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State of New Jersey

DEPARTMENT OF ENVIRONMENTAL PROTECTION

TRENTON 08625

OFFICE OF THE COMMISSIONER

October 8, 1976

Mr. L. V. Gossick
Unites States Nuclear
Regulatory Commission
Washington, DC 20555

Dear Mr. Gossick:

As part of our effort to evaluate the need for land use restrictions around nuclear power plant sites in New Jersey (see the enclosed March 10th statement by Commissioner David J. Bardin), we have commissioned a special analysis of the Oyster Creek nuclear power plant. This analysis, done by Peter R. Davis, basically applies the methodology embodied in the U.S. Nuclear Regulatory Commission's Reactor Safety Study to the specific design and operating experience of the Oyster Creek facility. The tentative interim conclusion in Mr. Davis' preliminary report is that the probability of a catastrophic accident over the remaining anticipated lifetime of the Oyster Creek facility is approximately one in one hundred. The main determinant of this result, which we recognize is a greater probability than one might expect, is the probability of the failure of the reactor to shut down after an anticipated transient.

We are circulating this preliminary draft for technical review to a small number of individuals and agencies, of which you are one. We would appreciate receiving any comments you may have by November 12th. To speed up this process, I would appreciate it if you would send a copy of your comments to Mr. Davis at 1935 Sabin Drive, Idaho Falls, Idaho 83401.

Sincerely,

Glenn Paulson

Glenn Paulson, Ph.D.
Assistant Commissioner

GP/vp
Enclosure: An Investigation of Hypothetical Catastrophic Accidents in the Oyster Creek Nuclear Power Plant - Preliminary Draft, September, 1976.

cc: Mr. Peter R. Davis
Mr. Ben Rusche
Mr. Saul Levine

E/37

Let's protect our earth

NEW JERSEY DEPARTMENT OF
ENVIRONMENTAL PROTECTION

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NEWS

David J. Bardin, Commissioner
Wes Denman, Public Information Officer

(STATEWIDE)
No. 76/231

Immediate release:
March 10, 1976

BARDIN CALLS FOR LAND USE CONTROLS IN VICINITY OF NUCLEAR POWER PLANTS

TOMS RIVER--Environmental Protection Commissioner David J. Bardin today told a New Jersey Senate committee that "the time has come for state and local government to initiate practical land-use controls around nuclear power plants."

Testifying here before the Committee on Energy and the Environment, chaired by Senator John F. Russo, Bardin said nuclear power plants will reduce New Jersey's reliance on fossil-fueled plants and help minimize the air pollution in the state from sulfur-containing oil and coal.

"This tradeoff is accompanied by other new risks related to radiation associated with nuclear power," Bardin said. But he added that in contrast to any other recent technological developments, "nuclear power is probably the most carefully scrutinized regulated technology in the United States today."

According to Bardin, "the beehive offers a rough analogy: It's useful and pretty safe. But bees can sting. A stinging swarm can kill. So the beekeeper takes precautions and the rest of us keep our distances. We don't put the kids' sandbox or the swimming pool up against the hive. That's common sense."

Bardin noted that the federal Nuclear Regulatory Commission (NRC) has a policy of limiting the human population near nuclear facilities, but said that while the policy has been in effect since the earliest days of the power program it has yet to be implemented effectively after a nuclear power plant has been licensed by the federal agency.

(more)

"We can and should minimize the human exposure to the risk of harm from a major nuclear accident," Bardin said, "by enforcing population density goals and other measures." He called for compatible land use near present and future nuclear plants. Certain industrial operations, for instance, might prove compatible, Bardin noted.

The Department of Environmental Protection is conducting an in-depth study of potential hazards and land use requirements, Bardin said.

The study headed by Dr. Glenn Paulson, assistant commissioner for science, will enlist the help of municipal and county officials and lay the groundwork for a full review later by all interested parties, Bardin added.

Pending completion of this report, said Bardin, DEP will not approve the construction of residential development or other heavily occupied facilities in the proximity of either the Oyster Creek nuclear plant in Ocean County or those at Artificial Island in Salem County. The action is taken under the state's Coastal Area Facilities Review Act (CAFRA).

Bardin said DEP also is studying the NRC's proposal on the clustering of nuclear plants as a means of reducing land use impacts. But he said DEP has advised the NRC that Ocean County seems a poor place for such clusters. He said DEP would determine at a future date whether the concept has a place elsewhere in New Jersey.

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(Text of testimony attached.)

Testimony of
David J. Bardin, Commissioner,
New Jersey Department of Environmental Protection
before the
Committee on Energy and Environment,
New Jersey Senate
Toms River, New Jersey

March 10, 1976

Senator Russo, members of the Committee:

Nuclear power is no stranger to New Jersey. The Oyster Creek station, located about 10 miles south of Toms River, has produced electricity since December of 1969, with a potential maximum output of 620 megawatts. The first of two units of the Salem station, located on Artificial Island in Salem County, will be started up later this year. This unit can produce 1100 megawatts of electricity. Its identical twin, scheduled to come on line later this decade, will have the same potential output. Other nuclear plants, both in New Jersey and in adjacent states, are at various stages in local, state and federal regulatory proceedings. The New Jersey Department of Environmental Protection (DEP) follows the licensing and regulatory processes for facilities in nearby states in addition to exercising our statutory responsibilities over nuclear power plants within the borders of the State of New Jersey.

The operation of both existing and future nuclear power plants will reduce New Jersey's reliance on fossil-fueled power plants and thereby help minimize the air pollution in the state from sulfur-containing oil and coal. This tradeoff is accompanied by other new risks related to the radiation associated with nuclear power. In contrast to many other recent technological developments ranging from aerosol cans and pesticides to the supersonic transport, nuclear power was subjected to extensive safety and environmental evaluation before its commercial introduction, and is probably the most carefully scrutinized regulated technology in the United States today.

The beehive offers a rough analogy: It's useful, and pretty safe. But bees can sting. A stinging swarm can kill. So the beekeeper takes precautions, and the rest of us keep our distances. We don't put the kids' sandbox or the swimming pool up against the hive. That's common sense. That's been our national objective regarding land uses near nuclear power plants, but this country has not fully implemented this common sense goal.

The State government must make the realistic and responsible protection of the public health and safety a keystone in all of its activities regarding nuclear power.

Review of Available Literature

There are many sources for scientific, medical and engineering information on nuclear power generally as well as on specific facilities. Some significant ones are:

- The Reactor Safety Study (WASH-1400), published by the Nuclear Regulatory Commission in October, 1975

- The Report of the Study Group on Light-Water Reactor Safety, published by the American Physical Society in August, 1975

- An Assessment of Emergency Core Cooling System Effectiveness, prepared by the Environmental Quality Laboratory of the California Institute of Technology in 1975

- The Environmental and Safety Analysis Reports prepared by the Atomic Energy Commission (AEC) and its successor agency, the Nuclear Regulatory Commission (NRC), on each proposed facility

This is only a partial list of key documents on nuclear reactors. Other materials from many governmental agencies, electric utilities and their consultants, and other organizations further expand the extensive base of information available on nuclear reactors.

