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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. 3

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 July, 1966 meeting.

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Introduction

The proposed Indian Point Unit No. 2 pressurized water reactor was previously considered by the Committee at its April and May meetings and subcommittee meetings were held on March 30, May 3, and June 23, 1966. The Staff has had meetings with the applicant on January 17-18, May 2, May 19, and May 26, 1966.

Consolidated Edison Company of New York, Inc. applied for a construction permit on December 6, 1965, and submitted the Preliminary Safety Analysis Report (PSAR) at that time. The First and Second Supplements to its application were filed on March 31 and May 31, 1966, to provide additional information requested by the Staff by letters dated February 28 and May 11, 1966, respectively. The Third Supplement was filed on June 20, 1966, to provide additional structural design information related to the containment as requested by Dr. Nathan Newmark during the May 19, 1966, meeting with the Staff.

As part of the overall safety review for the Indian Point II station, the advice of the following consultants has been requested:

1) Geological Survey; U. S. Department of the Interior,
2) Fish and Wildlife Service; U. S. Department of the Interior,
3) U. S. Weather Bureau; U. S. Department of Commerce,
4) U. S. Coast & Geodetic Survey; U. S. Department of Commerce,
5) Dr. Nathan M. Newmark (Seismic Design),
6) Dr. George Parker; Oak Ridge National Laboratory, and
7) Mr. James Proctor; Naval Ordnance Laboratory

The reports of those consultants from whom reports have been requested will be made available to the Committee upon receipt. We anticipate no written reports by Dr. George Parker and Mr. James Proctor.

Discussion

The 2758 MWt Indian Point II facility is the largest reactor that has been considered for licensing. In addition to its high power level, the licensing of this facility requires special consideration in that it will be located in a region with the highest population density considered to date. The applicant has analyzed the adequacy of the proposed design features of this facility and the suitability of the site for a maximum reactor power level of 2758 thermal megawatts. As is discussed in this report, the Staff believes that the proposed design and site are suitable at this power level. The estimated gross electrical output from the turbine under these conditions is 916 electrical megawatts. Although the turbine has an additional calculated gross capacity of about 10% (1021 MWe), the applicant has orally stated that there is no planned stretch in Indian Point II.

The applicant is aware of the special siting considerations involved in this facility and has proposed a containment and engineered safeguards system which we believe to be superior to that provided at facilities in less populated areas. These special features are listed below and discussed further in this report:

a) The containment is designed to have negligible leakage under the postulated maximum credible accident conditions. The penetration pressurization system, pressurization of channels along containment liner welds, and isolation valve

seal water systems have been provided to achieve this objective.

b) Even though negligible leakage is anticipated, two independent means of iodine removal within the containment have been provided. These are an air filtration system using activated charcoal filters and a containment spray system which uses sodium thiosulfate in the spray water as a reagent to aid removal of elemental forms of iodine.

c) The recirculating water system that is required to provide long term cooling of the core is located inside the containment. This arrangement is provided so that it will not be necessary to pump radioactive liquids outside the containment after an MCA unless both internal pumps fail.

d) Three diesel generators are proposed for on-site power. Two are required for operation of the minimum safeguards required to preclude containment overpressurization and core meltdown after an MCA.

The general design of this reactor is similar to the San Onofre, Connecticut-Yankee, and Brookwood facilities. The Staff review approach, in addition to demonstrating conformance with the General Design Criteria, has been directed toward determining the safety margin that exists in those areas that are most significant to the safety of this facility. These are: a) thermal and hydraulic design of the reactor core, b) design of the containment and engineered safeguards, and c) reactivity effects. The Staff indicated in Report No. 2 to the ACRS that the thermal and hydraulic design of the core were satisfactory. This report contains further consideration of the adequacy of the containment, and potential reactivity effects

(see special discussions in Appendices A and B and related criteria 7 and 17).

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 with an anticipated growth to 30,000 by 1985.

The cumulative population in the vicinity of the site is presented in the following table.

	Indian Point	
Distance	1960	1980
0.5 mile	46	100
1	1,080	2,100
2	10,810	20,910
3	29,630	59,520
4	38,730	78,800
5	53,040	108,060
10	155,510	312,640
15	326,930	670,210

The above table demonstrates the relatively high population in the vicinity of the proposed site. 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). For this reason, we have elected to evaluate potential off-site doses at 0.32 and 0.67 miles, which are assumed by the applicant for exclusion distance and low population distance, respectively. However, 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 the future, it appears that a relationship between population distribution, facility design, and site parameters should be developed to enable more meaningful comparative evaluations as other facilities are proposed for the more densely populated sites.

We have received a report on the meteorological aspects of the site from the Weather Bureau. In summary, the Bureau concluded that the atmospheric dispersion factors developed by the applicant during operation of Unit No. 1 were conservative. These parameters were assumed for the dose calculations presented in the "Accident Analysis" section of this report.

With reference to hydrology, geology, and seismology, the reports by the consultants in these fields have not yet been received. Informal communications with the respective consultants have indicated there are no problem areas that are not already adequately covered in the PSAR.

B. Conformance of Indian Point Unit No. II to Staff's General Criteria

We have evaluated the Indian Point II facility to determine if the "General Criteria for Nuclear Power Plants" as published by the Commission on November 22, 1965, have been satisfied. Conformance with Criteria 2, 4, 5, 8, 9, 10, 11, 12, 13, 14, 20, 22, 23, 24, 25, 26 & 27 is essentially similar to the recently reviewed

Brookwood facility and are discussed in our report to the ACRS dated February 21, 1966. Accordingly, only those criteria that are significantly different, or for which additional information has been provided, are considered in this report.

Criterion 1(a)

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for non-nuclear applications may not be adequate.

Three barriers which prevent significant release of fission products from the reactor fuel to the environment are incorporated in the Indian Point II design as discussed below.

1. The fuel element cladding provides the initial barrier and will be designed considering the effect on zircaloy of hydrogen embrittlement, internal fission gas pressure, thermal expansion, and uncertainties in fabrication. To assure fuel of the highest quality, fuel rods will be subjected to chemical analysis, tensile tests, corrosion tests, dimensional inspection, x-ray of welds, ultrasonic tests, and helium leak tests.

2. The primary coolant system will be designed in accordance with applicable codes, i.e.: ASME Boiler and Pressure Vessel Code, Section III, for the pressure vessels, and the ASA Code for Pressure Piping, B 31.1 for the piping. In the case of the reactor vessel, which is being fabricated by Combustion Engineering Co., the applicant has outlined in some detail the quality control procedures, the testing

during fabrication, the acceptance testing, the capability for periodic inspection, and the NDT shift surveillance program. The applicant is retaining the services of United States Testing Company, who will perform independent checks to assure proper quality control during fabrication of the reactor vessel.

The increase in the nil ductility transition (NDT) temperature with fast neutron exposure over the service lifetime of the reactor vessel is one means by which an increase in brittleness and susceptability to failure can occur. The Indian Point II reactor vessel is being designed to withstand a fast neutron exposure of 3.7×10^{19} n/cm^2 without undue operating restrictions. The anticipated fast neutron exposure during the service life of the vessel is estimated to be 0.85×10^{19} n/cm^2, which corresponds to an NDT temperature shift of 160°F. Thus, considerably less than the design fast neutron exposure should be encountered during the service life of the reactor vessel.

The initial NDT will be measured on specimens of the reactor vessel base material. Subsequently, additional specimens of base material will be irradiated in eight specimen capsules located between the active core and the reactor vessel wall. The fast neutron flux at the location of the capsules will be higher than that experienced by the reactor vessel wall. Thus, the specimens will be representative of the reactor vessel at a later time in life. The NDT of these specimens will be measured periodically.

During fabrication of the reactor vessel, radiographic, ultrasonic, magnetic particle, and liquid penetrant examinations of the material will be conducted as

appropriate to assure that the portion of the vessel inspected meets acceptance standards. The reactor vessel head closure studs will also receive comparable inspections both during fabrication and subsequently during refueling when the studs are removed from the vessel.

We believe that the quality control and surveillance programs for the reactor vessel outlined by the applicant are adequate for construction permit purposes. We believe that improvement in methods of in-service inspection of the reactor vessel should be considered in order to develop procedures which parallel as closely as is practicable the inspections given to pressure vessels in non-nuclear applications. The applicant, as a member of the Empire States Atomic Development Associates (ESADA), is exploring possible methods to regularly examine the pressure vessel for defects in body and cladding after installation and service. If a satisfactory procedure is formulated, it will be adopted for this vessel.

3. The containment vessel will be designed and tested to conform to applicable parts of the "Building Code Requirements for Reinforced Concrete" (ACI 318-63). The liner will be reinforced at each penetration according to the rules set forth in the ASME Code, Section VIII UG-36. An important adjunct to the containment vessel is the various engineered safeguards. These, too, will be designed in conformance with applicable portions of the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping.

Based on the foregoing, we believe that Criterion 1(a) is satisfied.

Criterion 1(b)

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated and erected to:

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

The effects of severe environmental conditions at the Indian Point II site have been considered and taken into account in the design of those portions of the facility important to safety. The containment and engineered safeguards systems will be designed to withstand wind loading of 30 psf coincident with the temperature and pressure conditions within the containment vessel associated with a major rupture of the primary coolant system. In addition, all safeguards systems will be protected against expected ice and snow conditions.

The applicant's proposed seismic design criteria, outlined below, are in conformance with a maximum horizontal ground acceleration of 0.1g, which we understand will be the design acceleration recommended by the U. S. Coast & Geodetic Survey for systems and structures important to safety. The seismic design criteria for those components which are necessary for the safe, orderly shutdown of the facility (designated as Class I by the applicant), are:

1. Stresses in all structural members of the containment vessel shall not exceed 0.95 yield under the combined dead load, 47 psig internal pressure (including the temperatures associated with this pressure) and 0.15g horizontal and 0.1g vertical earthquake accelerations acting simultaneously.

To demonstrate that margin exists under even higher than design pressures (and temperatures) the following additional criterion has been stated. With no earthquake or wind acting, the vessel shall be designed to withstand a 70.5 psig accident internal pressure without exceeding 0.95 yield.

2. All other structures and equipment important to safety shall remain functional under the same loading conditions. For those structures or components which are allowed to exceed yield, deformation shall not exceed 0.4%.

We anticipate that the report by our seismic design consultant, Nathan M. Newmark, will confirm that the proposed seismic design considerations are adequate.

Based on the foregoing considerations, we believe that Criterion 1(b) is satisfied.

Criterion 2

Same as discussed in Brookwood report.

Criterion 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

The applicant has stated the criterion that the containment, containment liner, engineered safeguards and components required to maintain containment integrity shall be protected against loss of function due to damage by the following missiles:

a) All valve stems up to and including the largest size to be used.
b) All valves up to and including the largest size to be used.
c) Massive chunks of metal up to 6 inches thick.
d) All valve bonnets.
e) All instrument thinbles.
f) Various type and sizes of nuts and bolts.
g) Pieces of pipe up to 10-inch diameter striking broadside or end on.

h) Complete control rod drive mechanisms.
i) Reactor vessel head bolts.

We believe that the foregoing satisfies Criterion 3.

It should be noted that Criterion 3 refers to potential missiles generated within the containment building. Containment damage might also result from external missiles such as those generated by massive failure of the turbine generator. The principal potential missiles that could be generated under these conditions would be parts of the turbine rotor and the turbine blades. The turbine blades would most likely be contained by the turbine housing and their failure would not appear to present a serious problem. However, containment integrity might be jeopardized from impact by massive parts of the turbine rotor. In Indian Point II, a plane normal to the turbine shaft will pass through the containment circle for about 1/2 of the length of the shaft starting at the high pressure end of the turbine. The staff is continuing its overall evaluation of this potential problem.

