NEWS from the office of THE GOVERNOR

FOR IMMEDIATE RELEASE 217-782-7355

SPRINGFIELD, Ill., July 26--Governor James R. Thompson released the report Thursday of the Ad Hoc Investigating Committee of the Illinois Commission on Atomic Energy, which concludes there is no need to close nuclear power plants in Illinois.

"I welcome ongoing debate and constructive criticism of the operation of nuclear power generation in our state. Because nuclear power is an important part of the Illinois economy and will remain so, we must make certain that every reasonable precaution is taken to ensure public health and safety. I am grateful for the committee's recommendation in this area and appreciate their hard work."

The Committee, appointed by the Governor after the accident at Three Mile Island, Pennsylvania, investigated the safety of Illinois nuclear power plants. The Committee based its conclusion in part on the substantial operating differences between the nuclear plants in Illinois and at Three Mile Island.

The Committee made 49 recommendations directed at the state, the Nuclear Regulatory Commission and the utility companies. The Governor said he is requesting the Department of Public Health, the Emergency Services and Disaster Agency, the Illinois Commerce Commission and the Institute of Natural Resources to review the recommendations. The State already is complying with some of the recommendations, he said.

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The Governor also directed his staff to determine the extent to which Illinois has authority to conduct independent safety audits and directed them to review the desirability of the state assuming the inspections and enforcement at all nuclear facilities, a task now performed by the U.S. Nuclear Regulatory Commission. Twenty-six other states have assumed these duties to date.

The Governor said he will ask Joseph Hendrie, Chairman of the Nuclear Regulatory Commission, to arrange for the NRC staff to meet with members of the committee to discuss recommendations concerning operating staff, operating procedures and technical aspects of facility operation.

"Implementation of these recommendations concerning facility operations are uniquely the responsibility of NRC," the Governor said. "I hope that the NRC will meet with our committee with an eye toward incorporation of as many recommendations as are appropriate in NRC's on-going review of nuclear power generation in the wake of Three-Mile Island."

The Governor sent copies of the report to the legislative committees investigating nuclear power in the state and asked the ad hoc investigating committee to be available to discuss any aspects of the report with legislators.

The Ad Hoc Committee, appointed on April 3, 1979, consisted of Dr. Philip Gustafson of Argonne National Laboratory; Dr. J. B. Van Erp of Argonne; Dr. George Miley and Professor Daniel Hang of the Nuclear Engineering Department at the University of Illinois, and Gerald R. Day, Executive Director of the Illinois Commission on Atomic Energy.

To
THE HONORABLE JAMES THOMPSON
GOVERNOR OF ILLINOIS
from
Ad Hoc Investigating Committee
Illinois Commission on Atomic Energy

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VICE CHAIRMAN
DR. PHILIP F. GUSTAFSON
SECRETARY
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EXECUTIVE DIRECTOR
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WILLIAM L. WATSON

July 13, 1979

The Honorable James Thompson
Governor of the State of Illinois
State Capitol
Springfield, Illinois 62706

Dear Governor Thompson:

On April 3, 1979, you requested the Illinois Commission on Atomic Energy appoint an Ad Hoc Investigating Committee to conduct a study of the existing nuclear facilities in Illinois and report to you no later than July 15, 1979.

As Chairman of the Commission, I appointed the following Ad Hoc Committee:

Dr. Philip F. Gustafson - Argonne National Laboratory - Chairman
Dr. Jan B. van Erp - Argonne National Laboratory
Dr. George Miley - University of Illinois
Professor Daniel F. Hang - University of Illinois
Gerald R. Day - Executive Director of the Illinois Commission on Atomic Energy.

This committee was also assisted by Mr. Gary Wright of the Department of Public Health and Mr. John Hasselbring of the Illinois Commerce Commission. Various other persons were also consulted.

The attached report is a result of their investigation and is submitted for your information. If you have any questions regarding this report, I have been assured that the Ad Hoc Committee will be pleased to meet with you or your staff at your convenience. You may contact them by calling Mr. Day, the Executive Director, at 782-5057.

If this Commission can be of any further assistance to you or the people of the State of Illinois, please let us know.

Respectfully yours,

George Ray Hudson

George Ray Hudson, Chairman
Illinois Commission on Atomic Energy

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NUCLEAR POWER REACTOR SAFETY IN ILLINOIS

A Report to the Honorable James Thompson,
Governor of the State of Illinois

by

The Ad-Hoc Nuclear Power Reactor
Safety Review Committee,
Illinois Commission on Atomic Energy

July 1979

ABSTRACT

Governor James Thompson of Illinois appointed on April 3, 1979, an Ad-Hoc Nuclear Power Reactor Safety Review Committee. This Committee was charged with the investigation of the safety of nuclear power plants in Illinois, in the light of the accident that took place on March 28, 1979 at the Three Mile Island (TMI) nuclear power station near Middletown, PA.

The primary conclusion reached by the Committee during its investigation is that, in view of the substantial differences existing between the operating nuclear power plants in Illinois and the TMI nuclear reactor, there does not exist any ground for shutting down nuclear power plants in Illinois. The Committee did, however, identify areas where potential improvements appear to be desirable. These areas of potential improvements have been described in a total of 49 recommendations, subdivided into the following four categories: (A) General and/or Policy Aspects (19 recommendations), (B) Operating Staff and Operating Procedures (10 recommendations), (C) Technical Aspects (16 recommendations), and (D) Long-Term Considerations (4 recommendations).

Among the principal recommendations, requiring if implemented, an action by the State, are the following: (A.1) Emergency Plan, page 12; (A.2) Agreement State, page 12; (A.3) State Safety Audits, page 12; (A.4) State-NRC Coordination, page 12; (A.14) Emergency Operating Centers, page 15; (A.16) Off-Site and On-Site Monitoring, page 15; and (A.17) State-Utility-NRC Coordination Regarding Public Statements, page 16.

Other important recommendations include: (B.1) Appointment of Individuals Having Higher Training and Analytical Ability for Duty on Shifts, page 17; (B.2) Training and Re-training of Operators, page 17; (C.11) Use of Computers in the Control Room, page 20; and (C.15) Man-Machine Interface, page 20.

In view of the limited time that was available for this investigation, many of the recommendations are still of a preliminary nature, requiring further, more detailed, study prior to possible implementation.

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NUCLEAR POWER REACTOR SAFETY IN ILLINOIS

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I. INTRODUCTION

Following the accident on March 28, 1979 at the Three Mile Island (TMI) nuclear power station in Pennsylvania, Governor James Thompson of Illinois appointed on April 3, 1979, an Ad Hoc Nuclear Power Reactor Safety Review Committee, which was to function under the responsibility of the Illinois Commission on Atomic Energy. This Committee was charged with the investigation of all aspects bearing on safety concerning nuclear power plants in Illinois, whether in operation or under construction.

The investigation was to proceed in two phases: Phase 1, lasting approximately three (3) weeks, after which the Committee was to submit a Preliminary Report bringing out advice as to whether any grounds exist for shutting down any one or all of the nuclear power plants operating in Illinois in the interest of protecting the public health and safety, and Phase 2, lasting approximately three (3) months, after which the Committee was to submit a Final Report to Governor Thompson and the Illinois Legislature concerning the safety of nuclear power plants in Illinois.

The Ad Hoc Committee consisted of the following members: Philip F. Gustafson (Chairman of the Committee) and Jan B. van Erp, both of Argonne National Laboratory; Gerald R. Day, Executive Director of the Illinois Commission on Atomic Energy; George H. Miley and Daniel F. Hang, both of the University of Illinois, Urbana. In addition, the Committee had the benefit of the participation of the following persons, in the capacity of liaison, observer, and/or advisor: Gary W. Wright, Illinois Department of Health; John Hasselbring, Illinois Commerce Commission; and James P. Hartnett, Illinois Energy Resources Commission (also University of Illinois, Circle Campus). The latter individual had to limit his participation to the early (Phase I) activities of the Committee, in view of other (overseas) commitments.

In carrying out its assigned task, the Committee met with representatives of the Commonwealth Edison Co. (Cordell Reed, Assistant Vice President, CECo, and others) on the following dates: 4/4/79, 4/17/79, 5/3/79, 6/19/79, and 7/3/79. Site visits to operating nuclear power plants of CECo were made on the following dates: 4/10/79 (Zion), 4/12/79 (Dresden), and 4/18/79 (Quad Cities).

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The Committee, or members thereof, met with Representatives of the U.S. Nuclear Regulatory Commission on 4/9/79 (J. Kepler, Director, Region III Office, NRC) and on 5/29/79 (Joseph Hendrie, Chairman NRC, and Harold Denton, Director Office of Reactor Regulation, NRC).

Members of the Committee had discussions with Representatives of the Illinois Power Co. (Leonard Koch, Vice President IPL, and others) on 6/6/79, 6/29/79, and 7/10/79.

Members of the Committee met on 6/7/79 with representatives of the International Brotherhood of Electrical Workers, the labor union representing the nuclear power plant operators.

Furthermore, the Committee, or members thereof, conducted numerous internal work meetings (e.g., on 4/30/79, 5/29/79, 6/19/79, 6/27/79, 7/6/79, and 7/11/79) as well as numerous telephone consultations.

In its dealings with the various parties involved (utilities, NRC, labor union), the Committee has encountered excellent cooperation, for which it wishes to express its great appreciation.

In the following, the impact of the TMI accident on the health and safety of the general public will first be discussed, so as to gain a better perspective relative to this highly publicized event. Then some of the main features and characteristics of current US nuclear power plants will be described. After that, the chronological sequence of events pertaining to the TMI accident will be discussed; also will be discussed the differences existing between pressurized water reactors (PWRs), designed and built by different manufacturers, that would have affected this sequence. Finally, the Committee's findings, recommendations, and conclusions will be presented.

II. HEALTH AND SAFETY OF THE GENERAL PUBLIC RELATIVE TO THE TMI ACCIDENT

Table 1 gives some of the important data regarding the radiation doses received by the general population living within 50 miles of the TMI nuclear power plant, as reported by the Nuclear Regulatory Commission (NRC). One notes that the radiation dose to an average person due to the natural background (i.e., due to natural sources of radiation such as cosmic radiation, radioactivity in the soil, radioactivity in building materials, etc.) in the TMI area is about 125 mrem/year. On the other hand, the cumulative radiation dose to an average person within the 50-mile radius due to the accident was approximately 1.5 mrem from the start of the accident on March 28, 1979, through April 7, 1979 (i.e., about 1% of the annual dose due to the natural background). Furthermore, it is estimated that the maximum radiation exposure to any person in the general population, living in the immediate vicinity of TMI, is less than 100 mrem (assuming continuous presence at a distance of 0.5 mile of the plant in the NE direction). For comparison, it may be mentioned that an average medical X-ray results in a radiation dose of between 40 and 50 mrem. Thus, the maximum dose received by any member of the general public is approximately equal to that associated with two medical X-rays; it is also equal to the dose accumulated by flying airline personnel in four to six weeks due to the increased cosmic radiation at greater height (about 1 mrem/hour). It should also be kept in mind that this maximum dose was received by only a very small fraction of the population.