Normal, Day-to-Day Releases of Radioactive Materials

Every commercial nuclear power plant now running will, under normal operating conditions, release small amounts of radioactivity into the air and water. While some have alleged that these planned releases pose serious risks to the health of human beings and other living things, they have provided no definitive proof. The NRC requires that the overall releases from any facility meet the requirement of being "as low as practicable". In addition, the NRC analysis of the normal rates of radiation released from the nuclear power plants in New Jersey into the air and water are well below current allowable releases (10 CFR Part 20, Appendix B and 10 CFR Part 50, Appendix I). The resulting exposure levels are well below current exposure standards for members of the general public.

For example, the NRC has calculated the maximum individual radiation dose at the plant boundary due to liquid and gaseous radiation releases from one proposed facility to be less than 5 millirem per year (mrem/yr). This is for a hypothetical person who lives at the fence around the facility 24 hours a day, 365 days per year. That increment is about 4 percent of the natural background dose of 130 mrem/yr caused by cosmic radiation and radioactive elements in the ground. DEP has calculated that persons living beyond the plant fence but within a fifty-mile radius would receive, on the average, an exposure of about 0.0008 (eight ten-thousandths) of a mrem/yr from plant operation.

These and similar analyses have led us to conclude that the normal, day-to-day releases from a nuclear power plant do not pose a significant incremental risk to the health of people living in the vicinity of such a facility. DEP continues, however, to monitor the radiation levels at existing nuclear plants independently of any other agency.

Transportation of New and Used Radioactive Fuel

DEP has also analyzed the slight additional radiation exposure that people would sustain due to the transportation of new and used fuel to and from a nuclear power plant. Typically this happens about once a year. For the pro-

posed facility, the additional average exposure to the entire population within 50 miles is 0.003 (three one-thousandths) mrem/yr; this, too, should be compared to the 130 mrem/yr from background radiation. Virtually all of this comes from the used ("irradiated") fuel rods which are sent out of New Jersey for storage and eventual reclaiming of additional energy resources, isotopes for use in medical treatment and research, and for other purposes.

Ultimate Waste Disposal

There are no facilities now proposed for construction in New Jersey to either process used fuel rods or to store or dispose of the radioactive wastes after such processing. The closest potential sites known at this time are in upstate New York and in Barnwell, South Carolina. We agree with the Congressional Office of Technology Assessment, which stated last October, "Satisfactory handling of nuclear fission wastes appears to be technologically feasible, though it has yet to be demonstrated". This area of work has not been funded adequately at the federal level. We support additional funds for research and especially demonstration projects in this area.

DEP does not believe that New Jersey residents will receive any additional exposure from this part of the fuel cycle, except for the transportation of the used fuel rods already discussed.

Major or Catastrophic Accidents

During the operation of any nuclear power plant, new radioactive material is produced by fission and by neutron activation reactions from the metals and other elements in the reactor system. There is no question that the inventory of radioactive fission products which would accumulate in the reactor's nuclear core during operation would, if released in significant quantity due to some catastrophic accident or deliberate act, pose grave risks to life and health for several miles downwind of the accident. For this reason, NRC scrutiny of any proposed reactor design places heavy emphasis on a review of safety-related features of the reactor core and associated hardware (including emergency cooling systems), the structural integrity of the building in which the reactor is to be housed, and factors external to the facility itself related to its proposed location (e.g., population density).

There are a series of types of accidents that can be envisioned, ranging widely in degree of severity. The standard AEC/NRC ranking ranges over Classes 1-9. A Class 8 accident, the design basis for reactor safety analysis, is termed by the NRC the "maximum credible accident"; NRC apparently considers Class 9 accidents incredible. A Class 8 accident is one that does not involve a major catastrophic failure of both the reactor core and the building around it. Rather it involves a series of failures that increase the normal, day-to-day releases substantially, but not necessarily to rapidly lethal levels.

DEP's continuous radiation surveillance at nuclear plant sites provides an extra level of safety assurance to the public for Class 8 and less serious accidents. This off-site surveillance system, while not able to detect in a timely manner the onset of an extremely unlikely but rapidly occurring catastrophic Class 9 accident, nonetheless has the capability to detect the development of malfunctions within the facility which, over the course of days or weeks, might result in significant releases of radiation into the local environment. Such releases can be detected even at levels well below those

currently allowed by NRC standards as well as levels approaching these standards. This State monitoring and reporting network is totally independent of any ongoing monitoring and reporting activity engaged in by any utility or by any federal agency.

A Class 9 accident means a rupture both of the reactor core and the building housing it. This could result in a rapid release of large amounts of radioactivity that would drift downwind. Current public controversy focuses on this risk. This controversy has been fueled, at least in large part, by the failure of the NRC's predecessor, the Atomic Energy Commission, to carry out in a timely manner long-planned safety studies; as the final report of the Energy Policy Project of the Ford Foundation (1974) noted, "an adequate experimental basis for nuclear safety systems is lacking". We condemn the federal budgetary and policy decisions that delayed this important safety research and testing work when the AEC was in charge of it. We look to the Energy Research and Development Administration to eliminate these areas of uncertainty.

Probability of a Major Accident

The NRC Reactor Safety Study estimates (in the absence of experimental data) the probability or frequency of lethal accidents for nuclear reactors as well as for other technological failures and for natural disasters. Final professional reviews of these estimates and the uncertainties in them are not completed, and may lead to significant changes in the NRC Study.

By way of perspective, the Study has a comparison between the frequency of an airplane crash killing up to several hundred people on the ground (for example, a large airplane crashing into a residential area such as near Newark airport) and the probability of a nuclear reactor accident killing the same number of people. In general, the serious reactor accident, according to the NRC Study, is about one thousand times less likely than the airplane accident, which is in turn projected to occur at the rate of about one every few centuries. However, some professionals have challenged that this underestimates the uncertainties that may lead to a major reactor accident.

The sciences of statistical risk analysis and radiation dose estimation have evolved gradually and relatively recently. In their present formative stages, they do not deal adequately with the probability of a series of simultaneous failures or with the potential long-lasting effects of resulting exposures. Apparently the series of events that led up to the fire at TVA's Browns Ferry reactor, for example, was not included as a possible set of events in the NRC Study.

To try to reduce some of these uncertainties, New Jersey petitioned the NRC in 1974 to conduct analyses of the likelihood and consequences of a Class 9 accident before licensing any new reactor configurations (NRC Docket PRM-50-10). Regrettably the NRC has yet to act on our petition.