In addition, the applicant has calculated the potential consequences of various postulated modes of pressure vessel failure even though in consideration of the materials of construction, the care in fabrication, and the anticipated conservative mode of operation (see answer to question 3, First Supplement), it contends that catastrophic failure of the pressure vessel is not credible. Accordingly, no special considerations have been provided in the Indian Point II containment design to mitigate the consequences of postulated missiles generated by failure of the pressure vessel.

The following results were calculated by Westinghouse for various assumed pressure vessel failure modes:

a) Circumferential failure at pressure vessel midplane - This failure would cause ejection of the top portion of the pressure vessel, including the core and core support structures, to a height of about 20 feet. The clearance above the top of the vessel to the dome of the containment is 160 feet; thus, this postulated mode of pressure vessel failure would not initiate failure of the containment liner.

b) Circumferential failure above the nozzles but below the core support flange - This failure mode would cause ejection of the vessel head, the core, and core support structure. This structure would hit the control rod missile shield (100 tons) and also the overhead polar crane (300 tons). This combined mass, assuming inelastic collisions and vertical rise, would raise the polar crane 30-39 feet leaving a distance of 16-25 feet vertical rise to the point where the crane projection would touch the containment. If vertical rise is not assumed and the missiles glance off the crane at some angle, the containment liner would probably be violated.

c) Pressure vessel head failure - If simultaneous failure of all 54 pressure vessel bolts were assumed, the head would rise to a height of 550 feet if unrestrained by the missile shield, polar crane or containment structure. The pressure vessel head would be restrained, however, by the missile shield and polar crane, but penetration of the containment liner would probably still result.

d) Longitudinal failure of pressure vessel - This mode of pressure vessel failure would result in displacement of the vessel against the surrounding concrete structure which would restrain the vessel since the annular clearance to the concrete is about 0.5 foot. No massive missiles would be generated such as in the above cases.

The Staff has employed Mr. James Proctor of the Naval Ordnance Laboratory to make an independent analysis of the above assumed modes of pressure vessel failure. Mr. Proctor has performed independent calculations of case b) above, and has reviewed the applicant's calculational model for all cases. He believes the applicant's model is conservative and agrees with the results.

Criteria 4 and 5

Same as discussed in Brookwood report.

Criterion 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures.

Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over the core lifetime.

The applicant has stated that the following criteria will be met during anticipated operational modes including the overpower conditions, assuming the worst combination of instrument errors, at any time during core life:

1. The minimum departure from nucleate boiling ratio (DNBR) as determined by the W-3 correlation will be greater than or equal to 1.3.
2. The maximum fuel center temperature will be below the melting point of UO_2 using the WAPD design curve for thermal condictivity of UO_2.
3. Stresses in the zirconium clad will be less than the yield strength.

Although these criteria fulfill the requirements of Criterion 6, we have extended our evaluation to include an assessment of the safety margin available before large numbers of fuel rods exceed design limitations. As presented in Report No. 2

(May ACRS Report), we have concluded that large amounts of failed fuel rods would not occur for the overpower condition even though large uncertainties (up to 30%) are assumed in the calculations. We also indicated in Report No. 2 that the probability of substantial center fuel melting was small and acceptable.

We believe that Criterion 6 is satisfied.

Criterion 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

The largest reactivity insertion rate due to the withdrawal of control rods at maximum speed is given by the applicant as 3.5×10^{-4} k/sec. The consequences of control rod withdrawal at this rate will be limited by various redundant preset trips to prevent fuel damage.

Although the applicant considers the rod ejection accident incredible since each control rod housing will be proof tested to 6300 psi prior to operation, they have stipulated that automatic and administrative controls will be provided to limit the amount of reactivity which could be inserted as a result of this accident.

The staff has evaluated a rod ejection analysis as discussed in the Second Supplement. Our evaluation of this accident is given in Appendix A of this report. Based on the data presented in the study, we believe that Criterion 7 can be met by suitable limitation of control rod worth and, if necessary, moderator coefficient limitations. We intend to continue to work with the applicant to evaluate any operating limitations required.

Criterion 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no components or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

In common with the Brookwood facility, all nuclear and process system parameters capable of scramming the reactor will be monitored by redundant instrumentation. Amplifiers associated with such instruments drive relays, the contacts of which are connected through three complete and independent logic chains which open the two redundant scram breakers. Two of these chains are energized during a scram condition and respectively actuate the shunt trip coils of the two breakers. The third chain is de-energized during scram and is connected to the two undervoltage coils (wired in parallel) of the breakers. Further, two contacts on each breaker interrupt both sides of the DC lines feeding the rod mechanisms. Thus, each logic chain is independently capable of scramming the reactor.

The proposed protection system is redundant and immune to individual faults occurring at the instrumentation, logic circuits and scram breakers. Fail-safe, in terms of partial or complete loss of electric power, is inherent in the proposed design.

The manual scram switch has three contacts, two of which respectively apply voltage directly to the shunt trip coils. The third contact interrupts the

undervoltage coils. Thus, no single circuit fault can disable the manual scram function.

Criteria regarding the bypassing of scram channels for maintenance, etc., have not been stated. However, the Brookwood design (assumed universally applicable) requires that such bypassing place the channel in a tripped mode.

Coincidence as well as redundancy is used throughout the protection system. Thus, there is capability for testing, at least up to the logic chains. There are no provisions for testing the logic chains and breakers at power. However, we believe that this matter can be satisfactorily resolved with the applicant.

The containment isolation signal is derived from three pressure sensors (2/3 logic). The circuitry following these sensors will be fail-safe with respect to voltage and/or instrument air loss. At this writing we understand that revisions are being made to the applicant's criteria which will require that, in some cases, isolation be accomplished by two automatic valves. We also understand that the circuits actuating such valves will be independent and immune to single failures.

Safety Injection signals will be derived from a coincidence of pressurizer-low-level (2/3 logic) and pressurizer-low-pressure signals. The associated circuits are redundant and immune to single failures. The sensor circuits fail safely, i.e. they call for safety injection, in the event the instruments are carried away by an accident. Voltage loss at certain of the circuits will preclude, rather than initiate, injection. This is a design feature intended to prevent inadvertent injection. However, manual actuation capability will be provided.

The design features described above are similar to the Brookwood facility.

In addition, the applicant has stated: "The principal criterion of control station design and layout is that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the control room." It is not clear at this time which displays and alarms can actually fulfill this criterion. This matter is under continuing study by the Division of Reactor Licensing and we expect to be guided in this study by Standards Group WG.3 of the IEEE. We believe that the control station design and layout can be made to conform to the applicant's criterion which is acceptable.

We have concluded that the applicant's criteria are in accord with Criterion 15, and that the proposed instrumentation and circuitry can be made to conform to this criterion.

Criterion 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can interfere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, and fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

Similar to the Brookwood facility, complete loss of A.C. voltage will initiate reactor scram through circuits within the instrumentation which trip when their respective channels are de-energized. Loss of D.C. voltage de-energizes the rod coils directly.

A voltage loss at the logic circuits feeding the scram breakers will scram the reactor via the undervoltage coils. The containment isolation circuits downstream of the pressure sensors will fail safely in the event of voltage loss. Loss of

instrument air at containment isolation valves will, in most cases, drive them to their "close" position. (Exceptions are valves which must remain open temporarily under accident conditions).

Not discussed in our Brookwood report, but of universal concern are the potential adverse effects of fires originating in the control and safety system wiring and/or within the control room itself. It has been the staff's position that a direct, analytical safety analysis relating to the possibility of reactivity excursions resulting from such fires is, in practice, impossible due to the random nature of fire damage and the nearly infinite variety of possible circuit faults (some "unsafe," some "safe") which could result. Rather, it has been our position that the natural complexity of reactor control and safety systems coupled with an overall design firmly based on applicable criteria, accepted codes, etc., constitutes the best, if not only, defense against serious fire-induced accidents.

In this regard, a literature search was conducted with the assistance of the computer facilities at the Nuclear Safety Information Center (NSIC) at Oak Ridge National Laboratory, to study the historical record of such excursions. NSIC has informed us that they were unable to find any records of incidents involving reactor damage as a result of fire-induced excursions.

We believe that Criterion 16 is satisfied.

Criterion 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

The Indian Point II containment vessel is similar to that of the Connecticut-Yankee vessel in that it is a totally reinforced concrete vessel with cylindrical walls, a flat base (with sump pit), and a hemispherical dome. The cylindrical walls are 4.5 feet thick below grade and taper to 3.5 feet thick where the dome joins the wall. The dome is also 3.5 feet thick. The free volume of the containment vessel is 2.6 million cubic feet, and the design pressure of the vessel is 47 psig.

The containment design criteria relative to material stresses are expressed in terms of load factors above design loadings and are discussed in Criterion 1(b). We believe that this approach is reasonable and provides assurance that the containment will be designed considering the pressures and temperature associated with the MCA acting simultaneously with the maximum earthquake or wind loading. Of particular importance in assuring a leak-tight vapor container is the method of securing the liner so that excessive stresses will not cause increased leakage. All portions of the liner and especially those in the vicinity of penetrations are to be carefully designed to consider the effects of all temperature, pressure and

earthquake loads. We anticipate that our consultants, Drs. N. M. Newmark and W. J. Hall, will discuss this problem in their report. In other respects, we believe the proposed criteria and concept of the containment are acceptable.

In the Indian Point II design, the recirculation loop of the engineered safeguards system to be used following an MCA is located entirely within the containment vessel (Figure 2-1, Second Supplement). Operation of this loop is required after all of the borated water from the refueling water storage tank has been pumped into the containment vessel as containment spray or core injection water. This water will collect in the pit below the reactor vessel and small sumps near the edge of the containment. It will be pumped by recirculation pumps (we understand that there will be two recirculation pumps even though four are shown in Figure 2-1) through the residual heat exchangers located within the containment. After cooling, the borated water can be directed back into the reactor vessel to remove core decay heat or be sprayed into the containment vessel to effect pressure reduction as necessary. Although provision is made in the design to circulate sump water outside the containment, such action would be required only if, (1) a small leak has occurred in the primary system and water must be recirculated through the high head safety injection pumps after the refueling water storage tank is emptied (in this case, the fission product inventory of the spilled coolant will be low since significant fuel failures should not occur), or (2) the redundant components of the internal recirculation loop have failed.

In order to determine the adequacy of the containment structure and associated engineered safeguards to accommodate the pressures associated with the largest

credible energy release, the applicant presented studies of the containment pressure after an assumed loss of coolant accident. These studies considered most available energy sources and a variety of situations including cases where: (1) all components of the safety injection system, containment spray, and fan-coolers operate, (2) no engineering safeguards operate, (3) safeguards are driven by two of the three emergency diesel generators, (4) delayed initiation of safety injection with and without some engineering safeguards is assumed, and (5) delayed hydrogen burning is assumed. The calculated metal-water reactions ranged from 1% for case (1) to 43% in 2300 seconds for case (2). Because of the difficulty in agreeing with the various assumptions used in the applicant's analysis, the Staff has developed what we believe is a simplified approach for evaluating the adequacy of the containment pressure design. The approach is presented in some detail in Appendix B of this report. It is based primarily on determining the capability of the containment to accept arbitrary energy releases including an assumed 75% metal-water reaction occurring at a linear rate within 2000 seconds.