The collective annual radiation dose received by the population (i.e., dose multiplied by the number of persons receiving the dose) is 270,000 man-rem due to the natural background, whereas the cumulative collective dose due to the accident is about 3,300 man-rem. Again one notes that the collective dose due to the accident is about 1% of the annual value due to the natural background.

Three types of radiation can be distinguished, namely alpha (α), beta (β), and gamma (γ). Of these, α radiation consists of charged helium atoms, β radiation consists of electrons, and γ radiation is similar in nature to the X-rays used for medical purposes.

Exposure of humans or animals to radiation may take place in essentially two different ways, namely (1) external to the body, or (2) internally by ingestion or inhalation of radioactive material. External exposure is essentially limited to γ-rays. Internal exposure can take place by any one or a combination of the above-mentioned three types of radiation (α, β, and γ), depending on the type of radioactive material that is ingested or inhaled. In the TMI accident, the radioactive isotopes that were released and caused exposure of the general public were essentially limited to the noble gases (primarily xenon and krypton), which are chemically inert, and therefore readily released from the lungs, when inhaled. Exposure to these gases results in γ-ray exposure only. Only trace amounts of iodine were released, none of which was ingested.*

From the foregoing information it can be concluded that the health effects of the TMI accident on the general population are indeed negligibly small.

III. MAIN FEATURES AND CHARACTERISTICS OF CURRENT US NUCLEAR POWER PLANTS

Figures 1 and 2 give schematic representations of a Pressurized Water Reactor (PWR). About 60% of the nuclear power plants operating and under construction in the U.S. are of this type. There are three U.S. manufacturers of this type of nuclear reactor, namely Westinghouse (W), Combustion Engineering (CE), and Babcock & Wilcox (B&W). The TMI plant was built by B&W with Burns & Roe as Architect Engineer.

We shall first describe the general principles involved in a nuclear power reactor. The reactor core contains the nuclear fuel (see Figs. 3 and 4) in the form of uranium oxide pellets, stacked in zircaloy tubes (called fuel cladding), and assembled in bundles (called fuel assemblies). The fission chain reaction in which heavy atoms (essentially uranium-235 and plutonium-239) are split by neutrons, takes place in the core region, thus producing heat. This heat is given off to the coolant (water) which is forced through the core by the primary coolant pumps. The coolant is prevented from boiling in the reactor core by pressurization to a pressure of 2250 psi by means of the pressurizer (Fig. 5). The heat taken up by the coolant is then transported to the steam generators (see Fig. 6), where it is used to produce steam. The cold water leaving the steam generators returns to the core and is there heated up again, etc. The steam produced in the steam generators is used to drive the turbine-generators, thus producing electricity. In order to replace the water that was used up in

*Furthermore, Iodine-131, which is the isotope of primary concern, decays relatively rapidly since it has a half-life of only eight days.

the production of steam, the steam generators are supplied on their secondary side with feedwater by the Main Feedwater System.

Contrary to the PWR, the Boiling Water Reactor (BWR), manufactured by the General Electric Company, is characterized by the fact that the cooling water is allowed to boil in the core (see Fig. 7). The steam produced in the reactor core at a pressure of 1000 psi is directly supplied to the turbine-generator, thus obviating the need for steam generators and a pressurizer. Water derived from the condensing steam in the condensor is returned to the reactor core by the Main Feedwater System.

The fission process in the core results in the generation of large quantities of radioactive fission products. Prevention of the dispersion of these fission products into the environment, thus avoiding exposure of the general public, is one of the primary concerns in nuclear reactor safety. From the very start of the development of nuclear reactors for commercial power production, great emphasis was placed on safety. This concern for safety has taken many aspects, namely (1) emphasis on high quality in design and construction, (2) inclusion of inherent safety-enhancing characteristics, (3) analysis of a large number of postulated equipment failures and their consequences, (4) incorporation of safety systems aimed at coping with postulated failures, and (5) performance of safety-related experiments aimed at improving the understanding of postulated accident sequences and verifying the performance of safety systems.

Among the important inherent safety characteristics of commercial nuclear power plants is the presence of a large fraction of uranium-238 (about 97%) in the fuel, in addition to the fissionable uranium-235 (initially about 3%); this fact completely excludes the possibility of a nuclear explosion (such as a nuclear bomb), due to the characteristic of uranium-238 to capture an increasing fraction of the neutrons as the fuel temperature increases (Doppler effect). Thus, in case of a postulated accident resulting in a substantial power rise of the reactor, the rate of the neutron chain reaction would be inherently limited.

Another important safety feature is the fact that all commercial nuclear power plants are equipped with a minimum of three clearly defined physical barriers against dispersion of the fission products, namely the cladding (Fig. 3), the primary coolant pressure boundary (Figs. 1 and 2), and the containment (Fig. 8). Often the containment may have two separate barriers (double containment system). Furthermore, the fuel pellets themselves, consisting of a ceramic material (UO_2) with a high melting point ($\sim 2800^{\circ}C$), have excellent properties for retention of the fission products. Among the main safety systems, which serve to protect the barriers or to mitigate the consequences of breach of a barrier, are the following:

(1) Reactor Shutdown (or Scram) System (RSS),
(2) Emergency Core Cooling System (ECCS),
(3) Containment Cooling System (CCS),
(4) Containment Isolation System (CIS), and
(5) Auxiliary Feedwater System (AFS).

All of these safety systems are designed to actuate automatically, i.e., without operator intervention. The RSS will automatically terminate the nuclear chain reaction by inserting the control rods (Fig. 1) into the reactor core if any of a number of safety-related variables exceed their safe limits. (The control rods contain a material -- boron, cadmium, or other -- that has the characteristics of capturing neutrons quite effectively, thus eliminating them from the chain reaction.)

The need for the ECCS (Figs. 9 and 10) derives from the fact that generation of heat in the fuel will not immediately stop following termination of the fission process: The fission products generated in the fission process are subject to radioactive decay which continues to produce heat. Immediately following reactor shutdown, the rate of decay heat production is approximately equal to 7% of the original thermal power of the nuclear station. This decay heat production will decline with time to less than 2% in one hour, to about 0.4% in one day, to about 0.2% in one week, and to slightly above 0.1% in one month (see Fig. 11). It is therefore important to maintain adequate cooling of the fuel immediately following reactor shutdown. The ECCS is provided to ensure such cooling for the case that the reactor were to lose its coolant (commonly referred to as a Loss-of-Coolant Accident or LOCA) due to a break in the primary coolant system or due to equipment malfunction, as was the case in the TMI accident where a relief valve stuck open.

The purpose of the CCS is to remove heat from the containment following a LOCA in order to avoid overpressurization of the containment. The CCS includes the Containment Spray System (CSS) which, in addition to containment cooling, has the function of reducing the amount of airborne fission products in the containment, thus limiting the release of radioactive material to the environment following a loss-of-coolant accident.

The purpose of the AFS is to provide feedwater for the case that the Main Feedwater System were to fail, thus maintaining normal cooling of the core. This safety function will be discussed in greater detail in the following, since its failure was the primary cause of the TMI accident.

IV. DESCRIPTION OF THE TMI ACCIDENT

Table 2 gives a preliminary version of the chronology of the TMI accident as released by the NRC. As is noted, the initiating event was the loss of the main feedwater supply. Under normal circumstances, this event should not have led to any difficulty: The AFS should have injected feedwater into the steam generators, allowing the nuclear power plant to stay on line. Unfortunately, due to a human oversight, all lines of the AFS were valved off so that no feedwater was injected into the steam generators, notwithstanding the fact that the AFS pumps did start up automatically as intended. The decrease of feedwater in the steam generators caused a rise of the pressure in the reactor coolant system (RCS) resulting in the shutdown of the reactor (reactor scram or trip) and the opening of a relief valve on the pressurizer. Since the relief valve did not reclose (as it should have), primary coolant continued to be released from the RCS causing a continued reduction of RCS pressure, which eventually caused automatic activation of the ECCS at 1600 psi. This should normally have allowed prevention of further deterioration of the accident. However, at this point, probably as a consequence of a faulty indication of the pressurizer's level measurement system, the operators deactivated the ECCS. Since the relief valve on the pressurizer continued to release coolant, the RCS further depressurized until at ~1350 psi it reached saturation conditions and started steam formation (flashing) throughout the RCS.

We shall not repeat here in detail what is described already in Table 2, but shall limit ourselves to the main events. At about 7½ minutes into the accident sequence, the reactor building sump pumps came on automatically. At this point in time, the Containment Isolation System had not yet been activated (the TMI plant requires a 4-psi overpressure in the containment for automatic actuation of the CIS). As a consequence of this situation, part of the radioactive primary coolant, which had been released through the pressurizer relief valve and was spilt onto the containment floor, was transferred to the Auxiliary Building. Since the Auxiliary Building is not part of the airtight containment system, release of radioactive gases (mainly xenon and krypton) could now take place. Due to the flashing in the RCS, the reactor cooling (RC) pumps started to cavitate. Since this condition is harmful to the pumps, the operators shut off the RC pumps approximately 1½ hours into the accident. At this point in time, cooling by natural circulation might have prevented damage to the fuel, since feedwater supply to the steam generators had been restored at about eight minutes into the accident. However, the presence of steam bubbles in the RCS, possibly combined with an unfavorable temperature distribution, probably prevented initiation of natural circulation. In any case, after shutting off the RC pumps, the core heat-up transient started, causing fuel damage, metal-water reaction, and production of hydrogen. This hydrogen appears to have been the primary source of the gas bubble in the reactor vessel, which has caused some considerable concern.

In the days that followed, core cooling was reestablished (using one RC pump and one steam generator, the gas bubble was transferred out of the reactor vessel, and preparations were made for establishing a stable long-term cooling mode.

V. DESIGN DIFFERENCES BETWEEN PWRs OF DIFFERENT MANUFACTURERS

As mentioned earlier, the TMI plant was designed and built by B&W, with Burns & Roe as Architect Engineer. There exist substantial differences between the PWRs of different manufacturers (B&W, CE, and W). These differences are such as to make the occurrence of a TMI-type accident for PWRs designed and built by either Westinghouse (W) or Combustion Engineering (CE) quite improbable. We shall limit ourselves in the following primarily to a discussion of those design features of Westinghouse PWRs that would affect the sequence of events in case of an initiating event similar to that for the TMI accident. (Nuclear power plants of the PWR type operating or under construction in Illinois are all designed and built by Westinghouse.)