Consequences of a Major Accident

If there is disagreement as to the likelihood of such events, there is no disagreement that the consequences of a major release of radiation would be potentially catastrophic. The NRC Study as well as previous studies show that a major radiation release passing over any human population could kill people either from radiation sickness or cancer and could contaminate land and water to such a degree as to render them unsafe for a substantial period of time.

This led the AEC to long ago define the concepts of "exclusion zone", "low population zone", and "population center distance" (10 CFR Part 100 embodies these criteria and definitions). The "low population zone" is an area "which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident". The NRC's catalog of "protective measures" includes evacuation and fallout shelters. The NRC has refused to actually define this zone either in terms of population density or total population, contending that "the situation may vary from case to case". In the case of Oyster Creek, the "exclusion zone" is land around the station owned by the utility; the "low population zone" in the State's view is a 3-mile radius around the plant; and the "population center distance" is a radius 4 miles from the plant.

Emergency Plans

These same concerns regarding the potential consequences of a major reactor accident have led to the development of emergency plan requirements and criteria by the NRC and other federal agencies. These plans involve preparation of detailed response procedures by the utility itself and by state and local officials. These officials include, at the state level, the Governor's office, DEP, the State Police, the civil defense agency, the Health Department and indeed almost all other cabinet agencies. At the county and local level, local officials as well as similar agencies (police and fire departments, civil defense agencies, etc.) are also involved.

These plans, developed in view of the specific characteristics of each nuclear power plant, include:

1. methods to assess the severity of an accident

2. potential corrective actions (fire-fighting, damage control)

3. methods of sheltering people or evacuating them from the path of the radioactive cloud

4. techniques to keep people from entering the affected area during periods of risk

5. methods to prevent exposure (protective clothing or breathing equipment)

6. establishment of chains of communication and responsibility among the various public and private agencies involved.

7. procedures by which accurate information is made available to the public during such an emergency

8. criteria for when to allow re-entry into affected areas.

To be effective, these plans require close cooperation between the many other agencies of state, county and local government and DEP's Bureau of Radiation Protection. The basic components of these plans are publicly available. The only confidential sections are those related to details of communication links, state and local police mobilization procedures, and the like.

The plan is flexible, in that it allows the nature of the overall response to be tailored to the extent of the emergency and conditions at the time (e.g., extreme weather and its effect on available transportation). A Class 8 reactor accident requires a far more substantial response than a Class 4 accident. The plan is periodically reviewed by the relevant agencies, and is tested at least in part once per year. These tests to date have stopped short of the actual evacuation of people, but have included the mobilization and deployment of agency personnel at all levels.

The maintenance of this emergency response capability is important. I have directed DEP's Division of Environmental Quality to convene annually a meeting of all appropriate agencies in the vicinity of each New Jersey nuclear power plant. The purpose will be to review all specific plans and procedures, update them where necessary, to make certain each agency understands its responsibilities, and to ensure that means of implementation are in good order.

Location of Nuclear Power Plants: State and Local Regulation of Surrounding Land Uses

The potential major-accident threat to the public health and safety prompted the NRC objectives of limiting the population density near nuclear facilities and controlling land uses so that "there is a reasonable probability that protective measures could be taken...in the event of a serious accident" (10 CFR Part 100). The NRC has not implemented its objectives, apparently hoping to pass this responsibility on to state and local governments.

Local decisions could play a key land-use role. Moreover, both of the New Jersey nuclear power plant sites are in the coastal area as defined in the Coastal Area Facility Review Act (CAFRA), although the Oyster Creek plant is only 1 1/4 miles from the Garden State Parkway, the present western boundary of the CAFRA coastal area in this region. A particularly demanding responsibility rests on DEP to evaluate land uses within the coastal area near nuclear power plants, including approval or disapproval of all residential developments of 25 units or more.

The time has come for state and local government to initiate practical land-use controls around nuclear power plants. Existing state law provides some of the regulatory tools for realistic and responsible implementation of a major national policy: the AEC/NRC policy of limiting the human population within reasonable distances of nuclear facilities. This policy has been in effect since the earliest days of the national civilian nuclear power program, but it has yet to be implemented effectively after a nuclear power plant has been licensed by the AEC or NRC. We can and should minimize the human exposure to the risk of harm from a major nuclear accident by actually enforcing low population density goals and other measures.

The appropriate land use controls need not necessarily foreclose development. For example, we recognize the distinct possibility that certain industrial activities could be most compatible with nuclear power plants. Non-labor intensive industry is one possibility. Industries using by-product heat are another.

As a first step to this end, I have directed the Assistant Commissioner for Science, Dr. Glenn Paulson, to analyze the following issues as they relate to the existing Oyster Creek plant and the nearly-finished Salem plant:

- population density
- meteorological characteristics
- engineering characteristics of the facilities
- the probability of and the potential radiation releases from various types of accidents
- the potential human consequences of various types of accidents under characteristic weather conditions
- the status of the emergency plans for these facilities
- applicable federal, state and local laws and regulations
- compatible land uses

Dr. Paulson will be aided by other appropriate DEP agencies, including the Division of Marine Services, directed by Donald T. Graham, the Bureau of Radiation Protection, headed by John J. Russo, and the Office of Coastal Zone Management, headed by David N. Kinsey. They will seek the help of municipal and county officials. Dr. Paulson's report will provide the basis for a full review by all interested parties of these issues and proposed land-use requirements.

Pending the submission of Dr. Paulson's report and appropriate followup procedures, DEP's Division of Marine Services, directed by Donald T. Graham, will not approve the construction of residential developments or other heavily occupied facilities in the State coastal area in the proximity of either the Oyster Creek nuclear facility or those at Artificial Island.

Nuclear Energy Centers and Mini-clusters

The clustering of nuclear power plants has been proposed, among other reasons, as a way to reduce the overall land use impact of nuclear power plants. For example, clustering would reduce the square miles of low population zones as compared with dispersed single nuclear power plants. Small clusters, perhaps half a dozen or so plants, appear more practical to consider than large aggregates.

In any event, Ocean County seems a poor place for either a mini-cluster or a large aggregate, and DEP so advised the NRC. The Pine Barrens wilderness, the highly developed barrier beach, the intermediate back bays and the adjacent wetlands do not provide the proper setting for a nuclear cluster. We are evaluating the NRC study of the clustering concept, which was required by the Energy Reorganization Act of 1974, to determine whether the concept has a place elsewhere in New Jersey.

