 31, 1966. This supplement provided additional information requested by the staff by letter dated February 28, 1966.

The status of the staff's review of the information presented in the First Supplement is given in this report. The principal items discussed are: (a) thermal and hydraulic analysis, (b) containment design requirements, (c) reactivity effects, and (d) items referred to special consultants. Much of the information presented in the First Supplement was requested to determine conformance of the proposed design with the "General Design Criteria for Nuclear Power Plant Construction Permits." The conformance of the proposed facility to these criteria will be discussed in the staff's report for the June ACRS meeting.

Discussion

The items discussed in this report are those which the staff believes are most significant to the safety of this facility. The analyses presented are intended to show the adequacy of the proposed design and the margin of safety provided.

A. Thermal and Hydraulic Analysis

We have reviewed the applicant's original presentation and subsequent answers to our questions (answer No. 4 of Supplement 1) concerning the thermal and hydraulic core analysis with the intent of determining the condition of the core during normal and overpower situations with particular attention to an assessment of the safety margin available before large numbers of fuel rods exceed design limitations.

Although we believe that damage to a single fuel rod is not necessarily a safety consideration, knowledge of how close the core is to a substantial number of fuel rod failures is of safety significance.

Table 1 and Figure 1 present the nuclear power peaking factors and a distribution curve showing the number of fuel rods operating at various relative power levels. The radial nuclear factor, F_r, indicates the power from a given rod with $F_r = 1$ representing the average rod power.

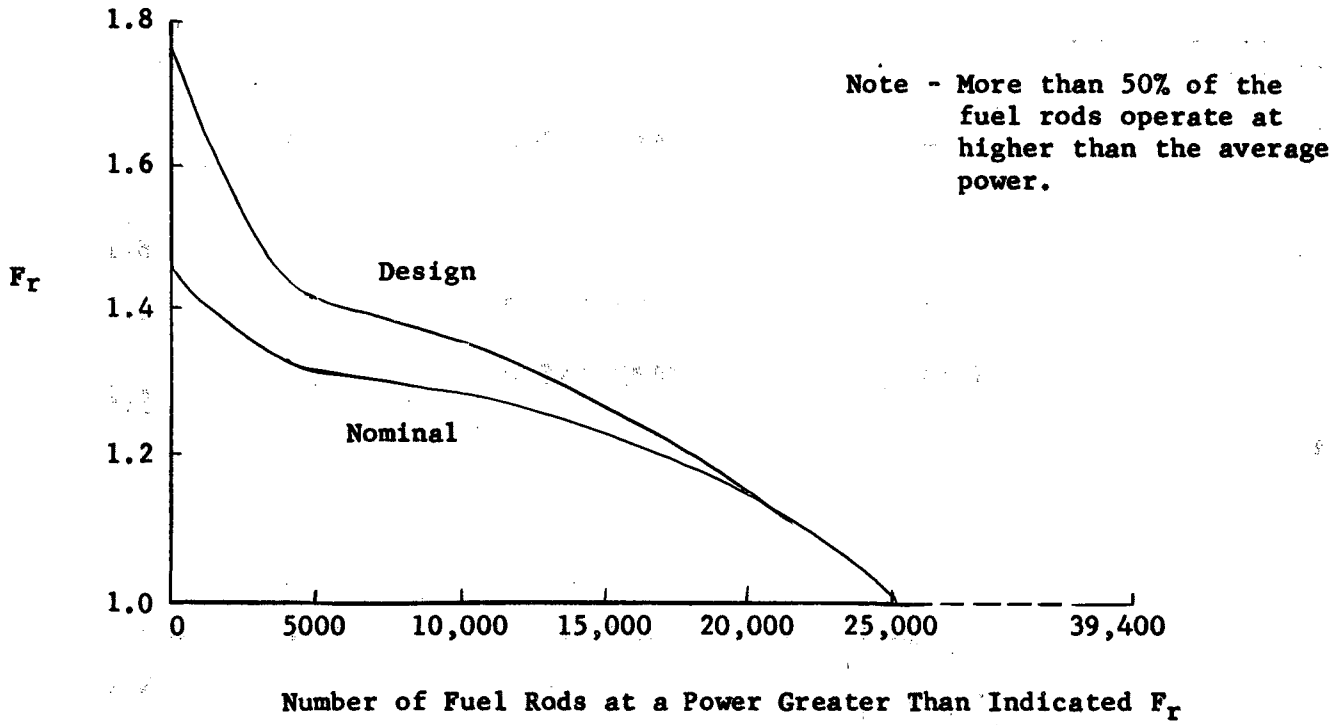
Table 1 - Nuclear Power Peaking Factors

	Nominal	Design
F_r	1.45	1.75
F_z	1.59	1.78
F_q	2.30	3.12

Where: F_r is the peak radial power factor
F_z is the peak axial power factor
F_q is the total local peaking factor ($F_r \times F_z$)