The initiating event of the TMI accident, complete loss of the main feedwater supply, may have various causes and could possibly also occur on a W-designed plant. However, the subsequent sequence of events would have evolved quite differently for the following reasons:

(1) The <u>manual valves</u> on the auxiliary feedwater system of W-designed PWRs are <u>locked open</u>, and are <u>used</u> only for maintenance procedures, <u>not for</u>

periodic testing (as is the case for B&W designed plants). The probability of the entire auxiliary feedwater system being valved out is therefore very low for a Westinghouse nuclear power plant. If the auxiliary feedwater system starts operating as intended following a loss of the main feedwater supply, no problem arises, and the accident sequence is terminated.

(2) Westinghouse steam generators are not of the once-through type, as is the case for B&W steam generators. Consequently, the inventory of water on the secondary side is considerably larger for Westinghouse steam generators than for B&W steam generators. Thus dry-out of the steam generators and heat-up of the primary system would have occurred considerably later in a Westinghouse PWR if one assumes that the auxiliary feedwater system did not take over (as was the case in the TMI accident).

(3) Westinghouse steam generators have a reliable secondary-side level indication, since they are not of the once-through type. Low level on two or more steam generators (i.e., 25% on the narrow-range level instruments) will automatically cause reactor trip (turbine trip would already have occurred on trip of all main feedwater pumps). Early reactor trip, reducing the reactor thermal power rapidly to 7% of the value of full power, would also extend the time prior to dry-out of the steam generators in case of a postulated failure of the auxiliary feedwater system. Back-up reactor trip signals for this plant condition are (a) low-low level (i.e., 10% on the narrow-range level instruments) on any steam generator, (b) pressurizer high level, (c) pressurizer high pressure, and (d) high primary coolant temperature (over-temperature ΔT).

(4) Westinghouse reactors do not have reactor power run-back following turbine trip as is the case for B&W reactors (in case of reactor power run-back, the reactor power is slowly reduced at a predetermined rate, as opposed to a rapid reduction as is the case for a reactor trip). Thus, in a B&W reactor, the thermal power stays up while in a Westinghouse reactor the power would have been quickly reduced. The signal

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in the TMI plant that finally caused reactor trip was high pressurizer pressure. On a Westinghouse plant, there would have been four or five signals preceding this signal [i.e., (1) turbine trip at power level > 10% of nominal, (2) SG low level, (3) SG low-low level, (4) pressurizer high level, and (5) pressurizer high pressure].

(5) Loss of the main feedwater supply combined with failure of the auxiliary feedwater system for the Westinghouse PWR would also result in a rise of temperature and pressure of the RCS and relief of primary coolant through the pressurizer relief valve. Upon depressurization of the RCS, the pressurizer relief valve did not reclose in the TMI accident. This is a failure that could also occur in a Westinghouse PWR. Such a small-scale loss-of-coolant accident would result in automatic actuation of the ECCS on a signal made up by coincident low level and low pressure in the pressurizer. If this signal were not to occur due to the swell of the pressurizer level* during a depressurization transient, the operator has about 50 minutes to actuate the ECCS manually without uncovering the core.

(6) Actuation of the ECCS in Westinghouse PWRs automatically results in actuation of the Containment Isolation System (CIS). This is an important difference with the TMI plant, where the CIS is not designed to actuate on ECCS. In the TMI accident, containment isolation occurred only five (5) hours into the accident at a containment overpressure of 4 psi. Because containment isolation took place so late in the TMI accident sequence, the containment sump pump was allowed to transfer radioactive primary coolant to the auxiliary building. This would not have taken place for a Westinghouse-designed plant.

The above gives some of the primary differences between Westinghouse and B&W PWRs. Similar differences exist between PWRs manufactured by Combustion Engineering Co. and B&W. It may be concluded from this that the probability of recurrence of a similar accident sequence as that of TMI is extremely small.

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*In the newer designs of the Westinghouse PWRs, pressurizer low level is not required for the ECCS actuation signal; it is now proposed to also eliminate pressurizer low level for ECCS actuation from all W-designed PWRs.

VI. FINDINGS AND RECOMMENDATIONS

The Committee's Preliminary Report, submitted on 4/19/79, had as its primary conclusion that, in view of the substantial differences existing between the operating nuclear power plants in Illinois and the TMI nuclear reactor, there does not exist any ground for shutting down nuclear power plants in Illinois. (See Attachments 1 and 2.)

Although the investigations of the Committee have not resulted in any findings requiring the shutdown of nuclear power plants in Illinois, the Committee has identified areas where improvements are desirable.

During its investigations, the Committee placed primary emphasis on currently operating nuclear power plants in Illinois. However, some of the recommendations the Committee wishes to make also pertain to plants under construction, and some recommendations pertain solely to future plants.

The following section gives the principal recommendations. For the purpose of easy reference, the Committee's recommendations have been subdivided into four categories, namely (A) General and/or Policy Aspects, (B) Operating Staff and Operating Procedures, (C) Technical Aspects, and (D) Long-Term Considerations.

In view of the limited time that was available to the Committee for its investigation, many of the recommendations are still of a preliminary nature, requiring further, more detailed, study prior to possible implementation.

A. RECOMMENDATIONS PERTAINING TO GENERAL AND/OR POLICY ASPECTS

Concerning the State of Illinois

A.1 It is recommended that the State of Illinois develop Emergency Plans, meeting NRC concurrence requirements, for each site of a major nuclear facility.

A.2 It is recommended that the State of Illinois review the desirability of becoming an Agreement State.

A.3 It is recommended that the State of Illinois conduct Independent Safety Audits, as necessary, covering major nuclear facilities (operating and under construction) within its boundaries; these independent audits should also include a review of the performance of relevant Regulatory Agencies. It is further recommended that these audits be carried out under the responsibility of the Illinois Commission on Atomic Energy, and/or other appropriate State Agencies.

A.4 It is recommended that a better coordination be established between the State of Illinois and the U.S. Nuclear Regulatory Commission (NRC). A move in this direction might be the assignment of a Representative from the NRC Office of State Programs to the NRC Region III Office in Glen Ellyn, Illinois.

Concerning Federal Agencies

A.5 It is recommended that the NRC adopt quantitative health and safety goals and criteria for use in all facets of its regulatory process. Such goals and criteria shall be compatible with health and safety goals used for regulation of other relevant aspects of our technological society. Specifically, the permissible risk levels, to be adopted for the nuclear energy technology, shall in general be smaller (but not to an excessive degree) than those applied in alternative energy-production technologies (dams for hydro-electric power generation, fossil-fired stations, solar energy, etc.) and with those applied in the manufacture, storage, and disposal of other hazardous materials. Such NRC safety goals and criteria shall be developed by NRC under the auspices of bodies such as the National Academy of Sciences, or the National Academy

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of Engineers, and shall be commented on and approved by the U.S. Congress. The Committee strongly supports a recommendation of a similar nature made by the NRC Advisory Committee on Reactor Safeguards (ACRS) contained in the ACRS letter, Max W. Carbon to Joseph M. Hendrie, dated May 16, 1979 (Attachment 3).

A.6 It is recommended that the NRC and the Department of Energy (DOE) extend and reinforce their capabilities and programs in the area of probabilistic risk assessment for nuclear and other energy technologies, and that they review and re-evaluate the potential contribution of operator error to the overall risk of nuclear power plants, in the light of the TMI accident.

A.7 It is recommended that NRC and DOE review their nuclear reactor safety programs so as to place greater emphasis on anticipated events and incidents of moderate and low probability (i.e., plant conditions I, II, and III), and less emphasis on hypothetical limiting faults of extremely low probability (i.e., plant condition IV). For too long both NRC and DOE have displayed a lack of perspective in this regard, having allocated most resources to the study of highly improbable limiting faults.

A.8 It is recommended that NRC review its extensive and complex body of regulatory requirements and guidelines (General Design Criteria, Regulatory Guides, Standards, Technical Branch Positions, etc.) in the light of the results of probabilistic risk analyses; it is recommended that areas, where possible changes may be introduced, be identified in order to obtain a more equal distribution of risk over the entire spectrum of potential accident-initiating events. It is further recommended that NRC strive to simplify its body of regulatory requirements so as to make it less specific to one reactor type. It is suggested that a review of regulatory requirements existing in other countries (Canada, UK, France, West Germany) may be helpful in this respect. It is noted in this connection that Canadian regulatory requirements have followed a probabilistic approach from their inception.

A.9 It is recommended that NRC implement a closer coordination between its various branches; in particular, it is recommended that this be done between the Office of Reactor Regulation and the Office of Inspection and Enforcement.

Concerning Illinois Utilities

A.10 It is recommended that the Illinois utilities require, in their dealings with the reactor manufacturers, that greater emphasis be placed on adequate protection against anticipated events and incidents of moderate and low probability (i.e., plant conditions I, II, and III; see also Recommendation A.7).

A.11 It is recommended that Illinois utilities, operating nuclear power plants, institute a greater managerial separation between the operating staffs of nuclear power plants and those of fossil-fueled power plants. This is in order to emphasize the substantial differences between power plants of the two types.

A.12 It is recommended that each Illinois utility, operating or constructing nuclear power plants, institute an internal Nuclear Reactor Safety Review Committee, charged with the responsibility of performing regular reviews of all aspects bearing on the safety of the operation, maintenance, design, and construction of nuclear power plants, as well as of operator training and performance. These Safety Review Committees shall have an advisory function, shall report directly to Top Management, and shall prepare regular safety review reports. These Committees shall be appointed by the utilities, and shall primarily, though not exclusively, consist of company employees knowledgeable in the area of operation of nuclear power plants, but not currently involved in such activities; the Committees may also have members which are not company employees. Most Members of these Committees are to be appointed for part-time duty and for sufficiently long time periods (e.g., staggered three-year appointments) to provide adequate continuity.

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A.13 It is recommended that each Illinois utility, operating nuclear power plants, establish a formal mechanism for the review of Licensee Event Reports (LERs); these reviews should cover both those LERs generated within the companies and those generated by other utilities. It is suggested that the Nuclear Reactor Safety Review Committees mentioned under A.12 may be charged with the review of LERs. It is further recommended that a formal mechanism be established for incorporation of the "lessons learned" from LER review into the operating procedures and operator training.

A.14 It is recommended that an Emergency Operation Center be established in the vicinity of each nuclear power plant. It is furthermore recommended that consideration be given to Alternate Emergency Operation Centers, to serve in case the primary centers were to become unavailable. Such centers are to be jointly used by State/Utility/NRC/Local Government representatives in case of a nuclear incident. These centers shall be maintained at all times in an operable condition and shall be provided with adequate and reliable communication facilities.

A.15 It is recommended that adequate and reliable Back-up Communication Systems be provided for each nuclear power plant, to serve in case of partial or total failure of the normal commercial telephone system.