Our major conclusions regarding the containment are:

1. With one safeguard system operating (all 5 cooling-fans or both containment spray pumps), the containment can tolerate the assumed metal-water reaction if hydrogen is burned as produced.
2. With two safeguard systems operating (all 5 cooling-fans and both containment spray pumps), the containment can tolerate the assumed metal-water reaction if delayed hydrogen burning occurs.

Because of mixing and stoichiometric requirements, the Staff believes that

complete delayed hydrogen burning is highly improbable. In addition, it seems reasonable to assume that the safety injection system can be relied upon to limit the metal-water reaction below the rate assumed by the Staff for evaluation purposes.

In summary, the Staff believes that the Indian Point II containment and engineered safeguards are sufficient to limit the maximum credible pressure of the containment to the design value of 47 psig following the MCA. It should be noted that although active quenching systems to prevent the postulated metal-water reaction are not required, engineered safeguards which remove heat from the containment must be assumed to operate.

We believe that Criterion 17 is satisfied.

Criterion 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Back-up equipment (e.g. water and power systems) to such engineered safeguards must also be redundant.

Two independent heat removal systems of different principles, each capable in itself of maintaining containment pressure below 47 psig, are provided in the Indian Point II design. These are: (1) the air recirculation system, and (2) the containment spray system. (Another heat removal system, safety injection to the core, is provided to prevent core meltdown and limit the metal-water reaction but is assumed to be inoperable in sizing the containment depressurization system components).

The air recirculation system consists of five motor driven centrifugal fans and cooling coil assemblies which will be provided to recirculate and cool the containment air during normal power operation, during any other time when the containment vessel is closed, and during an accident. Each of the five ventilation units consists of a demister, a cooling coil, a roughing filter, an absolute filter, a fan and a charcoal filter in that order. The charcoal filter is normally by-passed. Under accident conditions each unit will have a capacity of 65,000 cfm. When the filter mode is required, motor operated louvers will automatically direct flow to the charcoal filter. Simultaneous operation of two butterfly valves in the filter bypass lines is also required. In this mode the containment atmosphere is cooled and radioactive halogens are adsorbed on the charcoal.

Normal containment temperature, about 100-120°F, is maintained during full power operation with various combinations of the five air recirculation cooling systems in service. The ventilation system will be designed for continuous operation without interruption during and following the loss of all primary coolant and during all postulated subsequent energy additions to the containment vessel. If electrical power to the site is lost, emergency diesel generators will restore power to these units within 15-30 seconds. The fan motors are designed to operate continuously under accident conditions of about 271°F in a steam-air mixture with a density of 0.175 lb/cu. ft. at 47 psig for 48 hours and for 10 days at 5-10 psig conditions. In addition, each unit could operate under a pressure of 70.5 psig and a temperature of 298°F for one hour. Each of the five ventilation cooling

system units, is capable of removing more than 72,000,000 BTU/hr. Five air recirculation cooling units operating alone would limit the maximum containment pressure following the MCA to less than 47 psig as discussed in Criterion 17.

The ventilation ducts and equipment will be protected from missiles, are located in positions within the containment to achieve good mixing, and are located away from the primary system. The system will be designed to withstand the sudden release of the primary system energy and energy from chemical reactions without failure due to shock or pressure waves. This is accomplished through the use of dampers along the ducts which would open at slight overpressure.

The containment spray system, an independent backup of different principle than the fan cooler units, will be designed to reduce containment pressure and remove iodine from the containment atmosphere to the sump by a washing action. Sodium thiosulfate will be injected into the spray water to improve entrapment of iodine in the water. The heat removal capacity with both spray pumps operating is stated to be at least equivalent to the heat removal capacity of five fan-cooler units after the MCA.

Water will be pumped by two containment spray pumps at the rate of 2600 gpm each from the refueling water storage tank, which holds 320,000 gallons of 3000 ppm borated water, through separate lines to two separate spray headers located within the containment vessel. The spent hot spray water will collect in the containment sumps. If it is necessary to continue containment spray after the borated water supply is exhausted, the water can be drawn from the containment sump by

recirculation pumps located inside the containment and passed through the residual heat exchangers which are located inside the containment vessel. After cooling, the water can be directed back through the containment spray headers. The capacity of each residual heat exchanger is sufficient to maintain the containment pressure below 47 psig after the refueling water storage tank is emptied. If multiple component failures occur in the internal recirculation loop, spray can still be effected by the backup residual heat pumps located outside the containment.

The service water system provides cooling water to the air recirculation units and to the component cooling loop which in turn cools the residual heat exchangers in the safety injection system. Six electric-motor-driven centrifugal pumps will take suction directly from the river and discharge in triplicate to two service water headers which supply water to separate lines to the cooling component headers. The service water headers for each safeguard system (i.e., fan-cooler or component cooling heat exchanger) is valved so that half of each system is on each of the two headers. The capacity of any two of the service water pumps will be sufficient to supply the entire requirement for cooling water of the containment air coolers during a major loss of primary coolant accident and recovery. These pumps can be operated using electrical power from any two of the three auxiliary diesel-generators, if required.

Cooling between the recirculation and service water systems is provided by the component cooling system. This is a redundant system and can be supplied with emergency power.

With regard to the long term heat removal requirements and capabilities, we have considered the following, (1) at given times after the accident, what cooling equipment must operate, and (2) at given times after the accident, how much time is available before design pressure is exceeded if this cooling equipment becomes inoperable. Assuming the heat removal capability of a fan-cooler is approximately 20,000 BTU/sec at 47 psig, one cooler's heat removal capability is equal to the decay heat generation at 16 hours after the accident, two cooler's at 2 hours and three cooler's at 1/2 hour after the accident. Thus, within one day following the accident only a single fan-cooler must operate to balance heat transfer into the containment vapor phase. If safety injection and the associated recirculation system is assumed to function, this system will adequately remove all core decay heat, and operation of the fan-coolers or containment spray would not be required. The applicant has determined that it would require approximately 400 days following the MCA before the containment structure is capable of transferring the decay heat produced to the environment without benefit of active safeguards to provide cooling.

If one assumes all cooling equipment becomes inoperable, the time available before exceeding the design pressure depends not only on the time after the accident but on the containment pressure at the time of loss of the equipment. If, for example, a base pressure of 10 psig in the containment is used, it takes slightly over one hour after the accident. The time available for action increases to approximately 20 hours at 120 days after the accident. The following table

summarizes the allowable time for outage of all heat removal equipment as a function of time after the MCA.

Long Term Heat Removal Requirements

Time After Accident at Which Cooling Equipment Becomes Inoperable, Days	Time Required to Reach Containment Design Pressure, Hours		
	Starting Point, psig		
	0	10	30
1	6	4	1
10	12	8	2
30	18	12	3

Based on the foregoing considerations, we believe the safeguard systems provided are of sufficient redundancy and capacity to assure containment integrity under all credible circumstances, and that Criterion 18 is satisfied.

Criterion 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

The Indian Point II containment has been designed to have essentially zero leakage under accident conditions. The achievement of this design objective would result in negligible doses at the site boundary for the postulated maximum credible accident. Since most containment leakage paths occur at the penetrations, a penetration pressurization system will be installed to preclude leakage of the containment atmosphere at these locations. This system provides a pressurized zone at each penetration liner weld that is maintained at a pressure (about 50 psig) slightly above the containment design pressure. Pressurized zones are also provided for the containment liner seam welds, the two ventilation-purge duct penetrations, the personnel air locks, the equipment door flange and the spent fuel transfer tube. A penetration pressurization system in which the penetrations are continuously pressurized is a feature that has not been previously proposed for use in other licensed facilities.

The penetration pressurization system is divided into four sub-systems that are provided with two independent sources of pressurized gas to assure that the pressurized zones can be maintained. The four sub-systems are normally connected to instrument air. Two compressors are used although only one is required to maintain pressurization at the maximum allowable leakage rate of the pressurization system. Each sub-system contains an air receiver that can maintain pressurization for four hours if both air compressors should fail. The backup pressurization source for each sub-system is nitrogen bottles that provide a minimum supply of gas for 24 hours at the maximum allowable leakage rate.

The gas makeup rate for each sub-system is continuously monitored and recorded (with a high rate alarm setting) in the control room to assure that the leakage for the system is within specified limits. A tentative upper limit for long-term uncorrected air consumption has been set at 0.2% of the containment volume per day (sum of four headers). This limit has been set on the assumption that half of the leakage would be into the containment and half would be out of the containment. In addition, all penetrations that are outside the containment and in accessible areas will have a locally mounted pressure gage. The pressurized zones that are entirely within the containment (e.g., each containment liner seam weld channel) or in inaccessible areas will be provided with low pressure alarms and individual indicating lights in the control room. Isolation valves are also provided for each pressurized zone to enable further identification of leaking penetrations.

An isolation valve seal water system, similar to the system proposed for the Malibu facility, will also be provided to preclude potential leakage paths through piping systems that penetrate the containment liner. These pipes could present a possible source of leakage if radioactive gas were to leak through the isolation valves provided for each line. This system is designed to provide a high pressure water seal at the outer isolation valve so the pressure at the valve is maintained above the containment accident pressure. This design feature should eliminate this potential source of leakage. This system is not provided for closed piping systems inside the containment that do not connect to the containment atmosphere or a source of radioactivity, and which are provided with missile protection. The foregoing differs somewhat from the seal water system described in the PSAR. We intend to discuss this further with the applicant.

The containment will be tested initially at a pressure of 47 psig and at a lower pressure. The leakage will be determined by the reference volume method. The specified leak rate for acceptance of the containment after completion of construction is 0.1%/day at 47 psig without benefit of the penetration pressurization system. These tests will be performed with the penetration pressurization system vented to the atmosphere and not pressurized. Additional leak rate tests can be performed at any pressure up to the design pressure when the plant is not in operation and precautions are taken to protect equipment and instruments from damage. The periodic test pressure for the containment will be set at the operating license stage of review.

The Indian Point II containment is the second "negligible leakage" containment structure that has been proposed for pressurized water reactors. The double containment concept proposed for the Malibu facility, with its pump back system, is also designed to provide negligible leakage. The counterpart of the Malibu pump back system is the Indian Point II penetration pressurization system. Both facilities are provided with isolation valve seal water systems to preclude leakage through pipes that penetrate the liner. The Staff believes that the proposed Indian Point II design contains the following additional advantages:

(a) The 5 fan-coolers for Indian Point II contain activated charcoal filters for cleanup of any elemental iodine in the containment atmosphere and will significantly reduce the inventory of iodine available for leakage if leakage should occur. The spray system also includes sodium thiosulphate and should be more effective than the Malibu spray system.

(b) The penetration pressurization system, with its associated pressure gauges,

low pressure alarms, and indicating lights, should enable prompt identification and repair of any point of leakage in the containment liner.

To provide a basis for evaluating the adequacy of the proposed engineered safeguards, the staff has calculated the highest allowable outleakage from the containment under MCA conditions which would result in potential offsite exposures within the guideline values specified in Part 100. The details of this calculation are presented in the Accident Analysis Section of this report. On the basis of this evaluation, we believe that Criterion 19 is satisfied.

Criterion 20

Same as discussed in Brookwood report.

Criterion 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

The following sources of power are available to operate the essential equipment including the vital instruments and control systems:

(1) The 138 kv Buchanan Substation located approximately one-half mile from the station auxiliary transformer that it supplies.

(2) The station generator via the unit auxiliary transformer.

(3) The three emergency diesel-generator sets.

(4) The station 60 cell, lead acid batteries.