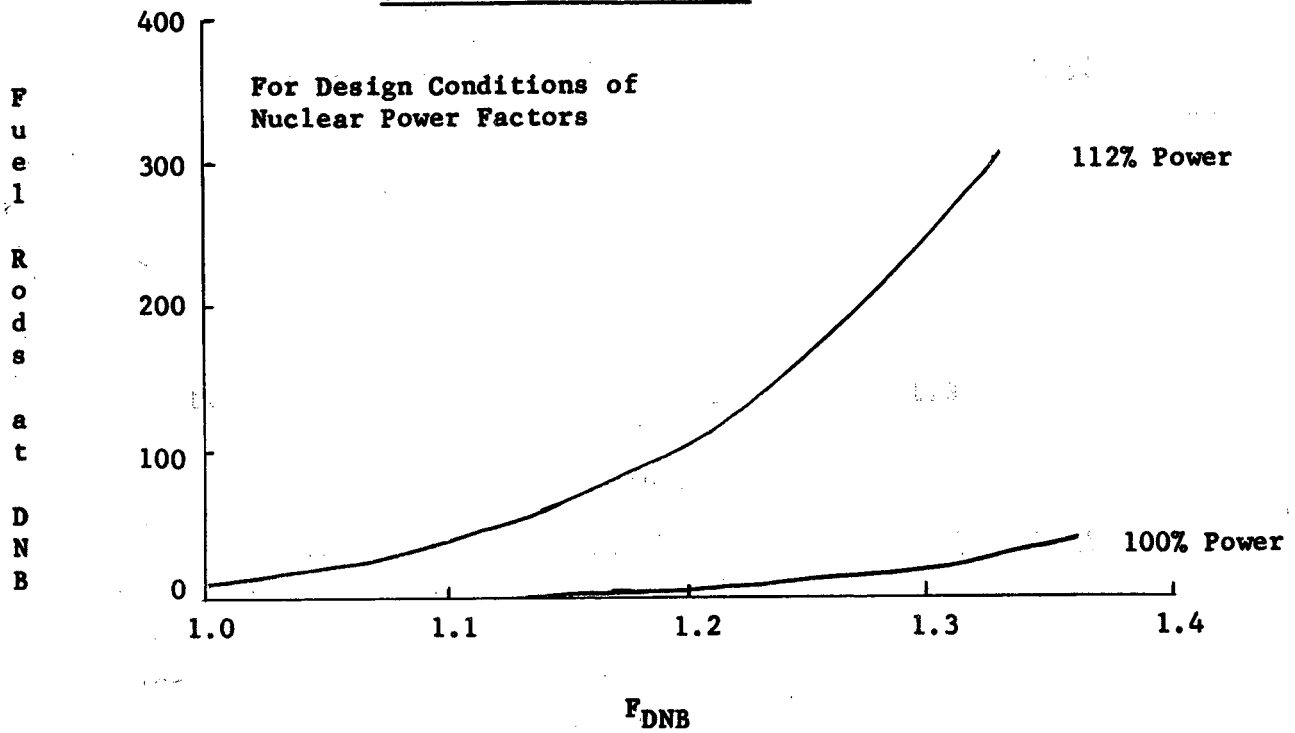
It should be recognized that the design values of nuclear power factors are not the customary hot channel values calculated for the hypothetical hottest rod in the core, but rather are the worst expected nuclear power factors for the entire core, and are required to cover situations such as xenon oscillations. Thus these design factors, although statistically determined as a "worst fit" of the calculation, and therefore conservative when compared to the "best fit" nominal values, are necessary to cover abnormal situations.