A.16 It is recommended that the need for, and advisability of, installing improved/additional Off-site and On-site Radiation Monitoring Systems be reviewed for each nuclear power plant. Such monitoring systems should be aimed at providing fast and accurate information in case of a nuclear incident. It is furthermore recommended that a clear definition of the objectives (e.g., amount and nature of the data, means of transmission of the data, etc.) be prepared, and that a study be performed concerning the various alternative solutions for achieving these objectives. Cost/benefit evaluations of the alternative solutions should also be made. The final proposal should clearly define the interfacing responsibilities of State, utility, and NRC with respect to ownership, operation, and maintenance of these radiation monitoring systems. It is recommended that the State of Illinois and Illinois nuclear utilities continue their current plans for a pilot project along the above lines, initially for a single station.

Concerning Interaction with the General Public

A.17 It is recommended that the State, NRC, and the utilities make adequate provisions and arrangements in order to avoid issuing conflicting public statements, which could cause public confusion in case of a nuclear incident.

A.18 It is recommended that representatives of the State, NRC, or the utilities, when making public announcements following a nuclear incident, provide sufficient information so as to allow the general public to place the actual risk in the proper perspective. The data provided should be explained in laymen's language. As an example, the significance of radiation doses should be explained by making comparisons to doses due to e.g., natural background (and its regional variations), air travel, use of X-rays and radioisotopes in medical treatments, etc. Also, factual information concerning radiation types (α, β, and γ) and radiation sources (e.g., noble gasses, iodine, etc.) should be provided.

A.19 It is recommended that representatives of the State, NRC, and/or the utilities, when publicly announcing a position which later turns out to be erroneous (e.g., due to misjudgement or lack of reliable information), correct this position publicly, swiftly, and with adequate emphasis, as soon as additional reliable information warrants doing so. (Example: During the TMI accident, NRC caused great public concern with its announcement about a large bubble consisting of an explosive mixture of hydrogen and oxygen. It turned out that the bubble was neither large nor explosive. Although this error was known to NRC shortly after the announcement was made, NRC did not correct its mistake publicly and with sufficient emphasis until required to do so in Congressional hearings.)

B. RECOMMENDATIONS PERTAINING TO OPERATING STAFF AND OPERATING PROCEDURES

B.1 It is recommended that the need for, and/or advisability of, appointing individuals (with job titles to be determined later), having in-depth knowledge of nuclear power plants and analytical ability (e.g., degree in engineering, or equivalent), be reviewed. A minimum of one of these individuals should be present during each operating shift. These individuals should have a reporting status to the Corporate Headquarters of the utility, and should serve in an advisory capacity to the Shift Supervisor/Engineer; they should not be responsible for the routine operation of the nuclear power plant. Their primary responsibilities under normal conditions may include the checking of control-room operations, the checking of safety-related systems, and the interaction with the NRC Resident Inspector. In case of an incident, these individuals may be called upon to assume primary responsibility during the incident and during the recovery operations, acting, however, still through the Shift Supervisor/Engineer.

B.2 It is recommended that the need for, and/or advisability of, a general upgrading of the training and re-training levels of operators be reviewed.

B.3 It is recommended that the training program for the operating staff place adequate emphasis on the importance of adherence to Operating Procedures and Technical Specifications; in particular, the training program should inform the Operating Staff about the potential accidents, and their consequences, that could be caused by non-compliance. Furthermore, it is recommended that disciplinary actions, to be imposed by the utilities, as appropriate, in case of a clear violation of Operating Procedures, be clearly explained in the training program.

B.4 It is recommended that the utilities make available upon request Statistical Data concerning the Performance of the Operating Staff during training and re-training programs to the Committee conducting the Independent Safety Audits (Recommendation A.3), if implemented.

B.5 It is recommended that the utilities institute a clear Procedure for the Review of Suggestions from, and/or Dissenting Opinions of, members of the Operating and Technical Staff in the area of nuclear safety. It is suggested that the internal Nuclear Reactor Safety Review Committee (Recommendation A.12) may be charged with the execution of this review procedure. It is further recommended that the utilities actively encourage suggestions from the Operating and Technical Staff in the area of nuclear safety.

B.6 It is recommended that the utilities institute a well-defined Incentive/Merit Program in the area of nuclear safety for the Operating Staff.

B.7 It is recommended that the Operating Procedures and Technical Specifications be reviewed relative to the conditions under which the operating staff may be required to override/augment automatic safety-related functions. It is also recommended that Operator Training be reviewed in this respect.

B.8 It is recommended that Operator Instructions and Training be reviewed relative to Periodic Testing, so as to prevent leaving safety-related systems in a degraded state of operability following periodic tests (e.g., leaving valves in the wrong status).

B.9 It is recommended that Supervisory and Management Procedures be reviewed with the aim of providing adequate checks on operator actions.

B.10 It is recommended that the need for, and/or advisability of providing an Improved Tagging System for indicating system status on the control board be reviewed. Such an improved tagging system should preclude the possibility of covering up status lights, which may give important safety-related information.

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C. RECOMMENDATIONS PERTAINING TO TECHNICAL ASPECTS

C.1 It is recommended that the Pressurizer Level Signal be eliminated in PWRs from all logic circuitry capable of actuating safety-related systems.

C.2 It is recommended to provide improved Containment Isolation. In particular, it is recommended to provide Containment Isolation with a lock-in feature (i.e., requiring positive operator action to defeat it), to be actuated simultaneously with the Emergency-Core-Cooling and Safety-Injection Systems.

C.3 It is recommended that the need for, and/or advisability of, a more reliable Pressure Relief System on PWRs be reviewed (e.g., the Pilot-Operated Relief Valve, or PORV, is connected to the primary coolant pressure boundary; it may be desirable that it be safety-grade).

C.4 It is recommended that the various Safety-related Signals be reviewed in order to determine the need for, and/or advisability of, using primary signals rather than derived signals. (Example: In case of the PORV it may be desirable to use a valve-position signal rather than a signal derived from the solenoid.)

C.5 It is recommended that the need for adequate Venting Capability of the primary cooling system be reviewed for PWRs, including installation of remote-control motor-operated valves for this purpose.

C.6 It is recommended that the feasibility, and/or advisability, of providing Level Measuring Capability on pressure vessels of PWRs be reviewed.

C.7 It is recommended that the need for, and/or advisability of, installing a continuous Monitoring System for the Degree of Subcooling of the coolant (i.e., T_{sat}-T) in the primary heat transport system be reviewed for PWRs.

C.8 It is recommended that the need for, and/or advisability of, providing Remote-control Capability and clear Status Indication for valves with safety-related functions be reviewed.

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C.9 It is recommended that the advisability of eliminating the Lead/Lag Networks in PWRs, used for speeding up the pressurizer pressure signal, be reviewed. This may be achieved by replacing in safety-related logic circuitry, the pressurizer pressure signal with a pressure signal derived from the pressure vessel or the primary system loops.

C.10 It is recommended that the advisability of a greater application of computers in the Control Room be reviewed. These computers could be used for routine status checks of safety and operational systems, for collecting and processing of data, as well as for aiding the operating staff in decision making concerning diagnostic evaluations and the sequencing of corrective actions during an accident.

C.11 It is recommended that the advisability of installing a separate Status Board, indicating the operability of safety-related systems, be reviewed.

C.12 It is recommended that the potential for Degraded Operation of the emergency core cooling and containment spray systems be reviewed, and that remedial measures be taken, if necessary. Such degraded operation could be due to accumulation of debris (e.g., piping insulation material), or vortex formation, in the containment sump.

C.13 It is recommended that the need for, and/or advisability of, providing protection against potential Containment Overpressurization through controlled venting be reviewed.

C.14 It is recommended that the entire range of Man-Machine Interfaces be reviewed for potential improvements. This pertains in particular to the control room layout (with its many recorders, and visual and audible alarm signals) as well as to the check-out procedures for safety-related and operational systems.

C.15 It is recommended that the need for, and/or advisability of, installing additional instrument, monitoring, and sampling systems (other than those recommended under C.6 and C.7) be reviewed for both currently-operating and future plants, in the light of the experience gained from the TMI accident. Such systems should be aimed at providing dependable information during accident conditions, as well as at giving a reliable and detailed record of all major events that took place. Areas of particular interest are the reactor core and the containment.

C.16 It is recommended that the need for adequate, and/or upgraded, environmental qualification be reviewed for safety-related systems (sensors, circuitry, motors, valves, etc.) in the light of the experience gained from the TMI accident.

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D. RECOMMENDATIONS FOR POSSIBLE LONG-TERM CONSIDERATION

D.1 It is recommended that the feasibility, and/or advisability, of adopting a limited number of standard plant designs for future nuclear plants be seriously considered. Such considerations should include cost/benefit analyses, factoring in the risk of freezing plant designs, and the resultant reduced ability to meet individual utility needs. Due consideration should be given to the distinct advantages arising from such standard designs which include shortened NRC licensing review, simplification (standardization) of reactor operator training and economy of plant construction.

D.2 It is recommended that both NRC and the Illinois nuclear utilities give due consideration to on-going industry studies involving the concept of Reactor Operator Training Institute(s) in the private sector.

D.3 It is recommended that Illinois nuclear utilities consider participation in nuclear industry plans concerning the dedication of one or more existing commercial nuclear power plants to research and training purposes.

D.4 It is recommended that Illinois nuclear utilities consider participation in industry programs aimed at reviewing, auditing, and upgrading reactor operating and training procedures.

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As stated earlier, the Committee's investigations did not result in any findings requiring the shutdown of nuclear power plants in Illinois. The probability of any serious accident occurring in Illinois is, and remains, extremely low. It should also be recalled in this connection that the TMI accident did not cause a single fatality, and that the impact of the TMI accident on the public health is negligible. The foregoing recommendations should therefore be placed in the proper perspective in that all technologies are subject to evolutionary development; changes are continually introduced in any technology to make further improvements.

As noted earlier, many of the foregoing recommendations are of necessity, at the time of writing this report, of a preliminary nature, requiring further study before a decision can be reached as to the advisability to proceed with implementation.

It should also be noted that numerous industrial study groups (consisting of representatives from utilities, reactor manufacturers, and research institutes), are addressing at this time potential areas for further improvement. Furthermore, both NRC ("lessons learned program"), as well as the President's Special Committee on TMI, are still conducting investigations concerning the TMI accident. These ongoing studies may in time lead to the identification of other areas where possible improvements could be made.

VII. CONCLUSIONS

The events at Three Mile Island constitute probably the most serious accident to date concerning the US civilian nuclear power program. The accident was caused by a combination of equipment failures, design deficiencies, and human errors. It is important to note, however, that although the economic loss is no doubt considerable, not a single life was lost in the accident, and furthermore that the effects on the health of the general public are negligible. In this respect, the safety record of the civilian nuclear power program has not been tarnished and continues to stand out quite favorably, if compared with other energy-producing technologies (coal, oil, etc.) where fatality rates for workers and the general public due to accidents and air polution are considerable, and where the environmental impact is in most cases much larger.