Preliminary Draft

AN INVESTIGATION OF THE PROBABILITY OF HYPOTHETICAL CATASTROPHIC ACCIDENTS IN THE OYSTER CREEK NUCLEAR POWER PLANT

Prepared for the

New Jersey Department of Environmental Protection

By

P. R. Davis

October, 1976

Preliminary Draft

TABLE OF CONTENTS

Preliminary Draft

LIST OF TABLES

Preliminary Draft

LIST OF FIGURES

Preliminary Draft

An Investigation of the Probability of Hypothetical Catastrophic Accidents in the Oyster Creek Nuclear Power Plant

By

P. R. Davis

September, 1976

I. INTRODUCTION

This report describes the results of an effort to determine the probability of an hypothetical catastrophic accident at the Oyster Creek nuclear power plant[1]. This study was undertaken for, and under the sponsorship of, the New Jersey Department of Environmental Protection and was directed by Dr. Glenn Paulson, Assistant Commissioner for Science. This effort is part of a larger effort to determine the necessity for, and extent of, appropriate land use requirements in the proximity of nuclear power plants located in the State of New Jersey.

The approach taken in the effort was as follows. First, a determination was made of the type of accidents which could occur in the Oyster Creek facility that had the potential for catastrophic consequences. ("Catastrophic" in this context means those accidents which might be expected to produce human fatalities from radiation damage beyond the boundary of the Oyster Creek plant within 30-60 days from such an accident.) Second, a determination was made of the specific applicability of the Reactor Safety Study (hereinafter referred to as WASH-1400) recently completed by the U.S. Nuclear Regulatory Commission (NRC)[2]; WASH-1400 assessed the overall probability of catastrophic accidents for the two general types of nuclear power reactors currently operating in the U.S. Third, using appropriate adjustments to WASH-1400, a numerical probability, with the range of uncertainties defined as well as possible, was established for the catastrophic accident potential in the Oyster Creek reactor.

II. NUCLEAR POWER REACTORS AND SAFETY

For background and perspective, this section represents a very brief description of nuclear power reactors and their safety features. A nuclear power plant consists of a nuclear reactor and those associated components (such as generators, turbines, and electrical generators) necessary for the production of electricity. A nuclear power plant is similar to a conventional power plant in most respects except for the means employed to produce heat. In a conventional power plant, a fossil fuel (such as oil or coal) is burned to produce heat for steam production; this steam is then used to rotate turbinegenerator units which produce electricity. In a nuclear power plant, a nuclear reactor is used for the heat generation; the rest of the plant is, in essence, identical to a fossil fueled plant.

For fuel, the reactor uses a compound of uranium which contains a small percentage of a special uranium atom (or isotope) known as uranium-235 or U-235. The U-235 atoms are "fissionable"; that is, they can be split apart when struck by a small atomic particle, the neutron. When a U-235 atom splits apart, or fissions, after being hit by a neutron, energy, in the form of heat, is released. The process of fissioning also produces "fission products", which are other radioactive elements and additional neutrons. The neutrons produced by the fissioning of one U-235 atom can be made to cause further fissioning of additional U-235 atoms and, if the U-235 atoms are properly spaced in a matrix of appropriate materials, a controlled, continuous "chain reaction" occurs which is accompanied by the release of large amounts of heat.

The part of the reactor which contains the uranium fuel is called the reactor core. The core is located inside a large vessel (the reactor vessel) which is connected by large pipes to either a steam generator or a turbine. Under normal conditions, pumps continuously circulate water through the reactor vessel when a chain reaction is occuring; this water removes the heat produced in the core by the fission process.

There are two types of power reactors in general use in the United States today. The most common is the pressurized water reactor (PWR). In this type, the water in the reactor vessel is kept under high pressure and is circulated to a steam generator. The steam generator is a large vessel designed to both remove heat from the water which circulates through the core and to transfer this heat to a lower pressure water reservoir which then boils to produce steam. The steam leaves the steam generator via pipes and is directed into a turbine-electrical generator unit which turns to produce electricity.

The second type of power reactor is the boiling water reactor (BWR). In this type, water circulating through the core is held at a lower pressure than in a PWR (though still higher than atmospheric) and allowed to boil. The steam produced from the boiling is piped directly to the turbine-generator units, causing them to rotate and produce electricity. The Oyster Creek reactor, the subject of this study, is of the BWR type.

The fission products produced in the fissioning process are highly radioactive atoms of various chemical elements. Their intrinsic radioactivity causes additional heat to be generated in the core above and beyond the heat associated with the actual fission of the uranium atoms; this additional heat is called "decay heat". Thus, even after the fissioning of uranium atoms in the core has stopped (e.g., by causing the neutrons to become unavailable for further fissioning), a substantial amount of heat continues to be generated from the decay heat process. There is no way to stop this decay heat from being generated. Even though it represents only a fraction of the full-power heat (perhaps 6-10% of the heat generated by the fissioning or chain reaction process when the core is producing maximum power), continuous cooling must be supplied to the core after the fissioning process is stopped to prevent overheating which could lead to a catastrophic accident. This is because, if the decay heat is not removed from the core, overheating from the decay heat can lead to fuel melting. If the fuel melts, the radioactive fission products are likely to be released from the core and could

find their way into the outside world. Because they are highly radioactive, the fission products represent a serious hazard to human health if the level of radioactivity is high enough. Needless to say, these materials can threaten other living things as well as man.

In order to minimize the chances that radioactivity could be released from a melted core, several systems are designed for and built into reactors to maintain core cooling even in the event of unlikely accidents which could interrupt normal cooling water flows; these systems are generally termed "emergency core cooling systems". Also, a large, airtight containment building is always erected around a reactor core so that even in the unlikely even of a radioactive release from the core and primary reactor system, the material might be confined inside the containment building rather than escaping to the outside world.

There are two general types of accidents which have the remote potential for causing a release of radioactivity from the core of a reactor. These two types are: (a) an accident initiated by a loss of (normal) coolant, and (b) accidents initiated as a result of brief (or "transient") conditions in the reactor.

In a loss of coolant accident (LOCA), a cooling water pipe that connects with the reactor vessel is postulated to rupture. Because the water contained within the pipe is under pressure (even in a BWR), it is expelled rapidly from the pipe, and eventually, if nothing else happened, the reactor vessel would become empty; this would stop the chain reaction. However, if emergency core cooling systems were <u>not</u> provided to refill the reactor vessel with water, the core would eventually melt just from the decay heat, releasing radioactivity through the pipe rupture into the containment. If the containment's integrity (leak tightness) were, for some reason, <u>also</u> lost during the accident, the radioactivity could be released to the atmosphere. Depending on weather conditions existing at the time of the accident, the location of people near the site, and the ability of them to either leave the area or take shelter, the radioactivity could be carried by the wind to persons living the site in quantities sufficient to cause their death. Under very unlikely conditions, many persons could be killed from such an accident (WASH-1400, Main Report, ref. 2).