Figure 1 - Fuel Rod Power Distribution Curve



We have based our evaluation of the thermal and hydraulic design of the core on the overall design condition of the core rather than on hot channel factors. In particular, we have calculated, using the statistical W-3 DNB correlation and the power distribution curve of Figure 1, the number of fuel rods that would experience DNB at the applicant's assumed overpower condition (112%). In addition, we have performed an "uncertainty" analysis of the DNB situation by arbitrarily assuming certain percentage errors in the DNB calculation. These errors, indicated by the factor F_{DNB}, can arise in several ways as we will subsequently indicate. Figure 2 presents the results of the analysis. A 10% uncertainty in the calculation of DNB results in an increase in the number of failed fuel rods from 10, at 112% power, to 40. Similarly, 33% uncertainty in the calculation results in 300 failed rods.

Figure 2 - Number of Fuel Rods at DNB as a Function of DNB Calculation Uncertainty



The combination of errors which can lead to various values of the F_{DNB} factor are indicated in Table 2.

Table 2 - Values of the F_{DNB} Factor

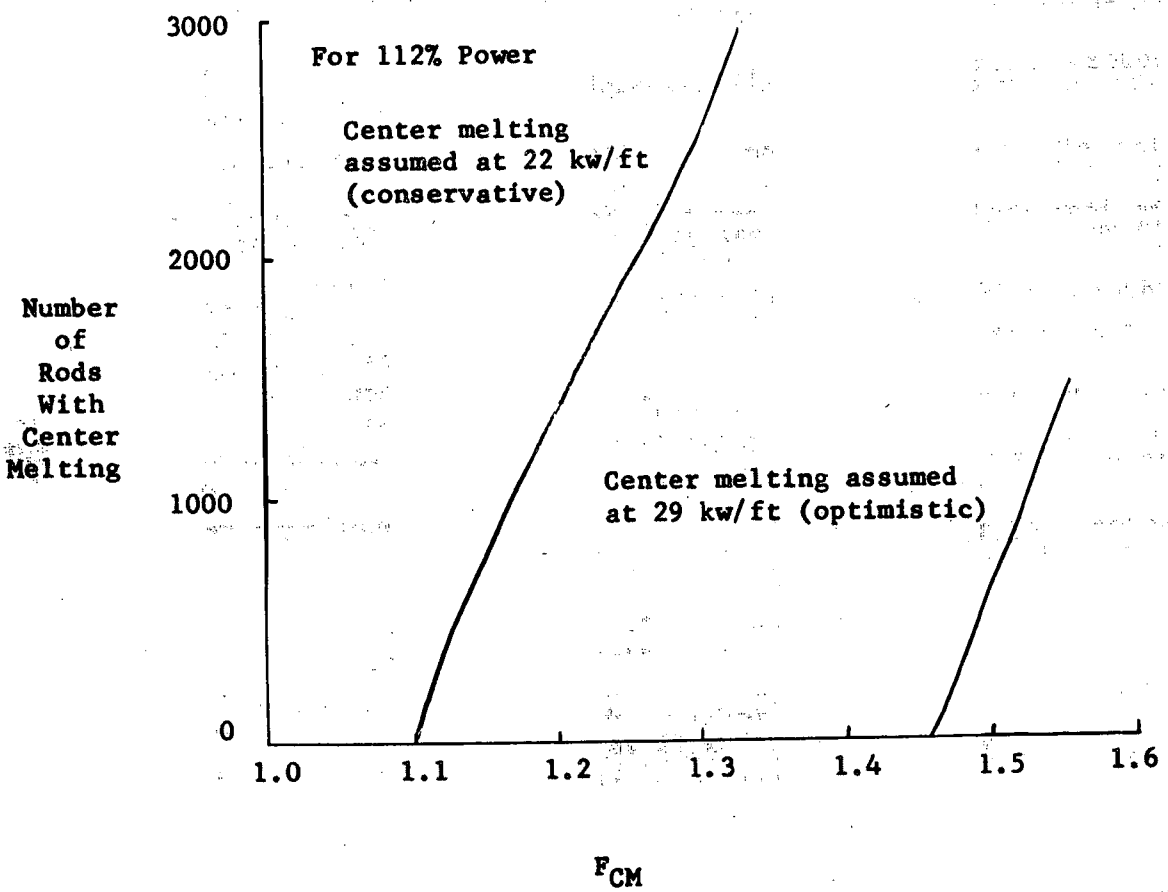
Uncertainty	F_{DNB}
Flow Error - 5%	1.05
10%	1.1
Radial Power Factor - 5%	1.1
10%	1.2
Axial Power Factor - 5%	1.05
10%	1.1
Flow Error & Radial Power Factor - 5% Each	1.15
- 10% Each	1.33
DNB Correlation - 10%	1.1
20%	1.2