The TMI accident should be considered as an important point in the evolutionary development of the nuclear industry. The lessons are being learned. All parties concerned (NRC, the utilities, the reactor manufacturers, the architect engineers, independent review groups, etc.) are studying the events that led to the TMI accident, as well as possible changes in equipment and operational procedures that will further reduce the recurrence of similar events to a vanishingly small probability.

What is needed most at this time is avoidance of hasty decisions and simplistic solutions. Above all, it is important to reflect that there is no valid justification for shutting down currently operating nuclear power plants in view of the events at TMI, other than for repair and/or installation of plant improvements, since nuclear reactors have on the whole accumulated a long and good operating record. It is also of interest to recall that accidents in other industries, even if the cause of numerous fatalities (e.g., accidents in coal mines, and in the airline industry) have usually not been a sufficient reason for closing down the entire industry. As regards nuclear power reactors, some improvements in equipment, operating procedures, and operator training may prove to be desirable after further study. Such improvements should be introduced at the appropriate time.

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Table 1

COMPARISON OF COLLECTIVE DOSES TO POPULATION
WITHIN 50 MILES OF THREE MILE ISLAND
NUCLEAR GENERATING STATION

Source	Whole-Body Collective Dose (man-rem)	Average Dose to Individual (mrem/year)
Natural Background		
One year's exposure (FES) (1970 population)	233,000	125
(1980 population)	270,700	
Normal Operation (FES) (1970 population)		
One year's exposure (all sources)	31	0.017
Gaseous effluents	2.05	0.0011
30-year operation	930	0.017
Preliminary Estimate of Accident Dose		
Cumulative through 4/7/79	3,300	1.5

1970 population	1,868,000
1980 census projections	2,165,651

Note: 1 mrem (millirem) = 0.001 rem
FES = Final Environmental Statement

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Table 2

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IE Bulletin 79-05A
April 5, 1979

PRELIMINARY

CHRONOLOGY OF TMI-2 3/28/79 ACCIDENT
UNTIL CORE COOLING RESTORED

TIME (Approximate)	EVENT
about 4 AM (t = 0)	Loss of Condensate Pump Loss of Feedwater Turbine Trip
t = 3-6 sec.	Electromatic relief valve opens (2255 psi) to relieve pressure in RCS
t = 9-12 sec.	Reactor trip on high RCS pressure (2355 psi)
t = 12-15 sec.	RCS pressure decays to 2205 psi (relief valve should have closed)
t = 15 sec.	RCS hot leg temperature peaks at 611 degrees F, 2147 psi (450 psi over saturation)
t = 30 sec.	All three auxiliary feedwater pumps running at pressure (Pumps 2A and 2B started at turbine trip). No flow was injected since discharge valves were closed.
t = 1 min.	Pressurizer level indication begins to rise rapidly
t = 1 min.	Steam Generators A and B secondary level very low - drying out over next couple of minutes.
t = 2 min.	ECCS initiation (HPI) at 1600 psi
t = 4 - 11 min.	Pressurizer level off scale - high - one HPI pump manually tripped at about 4 min. 30 sec. Second pump tripped at about 10 min. 30 sec.
t = 6 min.	RCS flashes as pressure bottoms out at 1350 psig (Hot leg temperature of 584 degrees F)
t = 7 min., 30 sec.	Reactor building sump pump came on.

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Table 2 (continued)

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TIME	EVENT
t = 8 min.	Auxiliary feedwater flow is initiated by opening closed valves
t = 8 min. 18 sec.	Steam Generator B pressure reached minimum
t = 8 min. 21 sec.	Steam Generator A pressure starts to recover
t = 11 min.	Pressurizer level indication comes back on scale and decreases
t = 11-12 min.	Makeup Pump (ECCS HPI flow) restarted by operators
t = 15 min.	RC Drain/Quench Tank rupture disk blows at 190 psig (setpoint 200 psig) due to continued discharge of electromatic relief valve
t = 20 - 60 min.	System parameters stabilized in saturated condition at about 1015 psig and about 550 degrees F.
t = 1 hour, 15 min.	Operator trips RC pumps in Loop B
t = 1 hour, 40 min.	Operator trips RC pumps in Loop A
t = 1-3/4 - 2 hours	CORE BEGINS HEAT UP TRANSIENT - Hot leg temperature begins to rise to 620 degrees F (off scale within 14 minutes) and cold leg temperature drops to 150 degrees F. (HPI water)
t = 2.3 hour	Electromatic relief valve isolated by operator after S.G.-B isolated to prevent leakage
t = 3 hours	RCS pressure increases to 2150 psi and electromatic relief valve opened
t = 3.25 hours	RC drain tank pressure spike of 5 psig
t = 3.8 hours	RC drain tank pressure spike of 11 psi - RCS pressure 1750; containment pressure increases from 1 to 3 psig
t = 5 hours	Peak containment pressure of 4.5 psig
t = 5 - 6 hours	RCS pressure increased from 1250 psi to to 2100 psi

Table 2 (continued)

TIME	EVENT
t = 7.5 hours	Operator opens electromatic relief valve to depressurize RCS to attempt initiation of RHR at 400 psi
t = 8 - 9 hours	RCS pressure decreases to about 500 psi Core Flood Tanks partially discharge
t = 10 hour	28 psig containment pressure spike, containment sprays initiated and stopped after 500 gal. of NaOH injected (about 2 minutes of operation)
t = 13.5 hours	Electromatic relief valve closed to repressurize RCS, collapse voids, and start RC pump
t = 13.5 - 16 hours	RCS pressure increased from 650 psi to 2300 psi
t = 16 hours	RC pump in Loop A started, hot leg temperature decreases to 560 degrees F, and cold leg temperature increases to 400 degrees F. indicating flow through steam generator
Thereafter	S/G "A" steaming to condensor Condensor vacuum re-established RCS cooled to about 280 degrees F., 1000 psi
Now (4/4)	High radiation in containment All core thermocouples less than 460 degrees F. Using pressurizer vent valve with small makeup flow Slow cooldown RB pressure negative

Pressurized water reactor (PWR)