The Buchanan substation is also connected to the Lovett station of the Orange and Rockland system and the Consolidated Edixon 138 kv transmission system via two overhead lines to Millwood East. This power source is connected at the Station to the auxiliary transformer which is used during startup, shutdown, and hot standby.

Once the main generator is sychronized to the 345 kv system, the Buchanan substation (external power) is still used to drive a feedwater pump and to supply two station service transformers. Most of the auxiliary load is supplied during operation by the main generator through the 40 MVA unit auxiliary transformer.

As a backup to the normal standby AC power supply, diesel generator sets will be provided with the capability of starting and supplying the power requirements of the engineered safeguards. There will be three emergency diesels that will automatically start on loss of voltage to the 480 volt bus stations, and can supply all of the engineered safeguards. If only two diesels are assumed to operate, those safeguards required to preclude containment overpressurization and core meltdown can be adequately supplied.

All components and structures of the emergency power supply system are vital to safe shutdown and isolation of the reactor and are therefore designed as Class I in terms of seismic design. This includes: Diesel generators and fuel oil storage tank, DC power supply system, power distribution lines required during an emergency, transformers of switchgear supplying the engineered safeguards, control panel boards and motor control centers.

We believe that there are sufficient normal and emergency sources of power available and that Criterion 21 is satisfied.

Criteria 22, 23, 24, 25, 26 and 27

Same as discussed in Brookwood report.

C. Accident Analysis

(1) Maximum Credible Accident

As stated previously in this report, the Indian Point II containment has been designed to have essentially negligible leakage following the assumed double-ended failure of the largest primary pipe. The achievement of this goal would result in insignificant off-site doses for the postulated accident condition.

The applicant has provided two independent iodine removal systems to limit the inventory available for leakage following an accident. These systems are the internal air filtration system (activated charcoal filters) and the containment spray system. Sodium thiosulphate will be injected into the containment spray water to aid removal of elemental forms of iodine. The Staff has engaged Dr. George Parker of the Oak Ridge National Laboratory to evaluate the design of these systems. Dr. Parker arrived at the following conclusions which he reported orally to the Committee at its May 1966 meeting:

(a) Water-logging of the charcoal filter units can be expected when operated in the anticipated MCA environment. Under these conditions, at least 90% removal efficiency can be assumed for elemental iodine, and zero efficiency should be assumed for the removal of organic iodine.

(b) The efficiency of the containment spray system is not greater than 90% for removal of elemental iodine. No removal efficiency can be assumed for organic iodine.

(c) The ignition of given charcoal filter units following the MCA is possible if the air flow through the given filter units is terminated. The applicant is aware of this problem and will provide temperature sensors, provision for backflow through a common discharge header, and spray systems to extinguish the fire (similar to the Brookwood facility).

With reference to filter ignition, we believe it is credible to assume that one or more fan cooling units may stop after the charcoal filters are loaded with iodine. To preclude ignition, sufficient backflow will be provided through the common discharge header to provide adequate cooling. In addition, if a fire should result, it would be extinguished by the spray system provided for this purpose. The adequacy of the filter backflow capacity and the filter spray system will be reviewed at the operating license stage.

As discussed under Criterion 19, the Staff has calculated the highest allowable outleakage from the containment under MCA conditions which would result in potential off-site exposures within the guideline values specified in Part 100. The following conditions were assumed:

Power level - 2758 MWt

Equivalent I^{131} available for leakage - 3.2×10^7 curies

(25% of total inventory)

Unfilterable iodine - 5% of inventory available for leakage

Fan capacity - 65,000 CFM each, 4 of 5 operating

Recirculation rate - 6 containment volumes per hour

Filter efficiency - as indicated in table below

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

No credit for containment spray system

The following table (also given in Report No. 1) presents a comparative tabulation of the potential off-site doses using the assumptions listed above and varying filter efficiencies. A TID-14844 calculation is given for comparison. To account for building dilution, the model suggested by Gifford and Fuquay has been used. The calculated doses for the 90% filter efficiency case are somewhat higher than those presented by the applicant. This results from the applicant's additional assumption of 70% filter efficiency for organic iodine, and a lower inventory of organic iodine.

		Integrated Thyroid Dose (rem)	
		Site Boundary (520 meters)	Low Population Distance (1100 meters)
Condition	Filter Efficiency (%)	2 hr.	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 for a filter efficiency of 90%, the recommended doses in 10CFR 100 are satisfied for a containment outleakage rate of 0.1%/day at 47 psig.

(2) Minor Accidental Releases of Radioactivity

In addition to the potential release of radioactivity under MCA conditions, several other means for the accidental release of radioactivity from this facility have been identified. These are:

(a) Steam generator tube failure

The applicant has stated that a steam generator tube failure will result in the blowdown of a significant portion of the primary system (about 6,000 cubic feet) into the secondary system. Under these conditions, a reactor scram and turbine trip will be initiated by the low primary system pressure trip and the secondary system steam will be dumped to the turbine condensers. The steam dump valve capacity of 40% of full load will prevent operation of the steam generator safety valves and subsequent discharge of radioactivity to the atmosphere via this route. However, radioactivity can be released by the air ejector. The air ejector effluent is monitored for radioactivity and will be diverted to the containment under these conditions. Thus, the only radioactivity release will occur before the air ejector effluent is diverted.

The potential off-site doses for this accident have been calculated assuming the primary coolant contains the fission

products resulting from 1% failed fuel elements. The most significant isotope is Xe-133 (concentration 200 uc/cm^3) and about 32,000 curies will be injected into the secondary system during the entire blowdown period. The applicant has estimated that less than 13,500 curies of Xe-133 will be released before the air ejector is diverted. The Staff has calculated that the resulting potential off-site whole body dose will be less than 0.5 rem (maximum dose for unrestricted areas - 10 CFR 20) for this activity release. We believe that this estimate is conservative, since the primary coolant should contain significantly less fission products than assumed, and that the potential dose is acceptable.

(b) Leakage from gas storage tanks

The maximum anticipated quantity of gaseous wastes in one storage tank is approximately equivalent to 13,500 curies of Xe-133. These tanks are in compartments such that any accidental venting operation or leakage would be released through the plant vent. As discussed in (a) above, the Staff believes that the off-site whole body dose for this condition would be less than 0.5 rem.

(c) Leakage from auxiliary building

The backup pumps for the recirculation system are located in the auxiliary building and would be required to pump radioactive water if both recirculation pumps failed within the containment. Leakage from those portions of the system located

outside the containment represents a potential means for the release of radioactivity. As discussed in Criterion 18, these components will be designed for minimum leakage. Also, the auxiliary building atmosphere is vented through absolute filters to the plant vent. If the components in the system were to leak at three times the minimum accepted rate, the two hour thyroid dose at the site boundary from this source would be 2.5 rem. We believe this dose is acceptable.

(d) Leakage through fan-coolers

The cooling coils for the five fan-coolers are cooled by water supplied by the service water system at a pressure of 20-25 psig. During the period following the MCA the containment pressure is above the service water pressure, and a leak at these coils would present a direct path for radioactivity to escape the containment. The applicant has stated that because of voltage limitations of the emergency diesels (voltage is 480 volts), it is not feasible to install high head pumps with the flow capacity required (5,000 gpm each) to assure that the service water pressure would be above the containment design pressure. To preclude significant leakage through this system, the applicant has proposed the following:

1) The containment leakage tests will be performed with the service water system depressurized and vented to the atmosphere. These periodic tests should assure that significant leakage paths do not exist at these cooling coils (the design pressure

of the coils is 150 psig).

2) The service water discharge will be continuously monitored for radioactivity. If leakage is detected, appropriate valves and test lines have been provided so that the leaking system can be identified and the leakage terminated.

In view of the short time period that the containment pressure is above 20-25 psig (about one-half hour if four fan coolers and one containment spray operate on emergency power) and the safeguards provided for detecting and isolating leaking systems, we believe that operation of this system in the manner proposed is acceptable.

D. Special Monitoring Considerations

Routine Effluent. Since both plants are pressurized water types, it is not anticipated that there will be need for the release of large quantities of gaseous effluent. Therefore, we are confident that the stack release limit previously established for Plant No. 1, which corresponds to the Part 20 limits, will be sufficient to permit operation of the second unit. Since the second unit will not use the superheater stack of Unit I but rather the Unit II plant vent, the amount of effluent which it can release will be reduced, but this still should not be a problem. The applicant was advised to give thought to the problem of how to divide the effluent limit between the stack and vent type of release points.

Liquid effluent can be expected to increase somewhat in proportion to the increase in power level. However, if the total quantity approaches 10 CFR 20

limits, one can always add additional evaporative type concentrating units to reduce the volume to permit shipment off-site for disposal. While the flow rate of the river gets rather low at times, it is so wide that it acts like a large lake to provide dilution. The volume of the river in the narrower part within a mile above and below the plant is equivalent to all the water the plant uses in a week. Further below the plant the river has an even greater cross section for several miles. In view of this, we do not believe that the plant will generate enough radioactivity to constitute a problem.

Environmental Monitoring. Con Ed already has a very extensive environmental monitoring program, which samples a wide variety of items at 44 sampling stations at a considerable distance from the plant. The items monitored at these sampling stations include the air, surface and drinking water, vegetation, fish and soil. We do not feel that any extension of this program is necessary for operation of another larger plant. We do anticipate that a requirement may be established for gamma dose monitoring at the site boundary if the gaseous emissions should exceed perhaps 10% of the annual average discharge limit which will be established at the operating phase of the licensing process. This has not been considered in the past because the stack emissions were so low.

Emergency Plans. It was brought out in a recent ACRS Subcommittee meeting that the emergency plans for this site mostly address themselves to radiation incidents which are related to 10 CFR 20 limits. It appears likely that some modification of the plans to include consideration of larger potential releases and updating these procedures may be required. It would not appear that the new

larger plant will influence these plans greatly, because additional engineered safeguards for it have been added to keep the magnitude of any potential accident within or below the same range as that for Plant No. 1.

E. Items Requiring Further Study

Specifically as a result of our review of this facility and generally because of our continuing review of pressurized water reactors, a number of problem areas have been identified. Many of these have been mentioned in the body of this report. We intend to keep the items listed below in mind as we continue our review of the Indian Point II facility and of pressurized water reactors submitted for subsequent licensing action.

1. Adequacy of diesel generator capacity.
2. Design of seal water system.
3. Consequences and causes of reactor vessel rupture.
4. Melting of core through reactor vessel.
5. Stresses in containment liner.
6. Applicability of RCC ejection point kinetics model.
7. Adequacy of air recirculation system.
    a. Flow rates under accident conditions.
    b. Back flow.
    c. Demister.
    d. Design of filters and charcoal beds.

8. Behavior of hydrogen after an MCA.

   a. Imperfect burning as released.

   b. Subsequent burning potential.

9. Control rod worth limiter.

Conclusion

On the basis of the foregoing safety evaluation of Indian Point Unit No. II, we have concluded that there is reasonable assurance that the facility can be built and operated at the proposed location without undue risk to the health and safety of the public. The Staff believes that the resolution of potential adverse effects of reactivity transients can be deferred to the operating license stage of review. As indicated, the control rod worth and moderator temperature coefficient can be appropriately limited if fianl design of the first core indicates that such limitations are necessary.