The second type of accident is initiated by a so-called "anticipated transient", an event which briefly but drastically disrupts the orderly transfer of heat energy from the reactor core. This can occur, for example, due to an abrupt <u>increase</u> in the amount of heat produced in the core, or by an abrupt <u>decrease</u> in the ability of the circulating water to remove the heat produced in the core. If the normal heat transfer balance is not restored quickly by the various systems designed to do so, or if the reactor is not quickly shut down, the core could overheat and possibly melt. The overheating process could cause certain pressure relief valves to be opened ("scrammed"), and, if a melting of the core had occurred, radioactivity could exit through these valves from the core into the containment building. Under some conditions, the containment building could fail to retain the radioactivity, and a release to the outside atmosphere would occur. This type of accident has been found to be more likley for a BWR than is a LOCA and is discussed in more detail below.

It should be emphasized that, to date, neither of the above accident sequences has yet occurred in a nuclear power reactor in the United States. (At present, some 58 power reactors are operating in the U.S.; the period during which they have been in service totals some 250 reactor-years of operation). In order for a radioactive release to the atmosphere to occur, several events must occur in sequence. These include (1) an initiating event, (2) the failure of more than one of the redundant systems designed to cope with the accident-initiating event, and (3) a failure of the containment building. If any one of the three events does not occur, there will be no significant radioactivity release. Calculations in WASH-1400[2] indicate that, for both general reactor types, the probability of a significant radioactive release is exceedingly remote; while useful, this conclusion does not provide guidance on specific facilities. The probability of such a release from the Oyster Creek reactor[1] specifically is the subject of this report.

III. RESULTS

Based on the preliminary scoping effort described in this report, the probability of a core melt accompanied by containment failure in Oyster Creek is assessed to be about 3.5×10^{-4} (or about 3 1/2 chances out of 10,000) per year. This would mean that, over the remaining 30-year designed lifetime of the Oyster Creek reactor, the chance of a core melt with containment failure is about 1×10^{-2} (1 chance in 100), although relatively minor modifications to the plant appear available which could significantly reduce this value. This result is based on serveral simplifying assumptions which are stated below; further, the relatively limited resources available for this effort did not allow an analysis fully comparable to that in WASH-1400 (which cost about $5 million). The result should thus be considered no more than a preliminary estimate, subject to revision on the basis of comments and review.

Also, the result is strongly dependent on the probability assumed for failure of "scram" to occur in a transient-induced accident. There does not appear to be a sound basis readily available at this time for selecting a "scram" failure probability. The value used in this study is based on a simplified analysis which relies on assumptions from other sources which review may show are not appropriate to use. The failure probability value thus calculated is at the high end of a rather wide spectrum of values derived by other investigators. Substantial additional effort, probably more properly a federal rather than a state responsibility, appears needed in this area in order to establish in the future a value which could be used with greater confidence. But, given the basic goal of the larger state effort to protect human life in the event of an acccident, this value provides useful guidance.

IV. ANALYSIS

This section describes the analytical evaluations which were undertaken in pursuit of the overall effort. These evaluations proceeded in four discrete parts, as follows:

A. Determination of the type of accidents with the potential for catastrophic off-site consequences (one or more short-term radiation deaths to the human population around the Oyster Creek site).

In order for radiation death to occur to any individual outside the reactor site boundary ("exclusion area") of a reactor site, radioactive material from

the reactor must be carried to the vicinity of the individual in sufficient quantities such that the resulting radiation-induced cell damage is extensive enough to cause mortality. The only source of radioactivity within the plant that has any substantial potential for releasing such quantities of radioactive material is the reactor core. The spent fuel storage pool could conceivably contain, within the stored fuel, a quantity of radioactive material approaching that contained in the core. However, as assessed in WASH-1400, the probability of a significant release from the spent fuel stored at the plant is negligible, on the order of one in one million or less per reactor per year; I concur with this assessment. It is also generally accepted that the only conceivable way for the radioactive material in the core to be transported off-site would be for a meltdown of the core to occur in conjunction with an above-ground failure of the integrity of the containment structure which houses the reactor system. The rationale for this conclusion is given in WASH-1400. It was thus assumed, for the purposes of this effort, that only accidents in Oyster Creek which are postulated to result in loss of above-ground containment integrity along with core meltdown will have the potential for causing immediate human fatalities outside the site exclusion area.

B. Determination of the significant accident sequences in WASH-1400 leading to core melt and loss of above-ground containment integrity.

After extensive investigation over a period of four years, the authors of WASH-1400 concluded that only about 60 accident sequences were of any substantial significance in assessing the general probability of a core meltdown in a BWR such as Oyster Creek. These sequences are shown in Appendix V, pg. V-27 of WASH-1400. As part of this effort, a further analysis was made of the relative significance of these 60 accident sequences. The result of this analysis is that two accident sequences dominate the probability of BWR core meltdown to the extent that all the others (some 58) would increase the core melt probability by only 4% over that contributed by the two dominant sequences.

For this reason, the analysis below focuses only on the two dominant sequences. These two sequences consit of accidents initiated by a severe anticipated transient (mentioned above in Sec. III), which would momentarily but drastically disturb the heat production and transfer process in the reactor. Following the occurrence of certain kinds of anticipated transients, certain safety systems (including ones that would totally shut down the reactor) must successfully operate in order to prevent a core meltdown, a release of radioactivity into the containment building and an eventual loss of containment integrity. The failure of the reactor shutdown systems following an anticipated transient constitutes one of the two major accident sequences leading to core melt. The second sequence involves the failure of the decay heat removal systems to prevent core melt after the transient occurs. Appendix A to this report contains a detailed numerical assessment of all of the dominant accident sequences considered in WASH-1400 and illustrates the dominant role of the two sequences discussed above. Thus, if there are no factors specific for the Oyster Creek plant which would drastically increase the significance of the other 58 accident sequences or decrease the significance of the two dominant sequences, then the probabilty of a core melt in Oyster Creek will also be dominated by the probability of these two accident sequences. The next subsection considers this area in more detail.

C. Determination of site and plant factors specific for Oyster Creek with the potential for changing the BWR core melt accident probability as assessed in WASH-1400.

The Oyster Creek plant's design and characteristics were investigated to determine: (a) if the 58 insignificant BWR core melt accident sequences in WASH-1400 could become significant for Oyster Creek, and (b) whether, and to what extent, the probabilities of individual occurrences in the two dominant core melt sequences in WASH-1400 might need to be altered for specific application to Oyster Creek.