An evaluation of the safety significance of these results must be based on (1) the probability of occurrence of the combination of the design nuclear power peaking factors with the maximum expected overpower conditions, and (2) more importantly on the shape of the uncertainty curve of Figure 2. This figure shows that a rapid rise in number of failed fuel rods would not occur for the 112% overpower condition until uncertainties of 20 to 30 percent are assumed. For the 100% design power condition, on the order of 20 fuel rods would be expected to fail for these large uncertainties. Furthermore, an important safety consideration is the consequence of DNB type fuel failure on the system in general. Failure of fuel rods due to burnout in all likelihood would be local in nature, would release fission products to the primary coolant, but would not result in any massive or incipient failures in the primary system. The staff has concluded, pending verification of our calculations by the applicant, that arguments based on the preceeding discussion show that the core thermal and hydraulic design does not lead to an unacceptable amount of fuel failures under assumed maximum core conditions.

We have also calculated, for the same considerations discussed in the DNB analysis, the number of fuel rods that would have center melting at 112% power with various assumed uncertainty factors in the calculation. Results, presented in Figure 3 for the 112% power condition, show a relatively rapid rise in the number of fuel rods which would experience center melting as the uncertainty factor approaches 30% using the conservative experimental data. The main contribution to the F_{CM} factor would be from uncertainties in the nuclear power factor (almost directly proportional). For the 100% design power condition the value of F_{CM} in Figure 3 should be increased by 12% to obtain similar center melting conditions. These results do not indicate the fraction of a given fuel rod with center melting. In general, melting would be limited to the central region of the rod with the vertical length of melting (perhaps

1 to 3 feet for some of the rods) dependent on the axial power distribution and the magnitude of the F_{CM} factor. Considering the conservatism of the data which has to be used to show substantial melting, and the unlikelihood of large errors in power factors, we have concluded that the probability of substantial center fuel melting is small and acceptable.

Figure 3 - Center Melting as a Function of Calculational Uncertainty



The foregoing technique of uncertainty analysis can also be extended to determine the number of channels in the core that could have bulk boiling under various conditions. We are presently attempting to evaluate this situation.

B. Containment Design Requirements

As indicated in Report No. 1 to the Committee, the staff has had difficulty in determining or agreeing with the various models used in evaluating the maximum containment pressure resulting from an assumed double-ended rupture of the largest primary coolant pipe. Our difficulty stems primarily from a lack of knowledge of the conservatism present in the applicant's model of core meltdown and metal-water reactions. For instance, if we consider only the possibility of a xenon oscillation with the variety of corresponding power distributions, it appears that one could postulate a core meltdown which would be more rapid and of larger magnitude than the one the applicant has calculated for his design condition. As discussed in Report No. 1, we are developing an approach which would base our evaluation on overall energy balance considerations, which we believe will allow a better perspective with which to judge the maximum containment pressure that could arise and the need, effect, and margin of safety provided by engineering safeguards.

Figures 4a and 4b are compilations of results given by the applicant as part of answer No. 4 of Supplement 1. Although the maximum pressure shown in Figure 4a appears to be too low by approximately 10-15 psi, we believe a figure such as Figure 4a can be used to "bracket" the worst (no safeguards) and best (all safeguards) values of containment pressure. Figure 4b shows the situation for sudden hydrogen burning of 65% of the hydrogen available from a 100% metal-water reaction with no other energy addition (for computer purposes the applicant assumes a 10 second burning period instead of a step addition of energy).