APPENDIX A

DISCUSSION OF ROD EJECTION ANALYSIS

1. Introduction

As requested the applicant has furnished a more complete description of their analysis of the control rod ejection transient. This work was done to support their design with regard to Criterion 7. Their report, designated as WCAP-2940, has been submitted as part of the Second Supplement to the PSAR for Indian Point 2. (Question 6 of the Second Supplement). Their report covers two major aspects of this problem; (1) the sensitivity of the analytical results to variations in various parameters, and (2) an attempt to estimate the amount of fuel melting required to fail the primary system. In the following we will discuss some of the results from the sensitivity study and present the major results from the applicant's consideration of potential primary system failure due to fuel melting.

This is the first opportunity the Staff has had to examine a detailed study of the rod ejection accident in a large core with a significantly positive moderator coefficient. Our results, based on the data given in WCAP-2940, are encouraging and do not indicate any serious problem areas. However, we intend to continue to work with the applicant to evaluate the adequacy of possible solutions to potential problem areas.

2. Discussion

The applicant has provided the results of four separate RCC ejection studies. The major results of the cases are presented in Table 1. The values given in the Table are approximate.

TABLE 1

MAJOR RESULTS OF RCC EJECTION CASES *

Item	Parameter	% ΔK OF EJECTED ROD			
		0.8	1.5	0.5	1.0
1	Initial Power (MWt)	0	0	2800	2800
2	Fq (Ejected Rod)	8.3	13.0	6.3	9.6
3	$\overline{\Delta H}$ (BTU/lb Fuel)	35	160	55	160
4	ΔE (Full Pwr Sec)	3	13	5	13
5	Volume % Melted	0	8	"0"	11

(1) Initial power in thermal megawatts.

(2) Fq is the power peaking factor associated with the distorted gross core power distribution for the ejected rod condition.

(3) $\overline{\Delta H}$ is the enthalpy rise of the fuel when volume averaged over all fuel in the core. The value given is for a time of 1.2 seconds after initiation of rod ejection. This is near the time of peak fuel temperature.

(4) ΔE is the nuclear energy generated by the excursion up to the time of 1.2 seconds after ejection measured in full power seconds (FPS).

(5) The % of fuel that is at least fully molten.

* It is important to note that the applicant believes that the lower rod worths for each initial condition are reasonable estimates of the maximum values to be expected during normal plant operation. The higher rod worths for each condition are provided only to show the sensitivity of the transient to this parameter.

The analytical method used in carrying out the transient analysis is based on computing the transient behavior of the average fuel pin in the core. This is done using a point kinetics code called CHIC-KIN. Spatial effects are incorporated by using appropriate weighting factors applied to the results and to the input data. The applicant compares their model with an adiabatic model used by Yasinsky which they show is conservative. However, it is not clear from their report that their use of the point kinetics model is equivalent to Yasinsky's adiabatic model and we intend to continue to work with the applicant to resolve this uncertainty. The applicant states that they have used their method to predict the SPERT transient results with some success and, pending further investigations, we will accept their method and examine only the results of the analysis.

The sensitivity of the transient analysis to variations in various parameters can be discussed in terms of $\overline{\Delta H}$, the average enthalpy rise in the fuel. In order to relate this parameter to fuel melting we have used the peaking factors given in Figure 2.3.8-4 of WCAP-2940 as a function of the ejected rod worth. This peaking factor, denoted Fq, is the ratio of the maximum power density to the average power density in the distorted (rod ejected) core. Using these peaking factors and the enthalpy value of 563 BTU/lb for the fully molten fuel condition given in Table 2.2.1 of WCAP-2940, we have obtained estimates for the enthalpy rise required to bring the volume averaged fuel <u>in the hottest pellets</u> to a fully molten condition for various ejected rod worths. We will designate this enthalpy rise as $\overline{\Delta H}_M$. These estimates are based on a volume averaged enthalpy in the hottest pellets of 30 BTU/lb in the zero power case

and 185 BTU/lb in the full power case.* Denoting these by $\bar{H}o$ and $\bar{H}f$ respectively, $\overline{\Delta H}_m$ can be found from the following relationship:

$$563 \frac{BTU}{lb} = \bar{H} + (Fq)\,(\overline{\Delta H}_m)$$

Where $\bar{H}$ is either $\bar{H}_o$ or $\bar{H}_f$. The results are given graphically in Figure 1. Of interest are the two low rod worth cases. From Table 1 we find that the full power case with a 0.5% rod ejection has an Fq of 6.3. Referring to Figure 1 we obtain a value of $\overline{\Delta H}_m$ of 55 BTU/lb for this Fq on the full power curve. According to Table 1 we see that the computed value for $\overline{\Delta H}$ is 55 BTU/lb indicating that for this case the average fuel in the hottest pellet is, for all practical purposes, fully molten. This result has been verified in subsequent discussions with the applicant. They indicate that for the 0.5% rod ejection case the center of the hottest pellet has vaporized and the volume averaged enthalpy is 550 BTU/lb. For the low rod worth case (0.8% from zero power) using a value for Fq of 8.3 we find from Figure 1 that $\overline{\Delta H}_m$ is 65 BTU/lb. The value of 35 BTU/lb given in Table 1 as the resulting $\overline{\Delta H}$ for this transient indicates that there is more margin to fuel melting in this case. These results are used in the following discussion of the applicant's sensitivity study.

In Figure 2 we have attempted to show graphically the effect on $\overline{\Delta H}$ of variations in three of the more significant fundamental parameters. The other parameters discussed in WCAP-2940 were shown there to have either a smaller effect

* Information obtained in subsequent discussions with the applicant

FIGURE 1

AVERAGE FUEL ENTHALPY RISE REQUIRED TO COMPLETELY MELT THE AVERAGE FUEL IN THE HOTTEST PELLET FOR A GIVEN PEAKING FACTOR

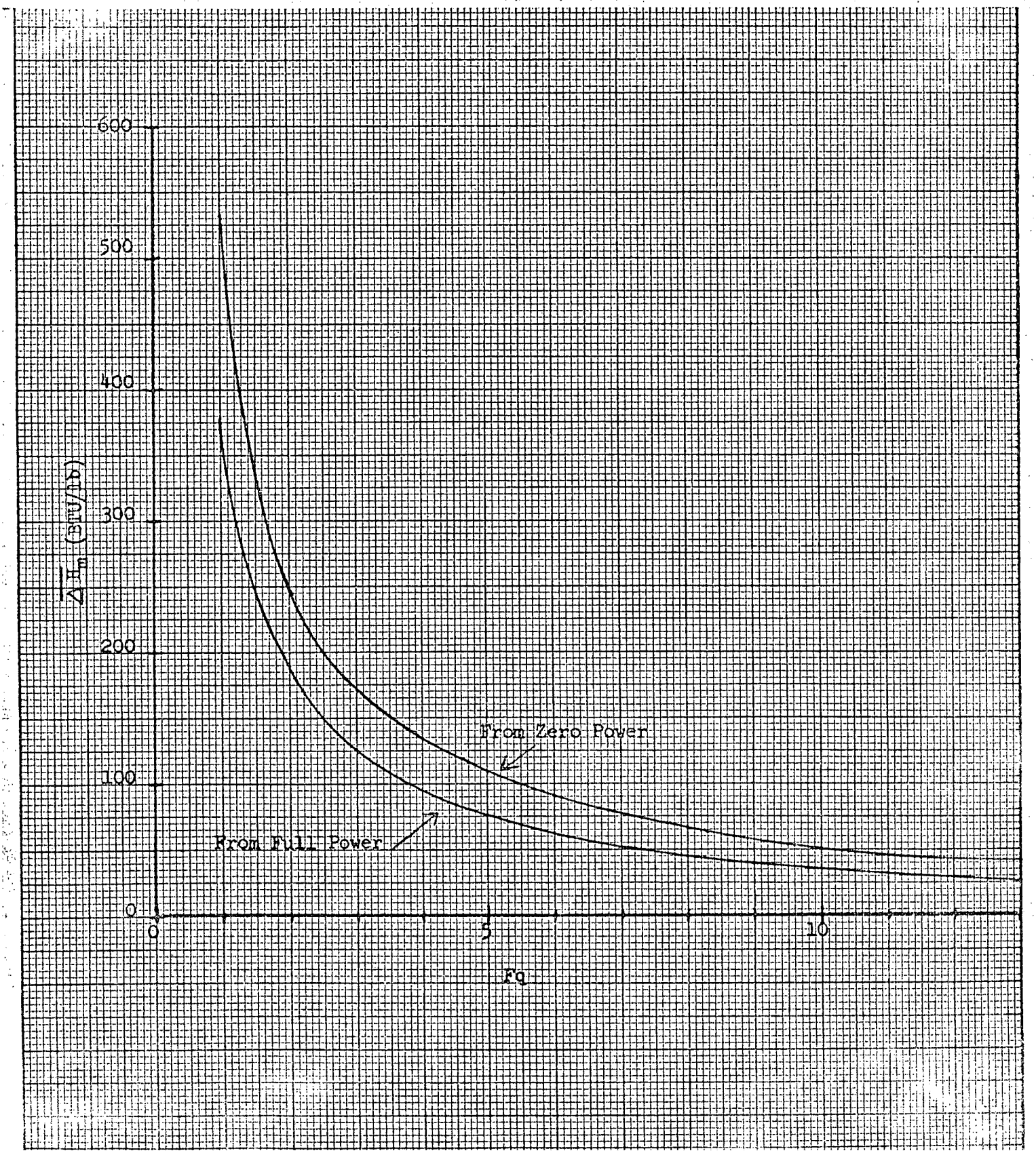
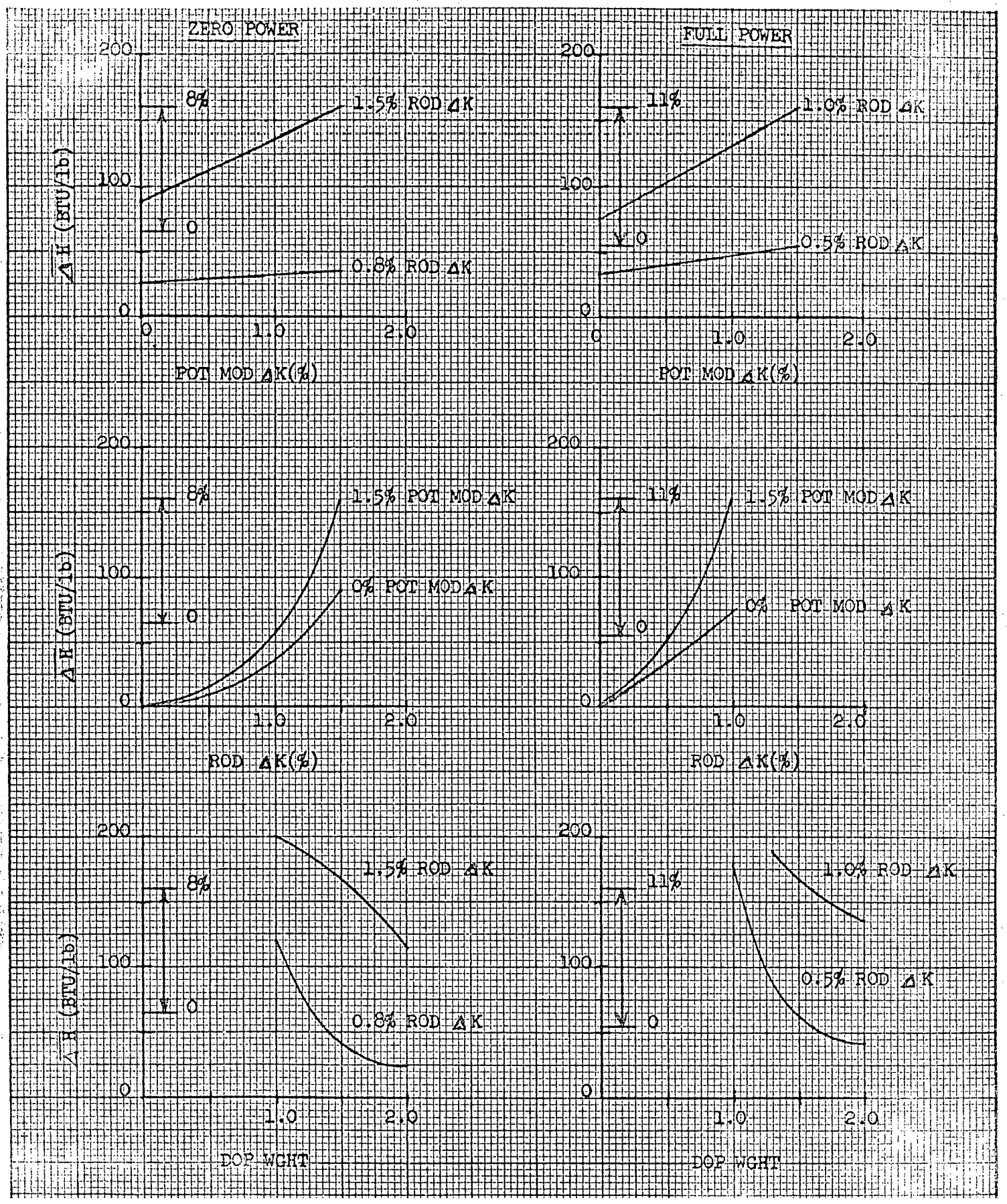