The WASH-1400 assessment of BWR core melt accident probability is based on the design of the Peach Bottom II[3] reactor. Although the General Electric Company designed and supplied both the Peach Bottom II and Oyster Creek reactors, several design and site-related differences were found which have the potential of making the direct application of the WASH-1400 results to Oyster Creek inappropriate. These differences and an assessment of their significance are:

1. Frequency of Anticipated Transients. According to WASH-1400, the average frequency of anticipated transients in a BWR is about 10 per year per reactor. This value was based on reactor operating experience for 1972 (WASH-1400, pg. V-36). Table I shows the actual number of anticipated transients that have occurred at Oyster Creek; it is based on semi-annual operating reports for Oyster Creek (References 4 through 16). The first column gives the time period covered, and the second column gives the number of reactor shutdowns ("scrams") which were precipitated by the transients. Since no scram failures occurred during the period from 1970 through 1975[17], then the number of scrams which were initiated by anticipated transients is also the number of anticipated transients. (All anticipated transients, by definition, result in a reactor condition requiring scram.)

TABLE I

Frequency of Anticipated Transient-Induced

Scrams for the Oyster Creek Power Plant

Time Period	Anticipated Transient-Induced Scrams
7/1/75 to 12/31/75	3
1/1/75 to 6/30/75	1
7/1/74 to 12/31/74	1
1/1/74 to 6/30/74	1
7/1/73 to 12/31/73	1
1/1/73 to 6/30/73	6
7/1/72 to 12/31/72	4
1/1/72 to 6/30/72	4
7/1/72 to 12/31/71	1
1/1/71 to 6/30/71	0
7/1/70 to 12/31/70	6
1/1/70 to 6/30/70	8
Total for the Period 1/1/70 to 12/31/75	36

As can be seen from Table I, 36 scrams occurred as a result of anticipated transients over the 6 year operating history covered in the table. This results in an average of 6 anticipated transients per year, as compared with the 10 assumed in WASH-1400. However, for the first year covered in the table (1970), an inordinately high number of anticipated transients occurred which, according to the utility[17], were the result of problems associated with the fact that the plant was just starting to operate. If 1970 is ignored, the table shows an average of 4.4 anticipated transients per year. Experience since the last half of 1973 indicates that, with the notable exception of the last half of 1975, an even lower rate might be expected in the future. For the purposes of this study, an average frequency of 4 anticipated transients per year was selected as representative of the Oyster Creek plant. (The effect of assuming different anticipated transient frequencies for Oyster Creek, to cover the full range of experience, is given in Appendix B, along with other sensitivity studies. These results are discussed in a later section of this report.)

2. Differences in Emergency Core Cooling Systems ECCS). The Oyster Creek plant does not have a so-called high pressure coolant injection system (HPCIS) as part of its ECCS. Peach Bottom II, the reactor used as the model for BWRs in WASH-1400, has such a system. The system is provided (along with other redundant systems) to protect the reactor core from overheating in the event that a small pipe rupture occurs in the primary cooling system. This difference would indicate that the probability of core melt in Oyster Creek would be greater from small breaks than that assessed for Peach Bottom II in WASH-1400. However, as an upper limit, it can be shown that if the HPCIS is assumed to always fail in Peach Bottom (this is equivalent to not having the system), the overall core melt probability would not significantly change; this can be verified by the information contained in Table A in Appendix A. The HPCIS failure probability is included in the small pipe break accident sequences (details are included in Appendix A). Since the failure probability of the HPCIS was assessed to be about one in ten in WASH-1400 (Appendix II, pg. II-394 et seq.), increasing the total contribution of these specific core melt accident probabilities by a factor of 10 would be equivalent to assuming the HPCIS always failed (or did not exist). As can be seen from Table A-1 in Appendix A, this increase would not result in an accident involving a failure of the HPCIS dominating the accidents initiated by anticipated transients. In addition, the Oyster Creek reactor has a system not included on Peach Bottom II (called the isolation condenser system or ICS) which is designed to provide reliable core protection for small breaks[17]. This system will be described in detail later.

A second difference in emergency core cooling systems between Peach Bottom II and Oyster Creek exists relative to the low pressure coolant injection systems. The Peach Bottom II reactor has systems which will both spray and flood the core, each of which is designed to provide emergency coolant to the reactor core in the event of a loss of coolant accident. The core spray system sprays water down on the core from above, and the core flooding system delivers coolant to the reactor vessel from the bottom to reflood the core. In the Oyster Creek reactor, there is no core flooding system. Instead, two redundant core spray systems are provided. (The reason for these differences is explained in Item 5 following.) As shown in

WASH-1400 (Appendix II), either the core spray system or the core flooding system is assumed to provide adequate core cooling for Peach Bottom II in the event of a loss-of-coolant accident. In addition, depending on the type of accident as well as assumptions made regarding the operation of the safety systems, various combinations of partial operation of the two systems can provide adequate core heat removal. An equivalent analysis for the Oysster Creek plant has apparently not been done either by the utility or by the U.S. Nuclear Regulatory Commission or its predecessor, the U.S. Atomic Energy Commission; it is also beyond the scope of this effort. Such an analysis is required in order to more accurately estimate the differences in core melt probability as a result of the detailed differences in these emergency core cooling systems for Peach Bottom as compared to Oyster Creek. However, in view of the fact that the systems on both reactors appear to be generally similar in terms of components, emergency activating signals, general arrangement, etc., there is no obvious reason to expect that the failure probabilities are substantially different. Also, the contribution to core melt from accidents which these systems are designed to mitigate is very small (as illustrated in Appendix A).

However, it should be noted that the Oyster Creek dual core spray system could be more susceptible to so-called "common mode failures" than the Peach Bottom II systems, since the two Oyster Creek systems are essentially identical. (An example of a common mode failure for the Oyster Creek system would be the plugging of all core spray nozzles in both systems by some mechanism.) The complete assessment of possible common mode contributions is beyond the scope of this effort; however, due to testing and maintenance procedures prescribed for the Oyster Creek plant, significant common mode contributions would not be expected.

A further consideration relative to the differences between Peach Bottom II and Oyster Creek is the effectiveness of sprays as opposed to flooding in cooling a core. It has been recently asserted[21], based on the results of foreign core spray tests, that the effectiveness of core spray cooling may be less than previously assumed. Since Oyster Creek depends exclusively on spray cooling from its two low pressure injection systems in the event of a loss of coolant accident, this could mean that the probability of core melt could be greater if these systems are less effective in fact than they have been assumed to be. However, recent tests by the General Electric Company at their San Jose, California facility have shown that core spray cooling is roughly as effective as has been assumed; additional confirmatory tests, according to GPU[17], are planned by Exxon Nuclear at their Richland, Washington test facility. It was, therefore, assumed that no increase in core melt probability for Oyster Creek would occur due to the possible reduced cooling effectiveness of core sprays. The validity of this assumption should be reviewed when the Exxon Nuclear test results become available.