Figure 4a - Containment Pressure Transients

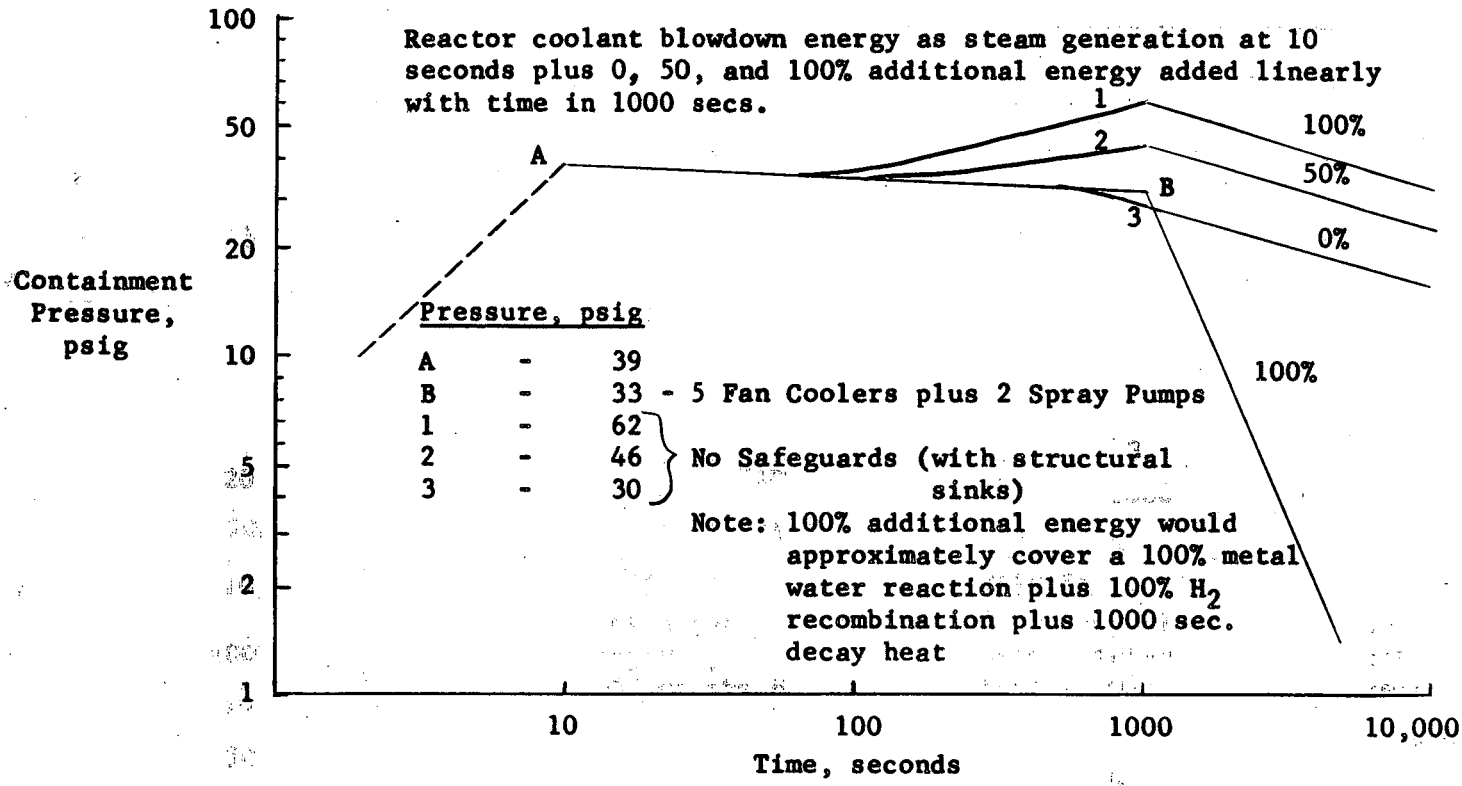
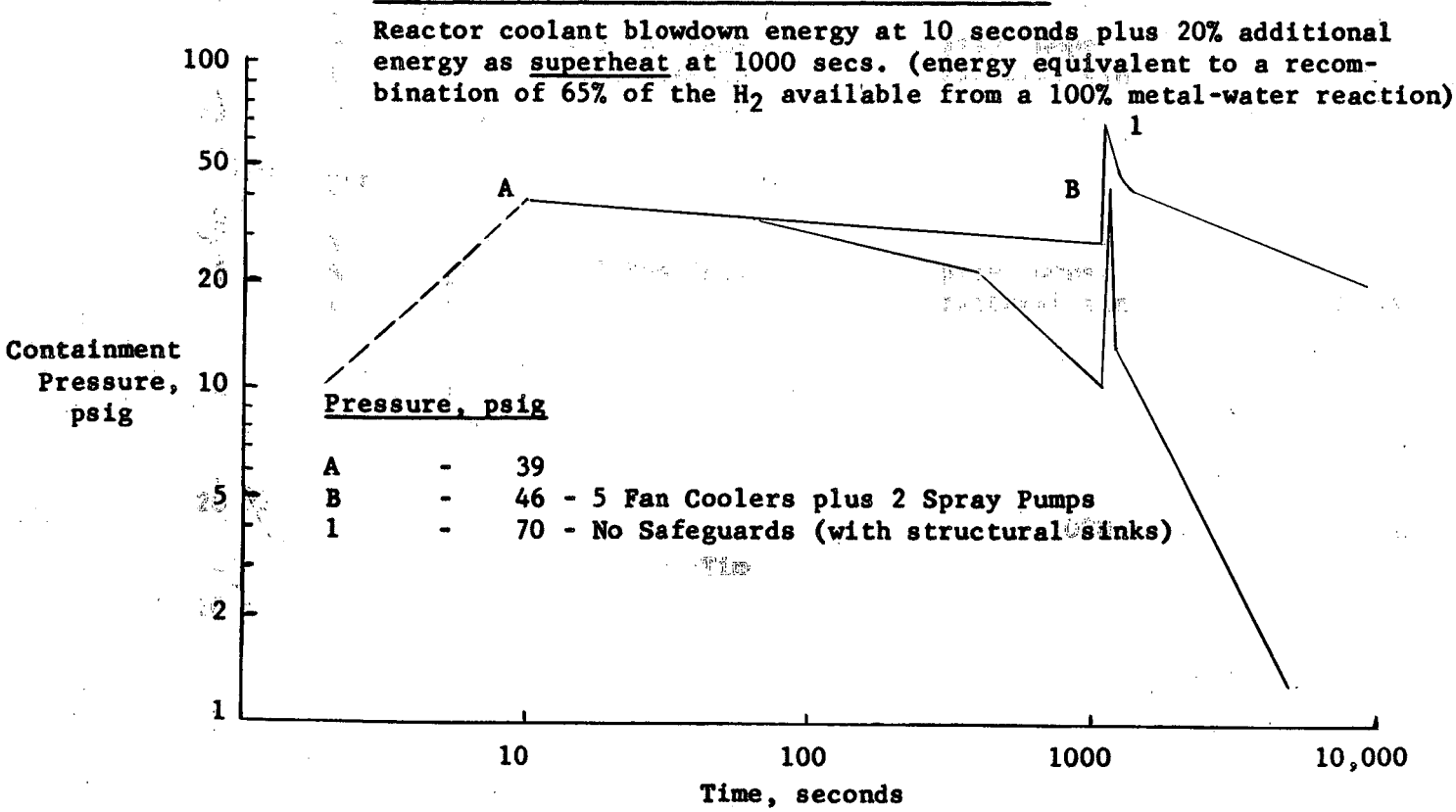