FIGURE 2

COMPARISON OF THE EFFECT OF SOME VARIABLES ON $\overline{\Delta H}$



on the results or to be chosen and used with enough conservatism to provide an adequate safety margin over the range of anticipated values. Two possible exceptions are the trip delay time and moderator inlet temperature. However, these are operational parameters and should be amenable to proper adjustment and control. Reading from top to bottom on Figure 2 the parameters are; potential moderator reactivity insertion (POT MOD ΔK(%)), ejected rod worth (ROD ΔK(%)) and the Doppler weighting factor (DOP WGHT). We should note here that a given value of parameter POT MOD ΔK(%) does not mean that this amount is necessarily inserted during the transient. For the high rod worth cases all of the potential moderator insertion is realized while for low rod worth cases only a fraction of the maximum available is obtained at the peak of the moderator reactivity curve. This is discussed in more detail in WCAP-2940 and is illustrated by the figures in the report showing reactivities as a function of time.

The three sets of curves on the left side of the page are for transients starting from zero power and those on the right for transients starting from full power. Two curves are given in each figure providing a second variable for consideration. Rod worth is used in the upper and lower sets and potential moderator insertion in the middle sets.

The region along the ordinate denoted by the arrow is an <u>indication</u> of the range of fuel melting involved. It extends from the value of $\overline{\Delta H}$ required to cause melting as computed above for the low rod worth cases to the volume percent of fuel melted for high rod worth cases as computed by the applicant and given in WCAP-2940.

Inspection of these curves reveal several significant features. The upper two sets show that substantial fuel melting would probably occur for both high rod worth cases even with a zero potential moderator reactivity insertion. Again we note that these rod worths are not considered credible during normal operation. The moderator coefficient has a relatively slight but not insignificant influence on $\overline{\Delta H}$ in the low rod worth cases. The difference between a zero potential moderator insertion and the potential 1.5% insertion computed by the applicant is only 10 BTU/lb in the zero power case and 20 BTU/lb in the full power case. The ejected rod worth is a more sensitive parameter as indicated by the central sets of curves. As expected, the sensitivity increases rapidly as the rod worth value is increased beyond prompt critical. Note that the difference between a "little" molten fuel and 11% fully molten in the full power case is due to only a factor of 2 in ejected rod worth. We note from these graphs that an important effect of increasing the moderator coefficient (more positive) is to increase the sensitivity of $\overline{\Delta H}$ to the ejected rod worth.

The effect of variations in the Doppler weighting values is presented in the lower set of graphs. The applicant has computed a value of 1.6 for this quantity. Any reduction below this value would be significant, particularly for the low rod worth cases. An important observation here is that increasing the weighting factor from the assumed value of 1.6 to a value of 1.8 results in a relatively slight reduction of $\overline{\Delta H}$. For the 0.5% full power case, the reduction is from 55 BTU/lb to 45 BTU/lb. A comparable reduction of the weighting factor to a value of 1.4 would result in a $\overline{\Delta H}$ of 75 BTU/lb.

Probably the most significant result of these considerations is that if we assume a failure criteria such that we do not allow melting of the average fuel in the hottest pellet, then the 0.5% rod ejection from full power with a $\overline{\Delta H}$ value of 55 BTU/lb is marginal. However, we can reduce this value of $\overline{\Delta H}$ from 55 BTU/lb to roughly 35 BTU/lb by reducing the potential moderator insertion to zero. It seems likely then, barring serious errors in the analytical model, that margin can be provided to prevent fuel melting in the rod ejection accident.

The applicant has shown in his report that shock wave failure of the pressure vessel would require rapid dispersal of 72,000 pounds of molten UO_2 (about 1/3 of the core) having an average enthalpy of 525 BTU/lb. This result is based on the experimental work done by our consultant. Mr. James Proctor of the Naval Ordnance Laboratory. The applicant has used these results to obtain the minimum explosive weight required to fail the Indian Point 2 pressure vessel. They obtained a value of 1700 pounds of TNT at an energy equivalence of 1050 calories/gram assuming a 50% energy transfer efficiency. This corresponds to about 0.6 full power seconds for the Indian Point plant. Failure is assumed to occur at a mid-meridian ring strain value equal to 50% of the total elongation of SA 302 Grade B steel. Although other modes of primary system failure, such as primary piping, may be at least as likely, they were not considered by the applicant in the report. Mr. Proctor has reviewed their results and concurs in the value of 1700 pound of TNT required for shock failure although not necessarily with their calculation leading to the fuel equivalence of 72,000 pounds. In discussing the foregoing with the applicant, he advised them that "quasi-static" pressure surge considerations are probably the more limiting case. Following his

suggestion the applicant has found that the quasi-static pressure surge could reach the failure point with only 7 or 8% of the fuel melted and dispersed. (See page 36 of WCAP-2940). The failure point is 6500 psia using the half ultimate strain criteria. Energy due to the Zr-water reaction contributed from the Zr cladding associated with the failed fuel was included in these results. Experimental data giving the percent Zr reacted as a function of fuel enthalpy was used to obtain this contribution. The result is that the Zr-water reaction contributes about half as much energy as does the failed fuel." An estimate of the energy transferred to the coolant required to produce this pressure surge may be made by integrating under the upper curve in Figure 4.1-7 of WCAP-2940. The result is about 3 full power seconds (FPS) up to 0.5 second and 4.5 FPS up to 1.0 second after initiation of the transient. This is in fair agreement with the experimental results of our consultant. Mr. Proctor states that it will take about 4 to 6 times as much energy input to the system to fail in the quasi-static mode as in the shock mode. Since the 1700 pounds of TNT required for shock mode failure is equivalent to 0.6 FPS, a factor of 5 would give 3.0 FPS required to fail in the quasi-static mode. As noted the agreement is reasonable considering the uncertainties involved.

3. <u>Conclusions</u>

The discussion in Section 2 indicates that considerable uncertainties are involved in predicting fuel failure and its consequences. Hence, the Staff believes that a very conservative fuel failure criteria is still required to guarantee the integrity of the primary system during a nuclear excursion.

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* Information obtained in subsequent discussions with the applicant.

It is difficult to specify a justifiable quantitative criteria; however, we believe that to provide a sufficient safety margin one should prevent excursions that could result in incipient vaporization of the hottest fuel or total melting of the volume averaged fuel in the hottest pellet. We believe that this criterion will preclude any significant amount of violent fuel failure which could lead to primary system failure. The results discussed in Section 2 indicate that the 0.5% rod ejected from the full power would be marginal in this case. However we believe that by suitable rod worth limitations and, if necessary, be reducing the moderator coefficient, the system can be designed to meet the fuel failure criteria specified above. The moderator coefficient could be reduced, for example, by insertion of some form of non-soluble boron into the core with consequent reduction of the soluble boron in the coolant. The applicant is investigating various means to implement this method of reducing the positive moderator coefficient. We intend to work with the applicant in determining quantitatively what steps must be taken to assure that no significant fuel failure can occur due to nuclear excursion accidents in this system.

APPENDIX B

CONTAINMENT DESIGN REQUIREMENTS

1. Introduction

As discussed in Reports No. 1 and No. 2, the Staff has been developing an approach to simplify the evaluation of the maximum pressure that could arise in the containment due to energy inputs from a loss-of-coolant accident. In this report we are presenting what we believe is an acceptable technique for evaluating the Indian Point 2 containment pressure design.

In general we will illustrate the "capability" of the containment to accept energy releases in which the rate and extent of the metal-water portion of the total energy release are arbitrarily varied and independent of specific detailed models. The discussion will cover (1) the model, (2) calculated containment pressures, (3) tolerable amounts of metal-water reactions, and (4) evaluation of the Indian Point 2 containment pressure design.

2. Model

Three items of importance in determining containment pressure after the blowdown are: (1) the amount of the metal-water reaction, (2) the time interval in which this reaction occurs, and (3) the remaining heat sources and the heat removal capabilities. After investigating the various models used to predict the first two items above, the Staff believes that one cannot at this time predict with confidence either item. However, as a result of our investigations we believe a simplifying approach for estimating the possible maximum reaction at any time is to assume a maximum credible extent of reaction and have this amount of reaction occur linearly with time with completion in an assumed time period. The extent of reaction and time period assumed by the Staff, as

described in Section 4, are more conservative than the values calculated by the applicant. We have concluded that the use of this type of model will result in peak containment pressures in close agreement with methods using detailed models with varying reaction rates (but the same extent and time of reaction). In addition, our model has the advantage of better identifying the areas of conservatism, i.e., the assumed extent and time of reaction.

With regard to the energy sources in the containment, we have chosen the following items which, except for additional secondary coolant stored, consist of all of the significant available energy sources:

(1) Primary coolant energy brought into the containment in the form of steam and producing the original blowdown containment pressure-total energy of 310×10^6 Btu.

(2) Core decay heat (assuming infinite operating time) as produced and brought into the containment in the form of steam-total energy of 40×10^6 Btu in 2000 seconds.

(3) Initial core stored energy brought into the containment in the form of steam at a uniform rate in 2000 seconds - total energy of 28×10^6 Btu.

* (4) Secondary coolant energy brought into the containment in the form of steam at a uniform rate in 2000 seconds - total energy of 40×10^6 Btu.

(5) Metal-water reaction energy brought into the containment in the form of steam at a uniform rate within an assumed time period - total energy from a 100% reaction of 115×10^6 Btu.

(6) Hydrogen-oxygen recombination energy added <u>directly</u> to the containment steam-air mixture <u>at a uniform rate</u> within the same time interval as the metal-water reaction energy - total energy of 89×10^6 Btu.

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*The total stored energy of the secondary coolant in the steam generators is 160×10^6 Btu. Items 2, 3, and 4, in a 2000 second time interval are equivalent to a 40% metal-water reaction (including hydrogen recombination energy).

With regard to heat sinks we consider the following three situations:

(1) structures only - considered as "0 safeguards."