3. Reactor Protection (Shutdown) System. The Oyster Creek reactor core contains 137 control rods,[1] all of which are designed to be inserted quickly into the core when scram (rapid reactor shutdown) is called for; as noted above, scramming is always required following an anticipated transient. The Peach Bottom II core contains 185 control rods, primarily due to the fact that the Peach Bottom core is larger due to its higher design power output. (See Item 6 below).

The effect of this difference in number of control rods relative to scram failure was not directly evaluated. Instead a preliminary, simplified, independent analysis was performed of scram failure probabilities, since the analysis provided in WASH-1400 has been found to be of questionable validity based on published reviews[18,19,25]. This analysis is presented in Appendix C; the results indicate that the scram failure probability for Oyster Creek is 1.4×10^{-4} (about 14 failures per hundred thousand) per demand, compared to the WASH-1400 assessment of 1.3×10^{-5} (about 13 failures per million) per demand for Peach Bottom. A sensitivity study which relates the effect of scram failure probability to core melt probability is given in Appendix B and is discussed further in a later section.

4. Recirculation Pump Shutdown. In the Peach Bottom II reactor, according to WASH-1400, the reactor recirculation pumps are automatically shut off ("tripped") when scram is called for following an anticipated transient. This pump trip (in conjunction with the manual injection of a liquid reactor poison into the primary reactor coding system) is considered in WASH-1400 to be an effective backup in the event of scram failure (Appendix V, pg. V-4[illegible]). Oyster Creek has no automatic recirculation pump trip[17]; thus, manual activation of this system would have to be relied on in the event of scram failure. Based on information provided in Appendix III of WASH-1400, a probability of 0.5 (one out of two) was assumed for manual recirculation pump trip failure in the event of scram failure. This failure probability value is based on human reliability, and includes the fact that the pumps must be tripped immediately (e.g., within a few seconds for the most serious transient and within a few minuts for less serious ones) after the scram failure in order to be effective. The effect of changes in this failure probability are discussed in a later section; details are provided in Appendix B.

5. Location of Primary System Recirculation Loop Piping. In Peach Bottom II, all recirculation loop pipes are located at an elevation above the core. In Oyster Creek, the pump discharge portion of the recirculation piping is located below the level of the core. This could mean that, if a pipe rupture ocurred in that portion of a recirculation loop (five such loops exist in Oyster Creek) below the core level, it would be difficult, perhaps impossible, to maintain enough water in the reactor vessel to reflood the core. For this reason, Oyster Creek depends on spray cooling (see Item 3 above). This analysis assumes that this difference is not significant overall, since core cooling from reflooding the reactor vessel is not relied upon in Oyster Creek as it is in Peach Bottom II.

6. Plant Size. As noted earlier, at maximum operation the Peach Bottom II reactor produces 1065 megawatts of electrical power, while Oyster Creek produces 620 megawatts. Because of this size difference, there are other plant design differences, including: (a) fewer main steam lines in Oyster Creek (2 vs. 4 for Peach Bottom II), (b) fewer recirculation loops, (c) reduced pumping requirements for emergency cooling, and (d) smaller sizes of the core, reactor vessel, containment vessel, etc, in Oyster Creek. None of these size-related differences significantly affect the accident probabilities.

Table II summarizes all of the differences described above and indicates their relative significance.

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TABLE II

Differences Between Peach Bottom II

and Oyster Creek Nuclear Reactors

Difference	Significance in Terms of Change in Core Melt Probability
1. Frequency of Anticipated Transients	4 per year assumed for Oyster Creek, 10 per year assumed in WASH-1400
2. Emergency Core Cooling Systems	
a. No High Pressure Injection System in Oyster Creek	Not significant
b. Oyster Creek Plant contains an Isolation Condenser System	Significant in providing residual heat removal capability following an anticipated scram (see Section D following)
c. Oyster Creek plant contains two core spray systems rather than one spray and one injection system	Not significant
3. Reactor Protection System	Failure probability assessed to be 1.4×10^{-4} per demand for Oyster Creek vs. 1.3×10^{-5} in WASH-1400
4. Recirculation Pump Trip (RPT) not automatic with scram in Oyster Creek	RPT failure assessed at 0.5 for Oyster Creek; it is considered negligible in WASH-1400
5. Some recirculation loop piping located below core in Oyster Creek	Not significant
6. Oyster Creek smaller than WASH-1400 reactor	Not signficant

D. Calculation of Core Melt Probability for Oyster Creek.

This section presents the method and results of the calculation of core melt probability for the Oyster Creek ractor. Due to the limited scope of this effort, the results need to be extensively qualified; future more detailed analysis might change them. The limitations of the results are described below:

1. Qualifications and Limitations of the Results. A fully comprehensive evaluation of the Oyster Creek core melt probability would entail substantially more effort than described herein. Extensive "event trees" would have to be constructed; as-built drawings of all the reactor sub-systems would be required; all active components (pumps, valves, etc.) would have to be identified and available reliability data for each of these components would need to be gathered and compiled; extensive fault trees for these systems would have to be prepared; and substantial effort would be required in evaluating any common mode failures, design adequacy, human errors, test and maintenance factors, etc, specifically pertinent to Oyster Creek. In short, an effort similar in magnitude to that expended in WASH-1400 to establish the core melt probability for Peach Bottom II would be required if one wished to ensure that the results approached the thoroughness of WASH-1400. The expense for such an effort would probably approach $1 million (WASH-1400 cost $4 million), orders of magnitude above the resources available for the effort described herein. In order to accomplish this analysis within the available time and resources available, it was necessary to make several simplifying assumptions. The validity of these assumptions should be reviewed and evaluated by appropriate individuals and organizations before the results can be generally accepted. The major simplifying assumptions are:

a. Except as noted elsewhere in this report, the failure probabilities for reactor systems involved in the core melt accident sequences are assumed to be identical for Oyster Creek to those determined in WASH-1400. This assumption is based on a necessarily brief review of the full Oyster Creek design and on discussions with personnel from Jersey Central Power and Light[17], the owner of Oyster Creek, as well as with appropriate officials of the State of New Jersey.

b. There are no features or activities related to the Oyster Creek site (such as major nearby airports, LNG facilities, etc.), which would, over the remaining lifetime of the plant, be significant in initiating an off-site accident (e.g., direct plane crash) leading to core melt. This assumption is based on a review of the site characteristic and environs as well as discussions with New Jersey state officials.