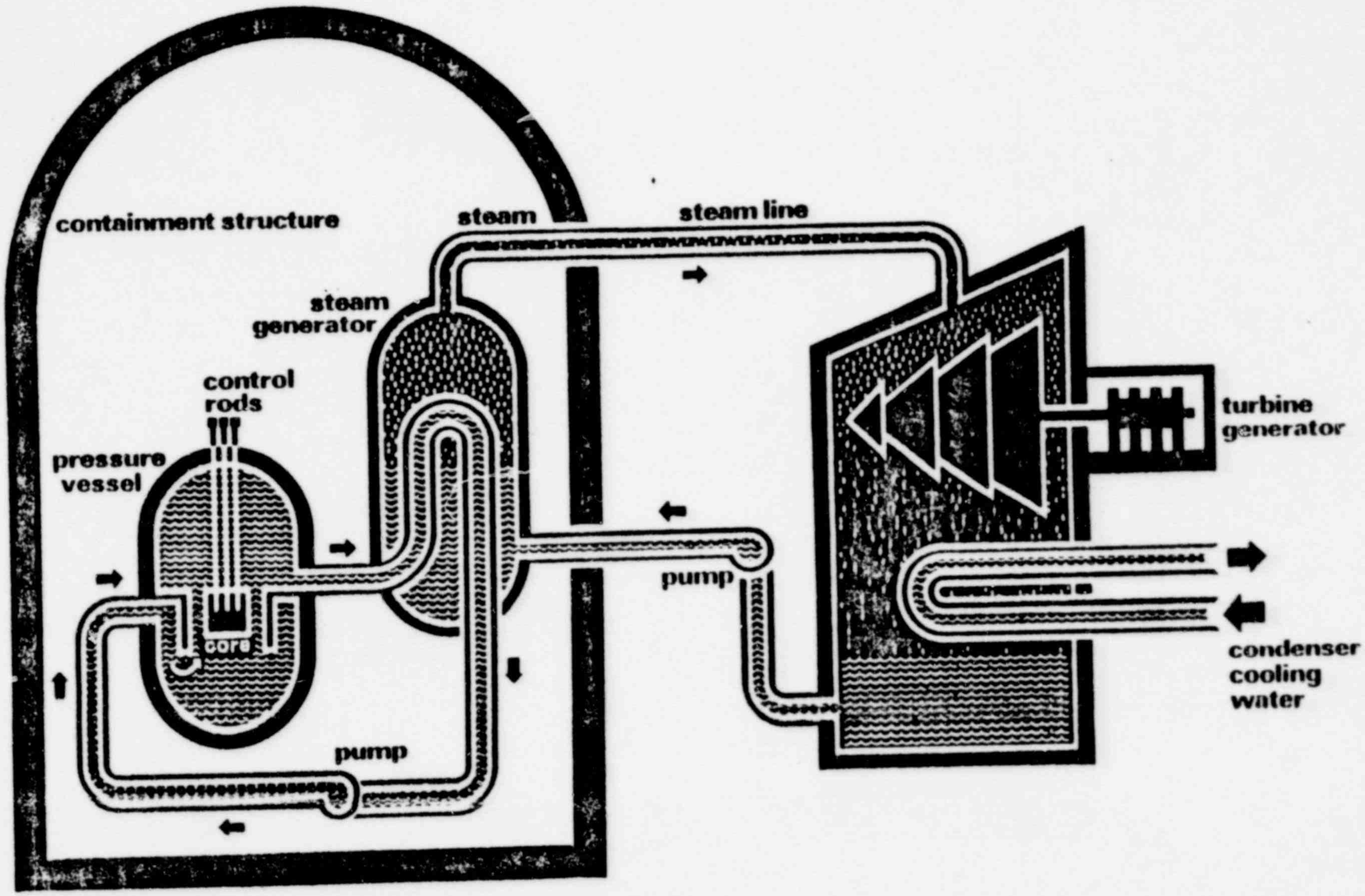

Figure 1

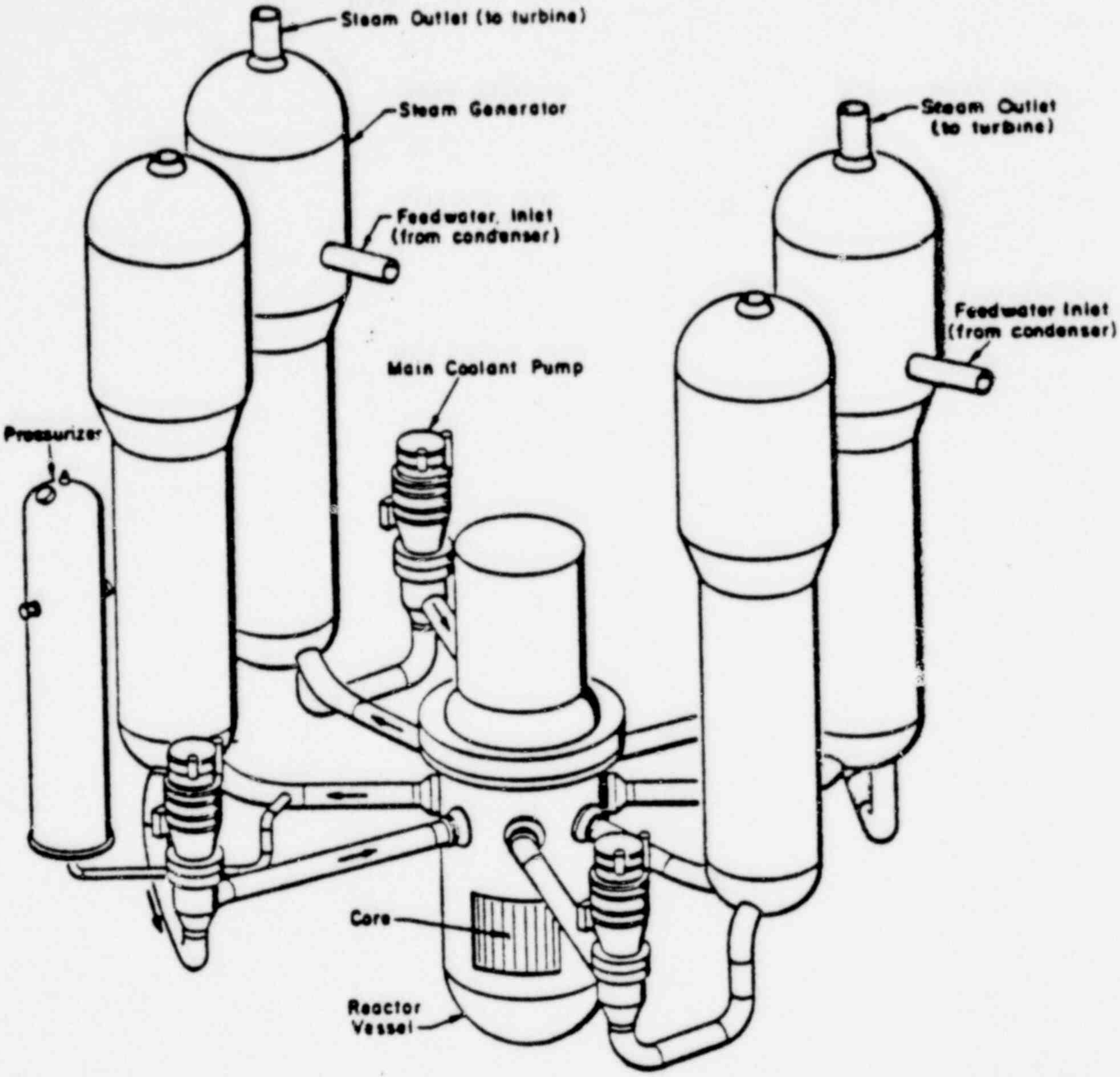

Figure 2: Schematic Arrangement of PWR NSSS.

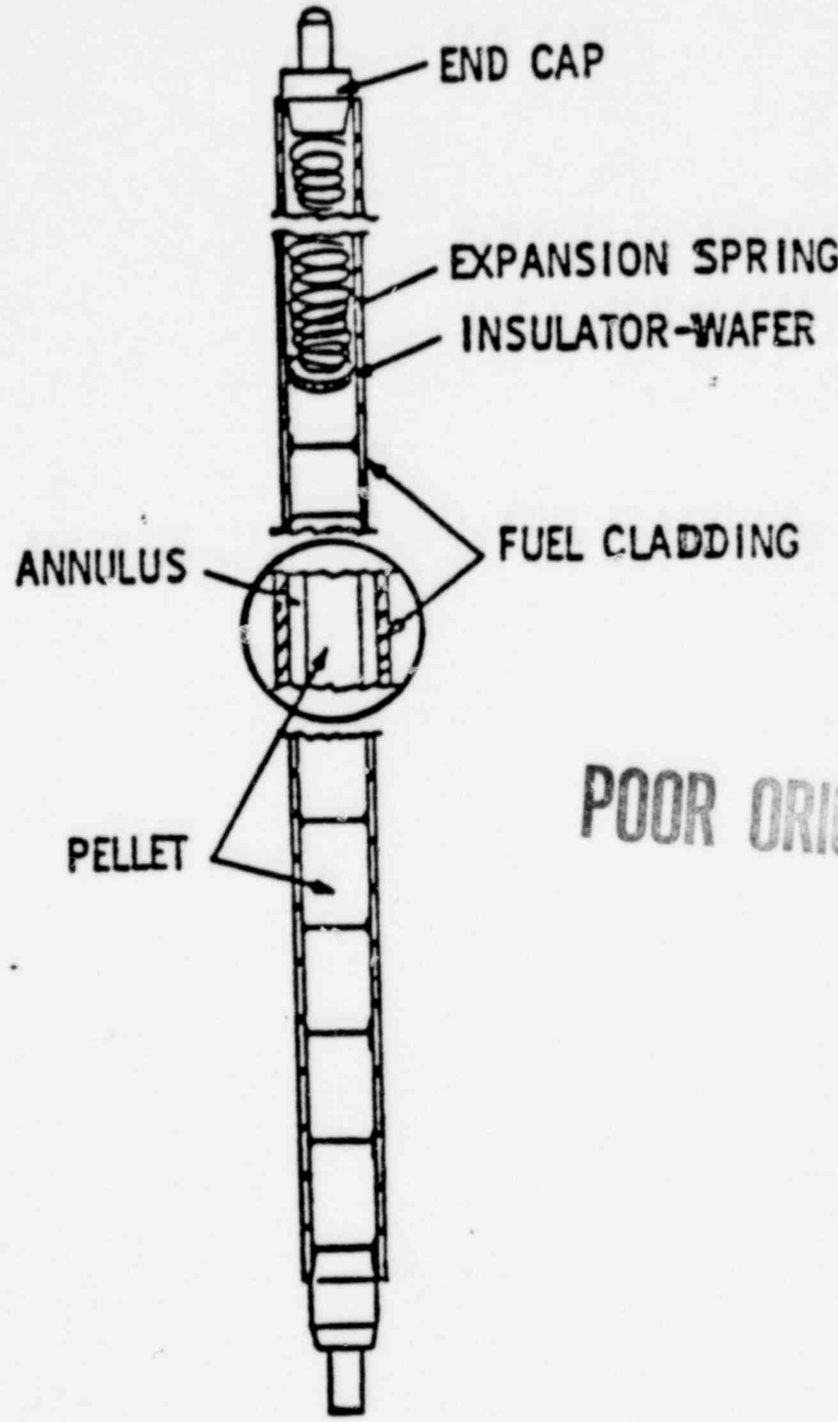

Figure 3: CUTAWAY OF OXIDE FUEL FOR COMMERCIAL LWR POWER PLANT

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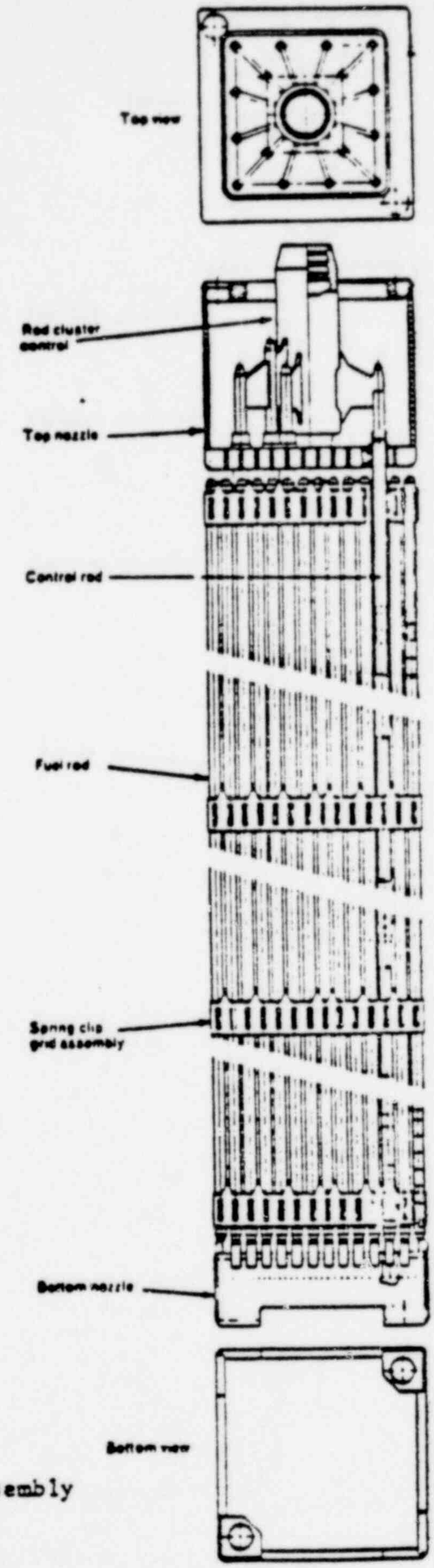