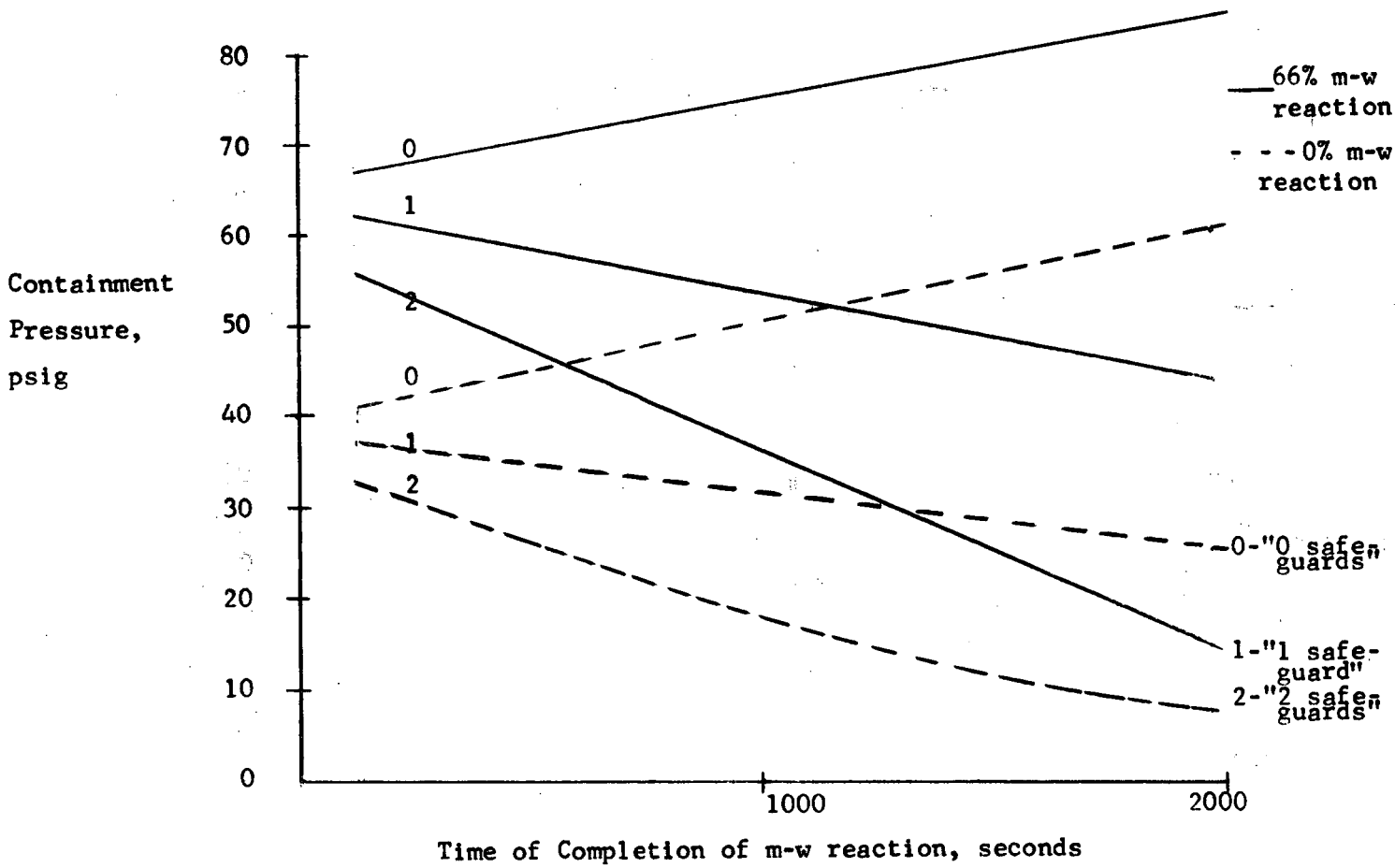
(2) structures plus 5 fan-coolers (equivalent of 2 containment spray pumps) - considered as "1 safeguard."

(3) structures plus 5 fan-coolers plus 2 containment spray pumps - considered as "2 safeguards."

3. Calculated Containment Pressures

Without consideration of any specific metal-water reaction model, we asked the applicant to supply the calculated containment pressures for given amounts of metal-water reactions occurring linearly within given time intervals using the heat sources and sinks discussed above. Figure 3 shows these calculated pressures at the time of completion of the metal-water reaction. The figure, for presentation purposes, indicates the pressures for only two amounts of reactions, 0 and 66%, completed in time periods up to 2000 seconds with the various sinks.

Figure 3 - Containment Pressure at Time of Completion of Metal-Water Reaction



As an example, a 66% m-w reaction completed in 1000 seconds with 5 fan-coolers operating results in a containment pressure of 55 psig. The 0 line is significant as it indicates that with structural sinks only, the containment pressure increases with time due to the decay heat, original stored energy, and secondary coolant energy brought into the containment. It illustrates the importance of integrated core decay heat over long time periods.

One consideration not included in Figure 3 is the possibility of a hydrogen-oxygen reaction being delayed and then occurring all at once at the completion of the assumed m-w reaction. This would result in higher peak containment pressures with heat removal systems operating, since a lower pressure and lower water content would exist at the time of the delayed burning thus resulting in more effective superheating of the containment atmosphere.

Figure 4 - Effect of Hydrogen Burning on Containment Pressure

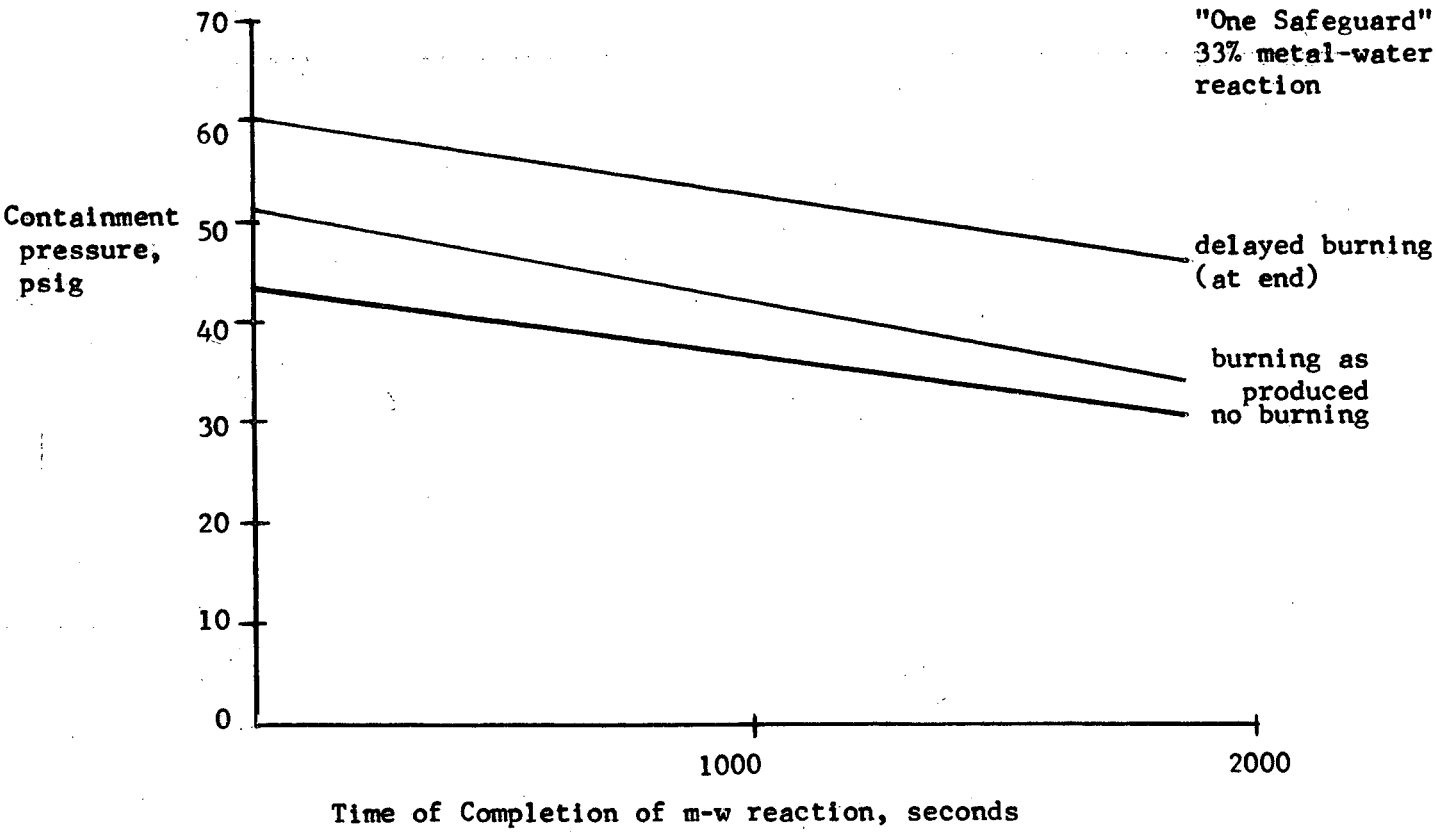


Figure 4 shows typical effects on containment pressure for the following 3 situations:

1. no hydrogen-oxygen reaction (hydrogen never burns),
2. hydrogen-oxygen reaction occurs as hydrogen is produced (the assumption used for Figure 3),
3. hydrogen-oxygen reaction delayed until completion of m-w reaction (there are numerous reasons which reduce this possibility).

At this point we will refer to our discussion of pressure coefficients presented in Report No. 2. In that report we attempted to summarize the containment pressure picture by determing coefficients which could indicate the expected pressure change associated with perturbations of given parameters. As stated previously we believe these coefficients can be used to obtain the pressure for many combinations of time and extent of reactions as, for example, they could have been used to obtain the results shown in the preceding Figures 3 and 4. We have reviewed the contents of Table 3 of Report No. 2 with the applicant and as a result have prepared a slightly revised set of coefficients which are shown below in Table 1.

Table 1

*Coefficients Related to Peak Containment Pressures

Parameter	Pressure Coefficients
Metal-water reaction (excluding H_2 burning)	+ 2 psi per 10% reaction
Hydrogen burning as produced	+ 2 psi per 10% reaction
Delayed hydrogen burning (at end)	+ 6 psi per 10% reaction
Spray pump	- 9 psi per pump per 1000 secs
Fan Cooler	- 3.5 psi per fan per 1000 secs
**Structural sinks	- 1 psi per 100 secs
**Decay heat	+ 0.5 psi per 100 secs
Original stored energy (for first 2000 seconds only)	+ 0.5 psi per 100 secs
Steam generator	+ 0.5 psi per 100 secs

* Coefficients should be used to find the decrease or increase beyond the original blowdown pressure of 39 psig.