Figure 4b - Containment Pressure Transients



Variations in the specific pressure results shown can occur due to several assumptions in the calculation. Table 3a attempts to indicate the effect of some pertinent assumptions on the final peak pressure result. The values shown are only approximate and vary somewhat for specific cases. One can also attempt to summarize the containment pressure picture by determining coefficients which indicate the expected pressure change associated with a perturbation of a given parameter. Table 3b is an attempt to show approximate coefficients. These coefficients are definitely not constant and should only be used as a guide for estimating peak pressure results.

Table 3

Uncertainties and Coefficients Related to Peak Containment Pressures

a. Uncertainties in Peak Containment Pressure - variations should be used to increase or decrease final pressure

Uncertainty	Pressure Variation
Heat removal by structural sinks	+ 10 psi per 1000 sec for no sinks
Change in total time for a given metal-water reaction (with structural sinks)	+ 5 psi per 500 secs. decrease - 5 psi per 500 secs. increase
Heat sinks during 10 second hydrogen burning (Figure 4b)	+ 5 psi for no sinks

b. Pressure Coefficients - coefficients should be used to find the decrease or increase beyond the original blowdown pressure

Parameter	Pressure Coefficient
Metal-water reaction (no H_2 burning)	+ 2.0 psi per 10% reaction
Hydrogen energy as steam generation	+ 1.5 psi per 10% reaction
Hydrogen energy as superheat	+ 7 psi per 10% reaction
Structural sinks	- 1 psi per 100 secs
Decay heat	+ 1 psi per 100 secs
Fan Cooler	- 3 psi per fan per 1000 secs
Spray pump	- 7.5 psi per pump per 1000 secs

In general, we are attempting to present some limits on the peak containment pressures resulting from use of the available energy sources. We believe Table 3b can be used to obtain the pressure for many combinations of time and extent of reactions. For example, a 30% metal-water reaction in 1000 seconds will produce the following peak pressures with no safeguards.