Figure 4: PWR Fuel Assembly

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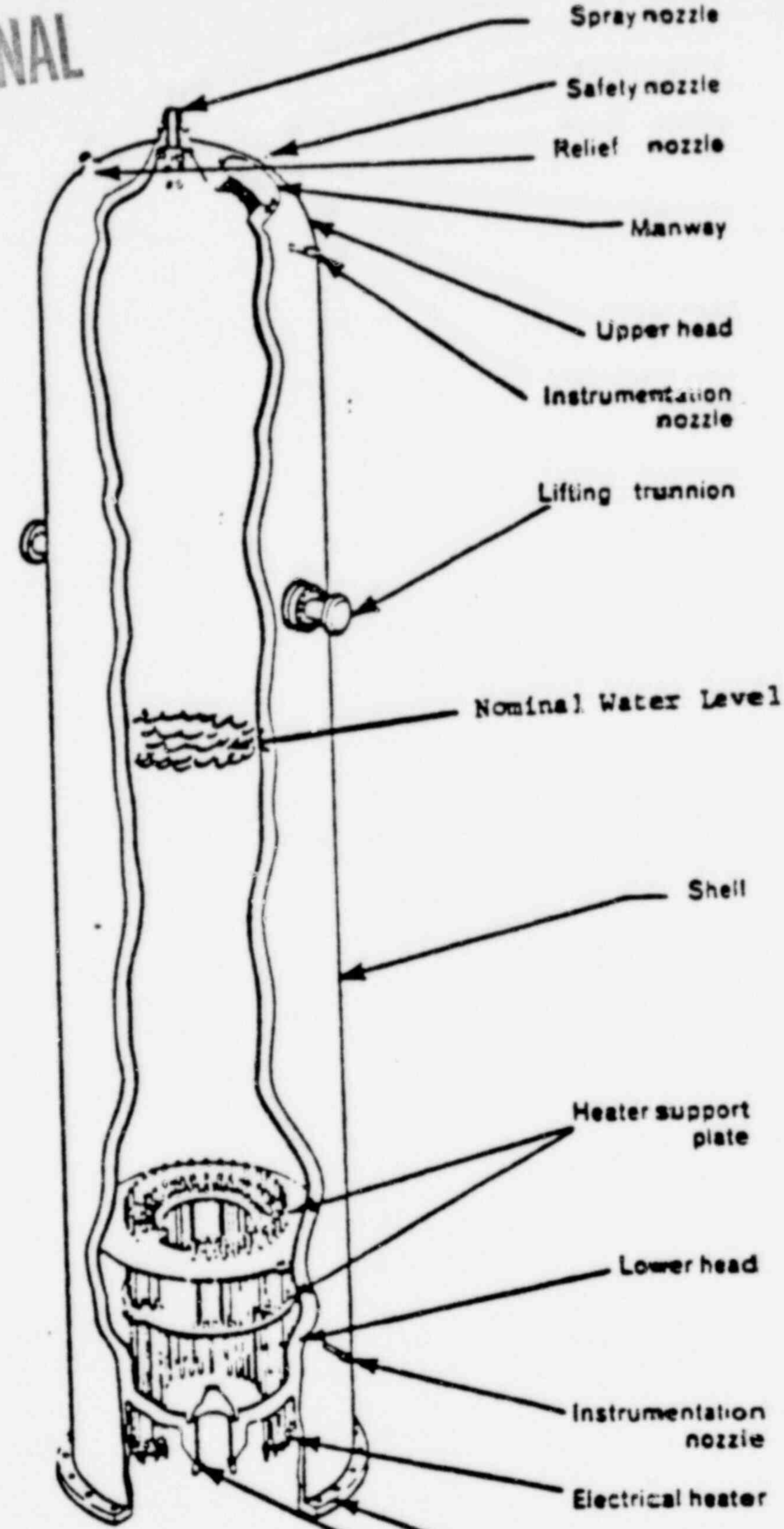

Figure 5: PWR Pressurizer

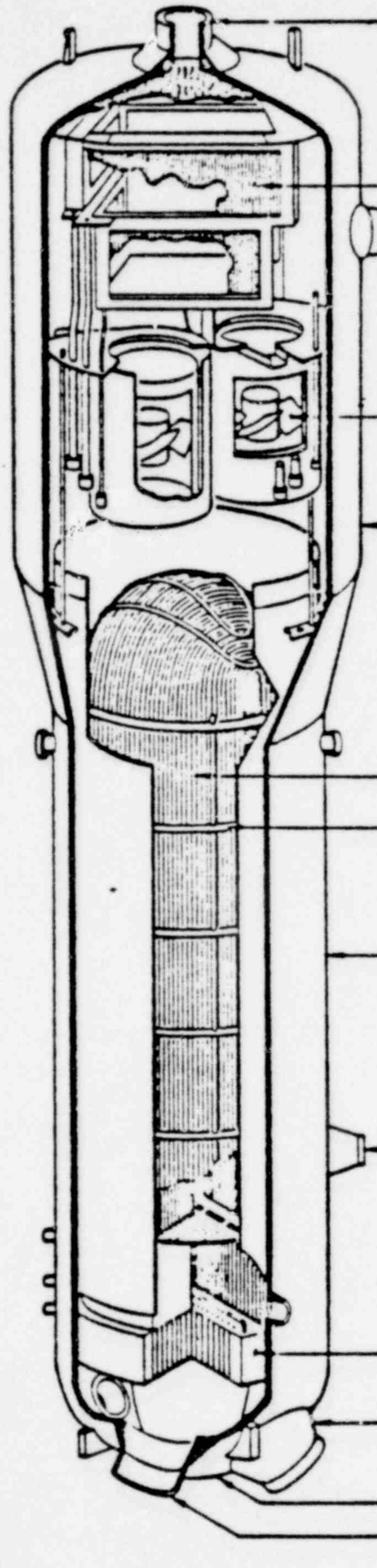

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Figure 6: PWR Steam Generator

Boiling water reactor (BWR)

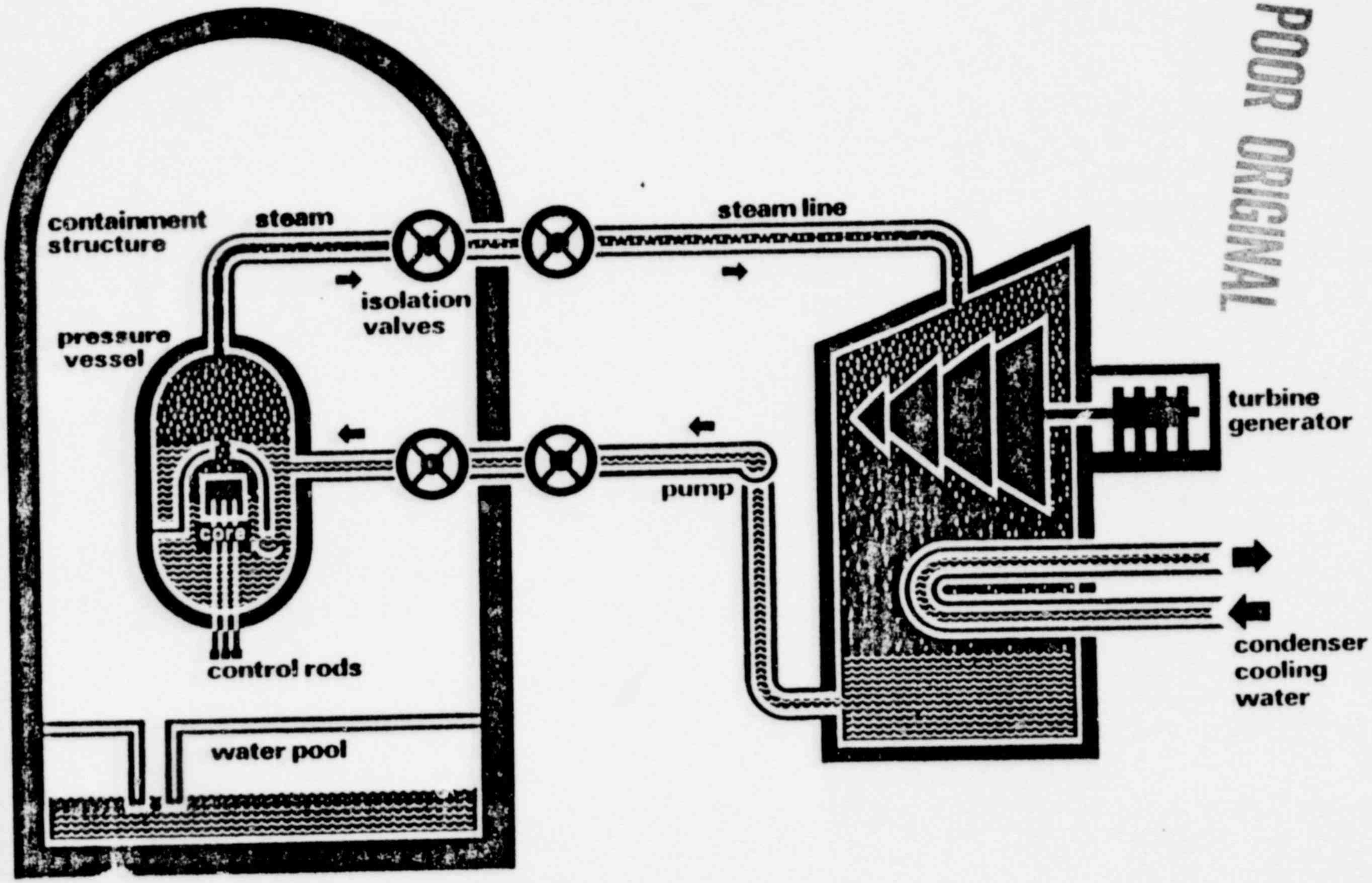

Figure 7

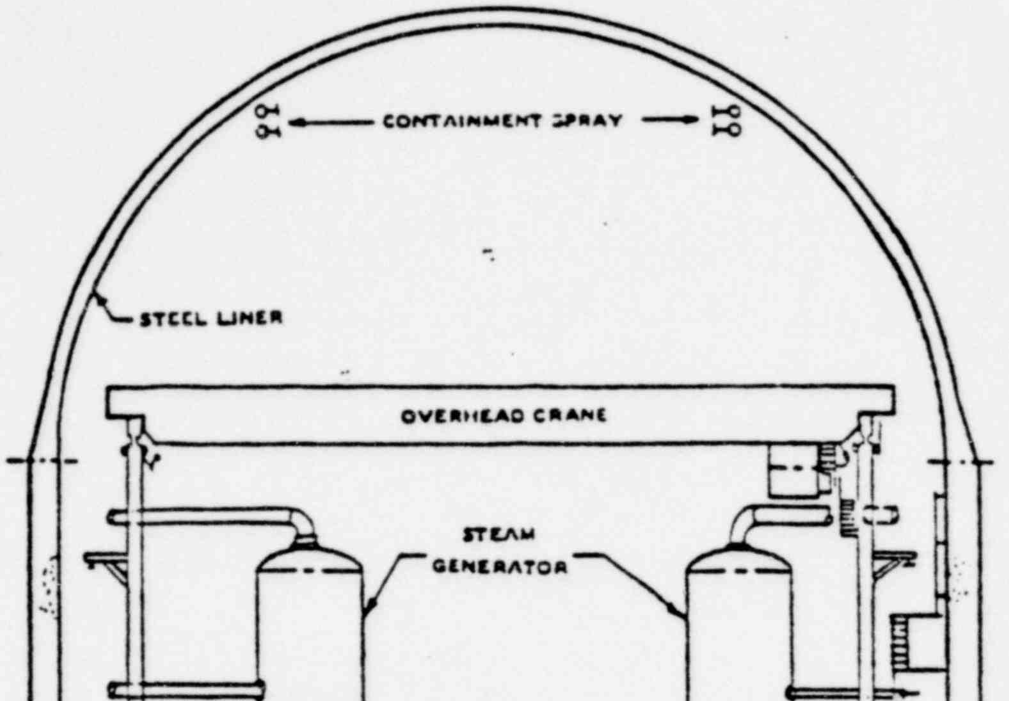

Figure 8: Typical PWR Containment

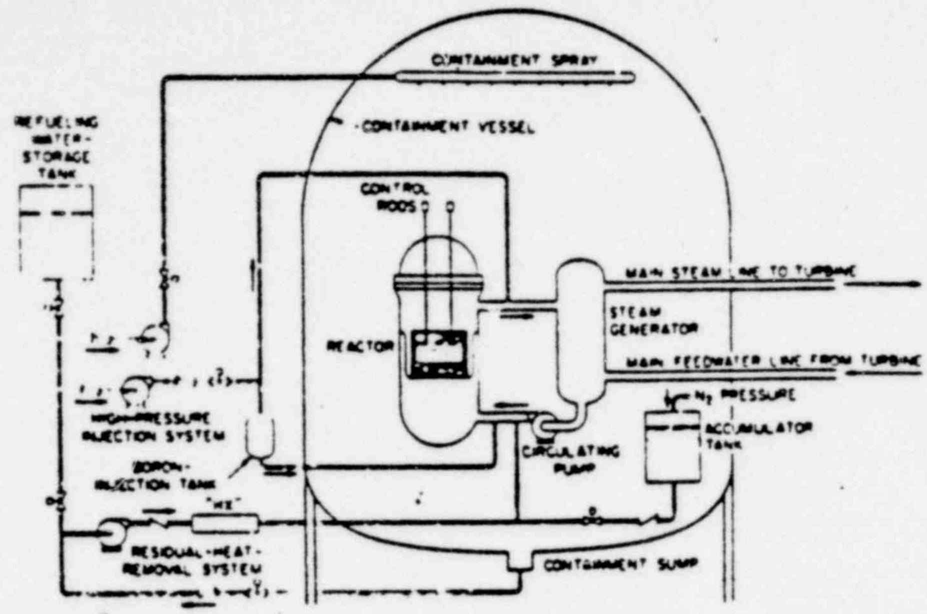

Figure 9: PWR emergency cooling system.