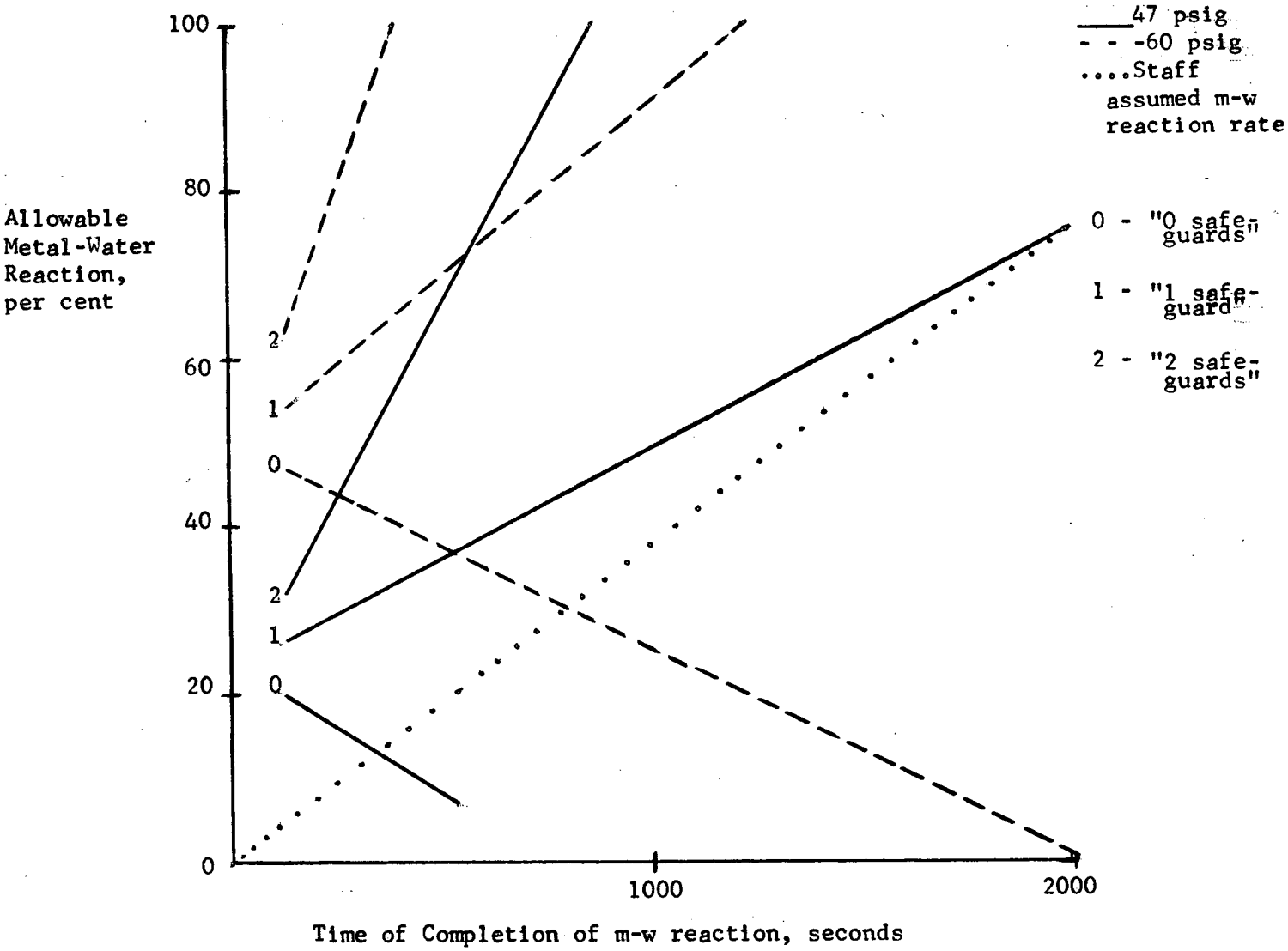
** After 2000 seconds structural absorption decreases below decay heat generation so that net increase in pressure of approximately 1 psi per 1000 seconds would occur. (See question 2 of Second Supplement to PSAR for long term effects).

4. Tolerable Metal-Water Reactions

We have replotted Figure 3 indicating the allowable (or tolerable) amount of m-w reaction so as not to exceed the design pressure of 47 psig (solid line) and an additional arbitrarily selected pressure of 60 psig (dashed line). We believe this type of representation more clearly illustrates the containment capability and sensitivity to m-w reactions. As an example of its use, examination of

Figure 5 shows that at 1000 seconds with "one safeguard" operating a 50% m-w reaction is tolerable without exceeding 47 psig (80% is tolerable without exceeding 60 psig). The dotted curve of Figure 5 is the staff assumed m-w reaction model discussed previously and limited to a maximum credible total reaction of 75% in 2000 seconds. For comparison, the applicant calculated a maximum reaction of 43% to occur within 2300 seconds. We have therefore shown a linear reaction from 0 to 75% occurring in 2000 seconds. This results in our model predicting a 40% reaction in 1000 seconds which, for our example, is lower than the value of 50% allowable without exceeding the 47 psig design pressure. We would therefore conclude that the Indian Point 2 containment can tolerate m-w reaction at 1000 seconds greater than that which we would conservatively have predicted to occur.

Figure 5 - Allowable Metal-Water Reaction Without Exceeding Containment Pressures of 47 psig and 60 psig



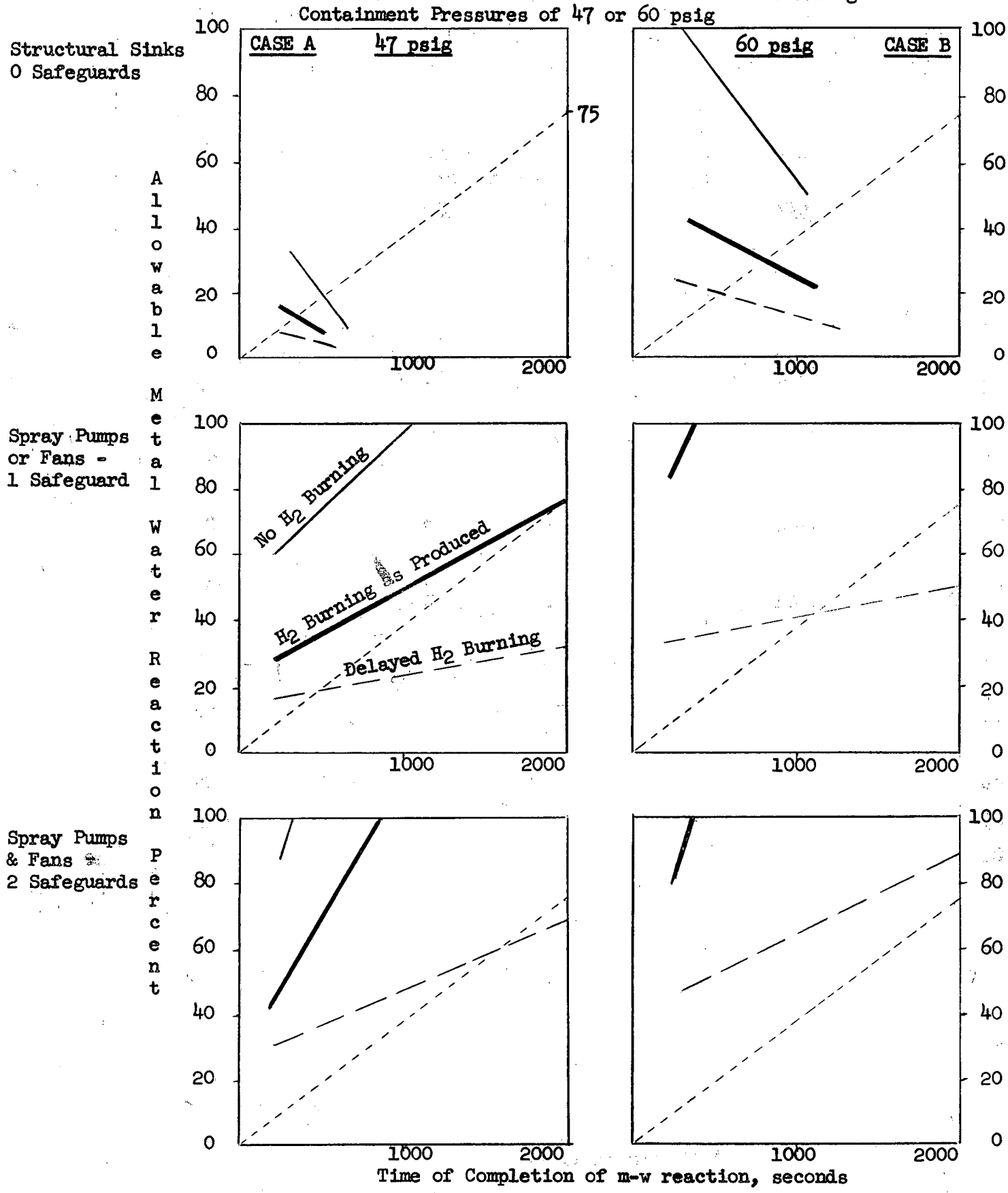
5. Evaluation of the Indian Point 2 Containment Pressure Design

To assist in evaluating the Indian Point 2 containment pressure design we have prepared Figure 6 which is an attempt to present the overall containment capability picture. The values have been obtained using our proposed model as discussed previously and are presented in the form of allowable metal-water reaction that causes the containment pressure to reach a defined pressure versus the time to complete the reaction. The dotted line is the Staff's assumed m-w reaction rate. For the 47 psig design case, where a plotted value for Indian Point 2 lies above this line, the containment pressure design would be considered adequate since the tolerable m-w reaction to prevent exceeding design pressure is more than the amount of reaction assumed possible. We have also included for completeness the cases of no hydrogen burning and for delayed hydrogen burning although we believe the later case, especially for complete delayed burning, is highly improbable based on both mixing and stoichiometric requirements.

It should be recognized at this point that the use of the decay, stored, and steam generator heat sources is somewhat conservative and if none of these energy sources were added to the containment atmosphere the allowable m-w reaction would increase by 40%. In fact, these energy sources and the choice of the metal water reaction rate are the areas of conservatism in our evaluation. The significant observations from Figure 6 for Case A (47 psig) are,

1. With no safeguard systems operating (only structural heat sinks) the containment could not tolerate any metal-water reaction in approximately 800 seconds without exceeding the design pressure.

Figure 6 - Indian Point II Containment Capability Curves
Allowable Metal-Water Reaction At a Given Time Without Exceeding
Containment Pressures of 47 or 60 psig



2. With one safeguard system operating the tolerable m-w reaction for hydrogen burning as produced is always greater than our assumed value. In addition a 100% reaction could be tolerated if it were completed in greater than 3000 seconds with a 25% reaction occurring immediately (0 seconds) - these values were obtained by extrapolation.

3. With two safeguard systems operating, the tolerable m-w reaction for the highly improbable situation of complete delayed hydrogen burning is essentially always above our assumed value. In addition a 100% reaction could be tolerated if it were completed in greater than 900 seconds.

The significant observations for Case B (60 psig) are the same as above except the time periods for items 1, 2, and 3, are 2000, 200, and 400 seconds respectively and for item 2 a 55% reaction could be tolerated immediately. This case has been included to indicate the sensitivity of our results to pressure.

In evaluating the above observations one should recognize that the Indian Point 2 containment design was not based on this analysis and therefore meeting any exact requirements specifically related to the Staff's assumed model is not necessary. We also believe that since we have some understanding of the areas of conservatism and the sensitivity of the model to its assumptions, we are in a position to accept more general, in contrast to more specific, requirements. Items 2 and 3 above which concern the energy removal capabilities of one and two safeguard systems could at this time be considered as requirements of an acceptable design (i.e., one safeguard is adequate for the case of

hydrogen burning as produced, and two safeguards are adequate for the case of delayed hydrogen burning). Item one indicates that with structural sinks only, an additional heat sink is necessary to satisfy heat removal requirements for remaining below design pressure (for Indian Point 2 approximately 15 minutes is available on a conservative basis before design pressures are exceeded). This area is discussed further by the applicant in Question 5 of the Second Supplement to the PSAR.

In summary, the Staff believes that the Indian Point 2 containment and engineered safeguards are sufficient to limit the maximum credible pressure of the containment to the design value of 47 psig following the MCA.