Blowdown pressure		+39 psig
*30% metal-water reaction (3 x 2)		+ 6
30% hydrogen recombination as superheat (3 x 7)		+21
Structural sinks		-10
*Decay heat		+10
	Total	66 psig

*The applicant's model usually keeps this energy stored in the core.

Fans and pumps would reduce this total pressure as follows:

Total Pressure		66 psig
Five fan coolers (5 x 3)		-15
Two spray pumps (2 x 7.5)		-15
	Total	36 psig

It is significant to note that the hydrogen recombination energy going into superheating the containment atmosphere is a principal pressure source. This can be noted by the following pressure equivalents for all the available energy sources.

Metal-water reaction (100%)	20 psi
Hydrogen recombination if used for	
a. steam generation	15
b. superheating	70
Decay heat	10
Structural sinks	-10

It is apparent that if one assumes significant amounts of metal-water reactions accompanied by hydrogen recombination in the form of superheating of the containment atmosphere, the containment pressure will significantly exceed the design pressure.

Evaluation of the safety implications of these results requires judgement as to the probability of some combination of worst cases occurring and judgement as to what maximum structural stresses one should permit taking into consideration the probability

of occurrence. For example, the containment is designed for a reference incident pressure of 47 psig, however, the containment stresses will not exceed 95% of yield for accident pressures up to 70 psig (1.5 times design) with no earthquake.

We are still evaluating the adequacy of the containment design using this approach and are not yet ready to reach the required judgements discussed above.

C. Reactivity Accidents

As stated in Report No. 1, the applicant's contention that no reactivity accident considered results in a potential safety hazard does not indicate how close their calculated situation is to a severe accident condition. We asked the applicant several questions (see question 3L and 11 of Supplement 1) which would hopefully lead us to some conclusions regarding this, but the answers were unsatisfactory in this respect. We will attempt to clarify our intentions with the applicant. In particular, we believe the applicant should evaluate and clearly discuss:

(1) the type of damage which could result from energy releases within the reactor vessel (i.e., 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,

(4) the possible mechanisms for inserting the reactivity.

In line with this, we would expect the applicant to explain the shutdown mechanisms involved in these accidents. This should include some sort of parametric study to account for uncertainties in the magnitude of these mechanisms and in their application to the analysis. We intend to work with the applicant to develop an approach that will enable us to continue our evaluation of reactivity accidents for this type of plant.

D. Special Consultants

In view of the siting problems associated with the size and location of this reactor, two special consultants have been engaged by the staff. The opinions of these consultants were not available at the time this report was prepared; however, they are expected to attend the full committee meeting in May. The consultants have been engaged in the following areas:

1. Iodine Removal

Mr. George Parker of the Oak Ridge National Laboratory has been engaged to review the internal air filtration system and the containment spray system. The applicant has assumed a filter efficiency of 90% for elemental iodine and 70% for organic iodine. As indicated in the staff's report for the April ACRS meeting, a minimum filter efficiency of 45% is required to meet the siting criteria recommended in 10 CFR 100 (assuming a containment leakage rate of 0.1% per day). Mr. Parker will review the systems (see questions No. 6 and 8) to determine if the above filter efficiencies can reasonably be realized in the post-MCA containment environment.

2. Pressure Vessel Failure

During the ACRS subcommittee meeting on March 30, 1966, the applicant indicated that if simultaneous failure of all pressure vessel head bolts were postulated, the ejected vessel head would penetrate the containment liner but would probably remain within the containment building. Heretofore, it had been at least tacitly assumed that the pressure vessel head would breach the containment vessel. Since it now appears that this mode of pressure vessel failure may not breach the containment vessel, the staff has engaged Mr. James Proctor of the Naval Ordinance Laboratory to make detailed calculations for this postulated failure mode. We expect Mr. Proctor's evaluation to show the degree of conservatism provided by the applicant's analysis, and to indicate if protection against this mode of failure is feasible.

Conclusion

We believe that the preliminary analyses performed thus far for the thermal and hydraulic design are satisfactory, but that additional information should be developed concerning reactivity effects. We are still evaluating the adequacy of the containment design. The staff expects to be in a position to discuss the overall acceptability of the applicant's proposal for the June ACRS meeting.