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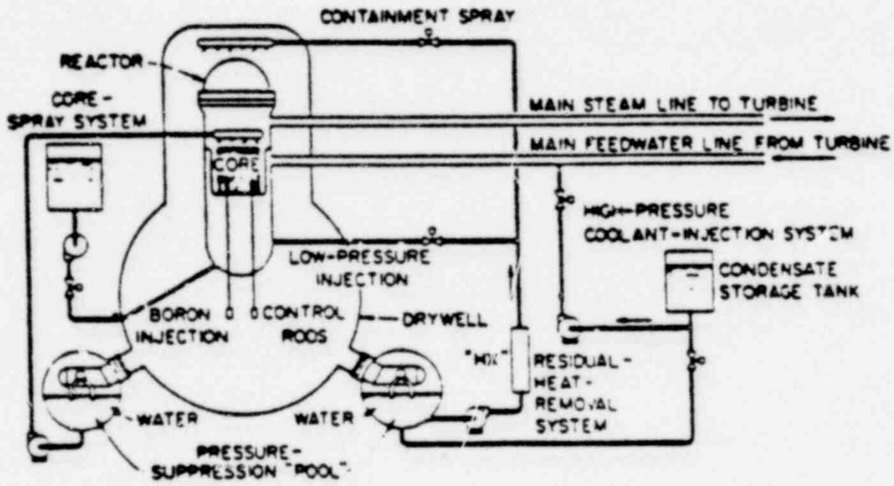

Figure 10: BWR vapor suppression containment system.

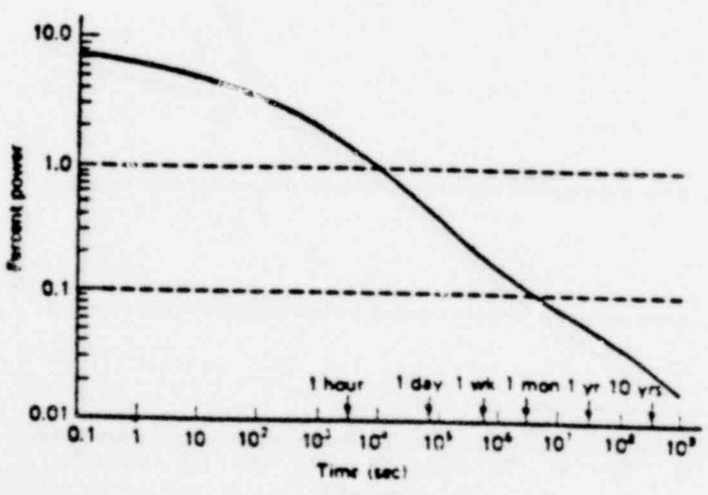

Figure 11: Heat production by decay of fission products.

Attachment 1

ILLINOIS COMMISSION ON ATOMIC ENERGY

110 EAST ADAMS STREET
SPRINGFIELD, ILLINOIS 62706
TELEPHONE: 217 / 782-5057

April 11, 1979

The Honorable James Thompson
Governor of the State of Illinois
State Capitol
Springfield, IL 62706

From: Ad Hoc Nuclear Reactor Safety Committee
Subject: Summary Report of Activities

Dear Governor Thompson:

The Ad Hoc Nuclear Reactor Safety Committee of the Illinois Commission on Atomic Energy has met twice since their appointment by Governor Thompson and Representative George Ray Hudson, Chairman of the Commission. Both meetings were held with Commonwealth Edison Administrative and Technical Staff personnel. The first meeting was on April 4, 1979, and the second day-long session of April 10, 1979, included an inspection of the Zion Nuclear Power Plant. This plant is a Pressurized Water Reactor similar in type to the one at Three Mile Island Station in Goldsboro, Pennsylvania. The Zion facility was designed by Westinghouse while the one at Three Mile Island was designed by Babcock & Wilcox.

The investigating committee has reviewed the Zion plant in the light of events that took place at Three Mile Island. We have noticed that considerable differences exist between the Three Mile Island plant and the Zion facility that would make it extremely improbable that a similar accident would occur at Zion.

The Committee has further consulted with the Nuclear Regulatory Commission representatives on this matter. On the basis of the inspections performed up to now, the Committee sees no reason why the Zion facility should not be allowed to continue operation.

The Committee will visit and inspect the Dresden and Quad Cities reactors in the near future and will provide the Governor with an evaluation of these facilities. A more detailed report will be made in the next few weeks.

Among the items looked into during the inspection of Zion were: Operator Training Safety Systems, Security, and Security Qualifications. A visit was also made to the Westinghouse Training Center where a series of simulated accidents were performed and evaluated.

Refresher training for all personnel was looked at as well as the qualifications necessary for a work crew at the station.

Respectfully submitted,

Gerald R. Day

Gerald R. Day
Executive Director
Illinois Commission on Atomic Energy

GRD:gfs

Ad Hoc Nuclear Reactor Safety Committee

Dr. Philip Gustafson, Chairman
Dr. J. B. van Erp
Dr. George Miley
Professor Daniel Hang
Gerald R. Day

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Attachment 2

ILLINOIS COMMISSION ON
ATOMIC ENERGY

110 EAST ADAMS STREET
SPRINGFIELD, ILLINOIS 62706
TELEPHONE: 217 / 782-5057

April 19, 1979

The Honorable James Thompson
Governor of the State of Illinois
State Capitol
Springfield, IL 62706

From: Ad Hoc Nuclear Reactor Safety Committee
Subject: Preliminary Report on the Safety Status of
Operating Commercial Reactors in Illinois

Dear Governor Thompson:

The Ad Hoc Committee charged with the responsibility of reviewing the nuclear safety status of power reactors in Illinois consists of the following individuals:

Dr. Philip F. Gustafson, Chairman - Argonne National Laboratory
Dr. J. B. van Erp - Argonne National Laboratory
Dr. George Miley - University of Illinois
Professor Daniel Hang - University of Illinois
Gerald R. Day, Executive Director-Illinois Commission on Atomic Ener

In addition the following individuals have acted as liaison, observers or advisers to the Committee:

Gary Wright - Illinois Department of Public Health
John Hasselbring - Illinois Commerce Commission
Professor James Hartnett - University of Chicago

The Committee has concluded that there are no technical reasons why the nuclear power plants now in operation in Illinois should not continue to operate. The Committee bases its conclusion on technical discussions with Commonwealth Edison engineering and operating personnel and with engineering staff from Westinghouse and General Electric. The Committee as a whole or members thereof have toured the Zion, Dresden, and Quad Cities nuclear stations, talked with operating personnel, studied the plant design, safety circuitry, and operating procedures. In addition, the Committee has visited the Westinghouse simulator at Zion and the G. E. simulator at Dresden, witnessed a number of scenarios involving one or more abnormal events, including the designed response and operator actions leading to restoration of normal operation, or reactor shut-down in a safe mode depending upon the actual circumstances.

Region III of the NRC has maintained a record of performance of all operating reactors in the region (which includes Illinois). According to their criteria all commercial reactors in Illinois presentl have good operating records, and have indicated significant improvement c past performance.

Commonwealth Edison, as well as the reactor vendors (G.E. & Westinghouse), have identified areas for improvement both by design/equipment change and/or by new operating procedures. The Nuclear Regulator Commission has issued interim operating guidance and procedures to be followed by all reactor operators pending a full and detailed investigation of the Three Mile Island 2 accident. Of prime importance are the operating instructions to be followed in regard to pressurizer water level in a PWR during transient operations (i.e. a rapid pressure change), and to assure containment isolation during abnormal operating events. The need to automate some auxiliary systems which are now manually operated was also stressed by the NRC.

In summation, the Ad Hoc Committee ha looked at the operating experience and proficiency of the operators of the reactors in Illinois and coupled with the enhanced attention to operating procedures instituted by the utility, we feel that continued operation of the reactors is justified at this time.

After an in-depth study has been completed by the Members of the Ad Hoc Committee and their consultants of all design and operating procedures, a complete detailed report will be made to you. This report will be available on or about July 16, 1979, based on present planning by the Committee. The final report will include the recommendations of the Committee on design features, operational control, and administrative procedures.

Respectfully submitted,

Gerald R Day

Gerald R. Day
Executive Director
Illinois Commission on Atomic Energy

GRD:gfs

Attachment 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 16, 1979

POOR ORIGINAL

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: REPORT ON QUANTITATIVE SAFETY GOALS

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards recommends that consideration be given by the Nuclear Regulatory Commission to the establishment of quantitative safety goals for overall safety of nuclear power reactors. This could be helpful, for example, in developing criteria for NRC actions concerning operating plants. The ACRS recognizes the difficulties and uncertainties in the quantification of risk and understands that in many situations engineering judgment will be the only or the primary basis for a decision. Nevertheless, the ACRS believes that the existence of quantitative safety goals and criteria can provide important yardsticks for such judgment.

The ACRS believes that such NRC goals and criteria should be proposed for comment, not only by the public but by the Congress. Ultimately the Congress should be asked to express its views on the suitability of such goals and criteria in relation to other relevant aspects of our technological society, such as large dams, and manufacturing, storage, or disposal facilities for hazardous chemicals.

The ACRS believes that it is time to place the discussion of risk, nuclear and nonnuclear, on as quantitative a basis as feasible.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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