2. Description of Dominant Core Melt Accident Sequences. As discussed earlier in Section IIB, and as numerically assessed in Appendix A, which in turn is based on WASH-1400, there are two accident sequences which dominate the core melt probability in a BWR as analyzed in WASH-1400. These two sequences are also assumed likely to dominate the core melt probability for Oyster Creek (Section IIC). The two accident sequences are: (a) the occurrence of an anticipated transient followed by the failure of systems designed to rapidly reduce core power, and (b) the occurrence of an anticipated transient followed by an

eventual failure to remove core decay heat WASH-1400 assumes that both of these accidents are followed by a rupture in the containment structure and thus lead to a release of radioactivity to the atmosphere. In order to understand the assessment of the likelihood of these two accident sequences in the Oyster Creek reactor, it is first necessary to understand the basic features of these hypothetical sequences.

a. Anticipated transient followed by failure to reduce core power. As noted above, an aniticipated transient is an event which leads to an imbalance in the core power production and core heat removal processes so that the core temperature begins to rise. Unless core power production is reduced, such an occurrence could lead to core melt and a catastrophic accident. A discussion of which events can constitute an anticipated transient is discussed in Appendix V (Section 4.3, page V-36) of WASH-1400. As shown in Section III-C.1. of this report, the average frequency of anticipated transients at the Oyster Creek plant since 1970 is about 4 per year.

Following the occurrence of an anticipated transient, the required core power reduction can be accomplished in one of two ways. The first way (and the one which has always successfully occurred in the Oyster Creek plant and all other BWRs) is for the automatic reactor protection system (the "scram" system) to rapidly insert control rods into the core.

If the automatic scram system fails, immediate power reduction can be accomplished by the manual tripping (shutdown) of the reactor recirculation loop pumps. This step must then be followed within about ten minutes by the injection of a neutron-absorbing material (a reactor "poison"); this injection, which can only be accomplished as a result of manual activity by the operator, further reduces the probability of a core melt.

In order to assess the probability of a core melt accident from an anticipated transient without core power shutdown in Oyster Creek, it is necessary to know the probabilities associated with all of the events described in the preceding discussion. Table III describes each event, summarizes the assessed probabilities both for Oyster Creek and the BWR analyzed in WASH-1400, and indicates where the assessed probability is found in this report and in WASH-1400 respectively.

With the information contained in Table II, it is possible to compute a probability for a core melt accident in Oyster Creek caused by an anticipated transient followed by failure to reduce core power. One way to perform such a computation is to construct an "event tree" as was done in WASH-1400. The event tree describes the inter-relationships between the various events in Table III and, if enough probability information is known, leads directly to the computation of the probability of the event of interest (in this case, core melt from an anticipated transient followed by failure to reduce core power). Figure 1 shows such a computation associated with the event tree. As shown in Figure 1, the computed probability of a core melt accident from an anticipated transient event followed by failure to reduce core power based on the assumptions and analyses presented herein is 3.4×10^{-4} per year, about 3.4 times per hundred thousand. Over the remaining 30-year lifetime of the plant, this means that the likelihood of this accident is about one chance in 100.

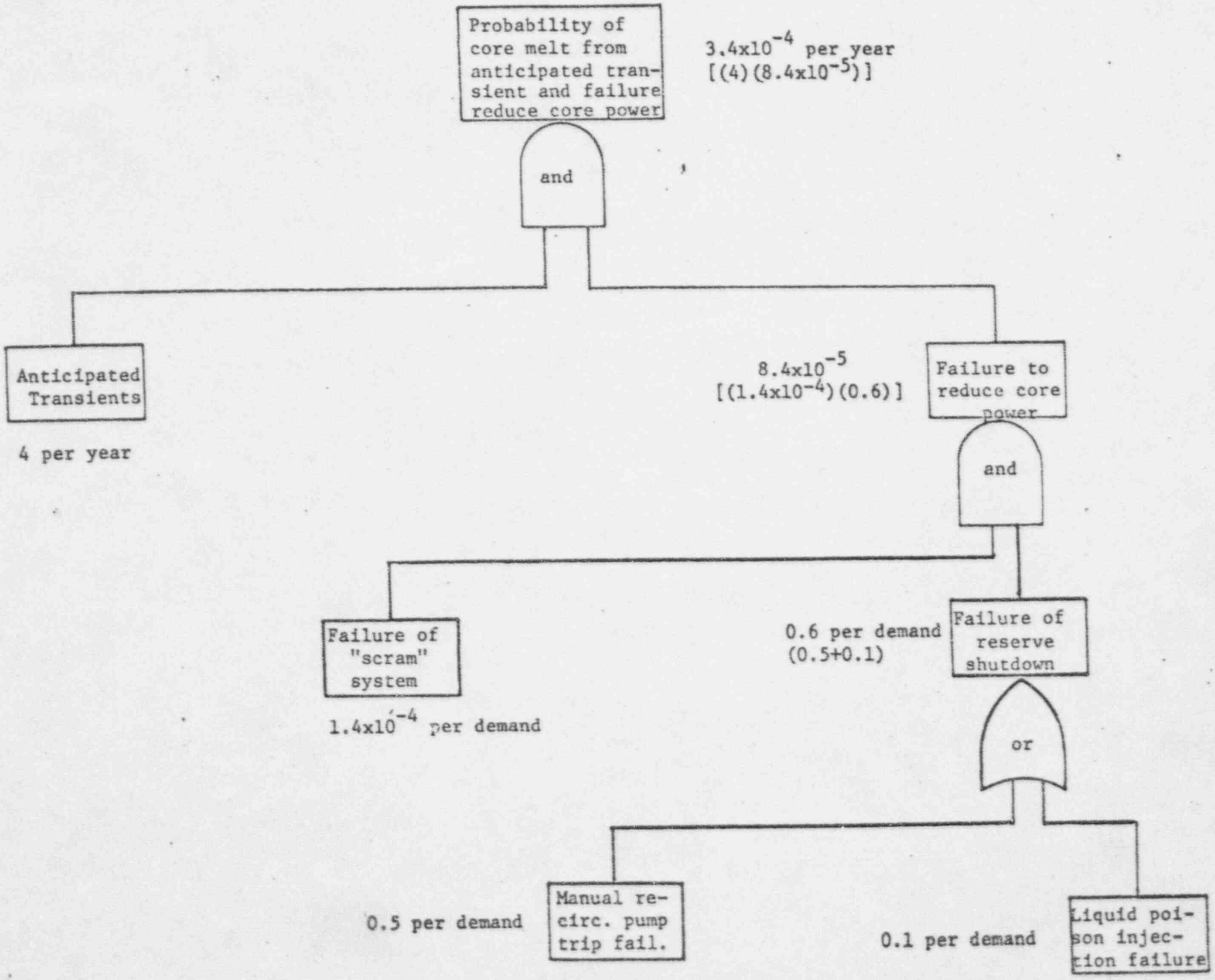

FIG. 1 Computation of Probability of Core Melt in the Oyster

TABLE III

Assessment of Probabilities Associated with
Anticipated Transients without Core Power Shutdown

Event	Probability	
	WASH-1400	Oyster Creek
1. Anticipated Transients: Frequency per year	10 (Appendix V, Sec. 4.3)	4 (Section II-C.1.)
2. "Scram" system failure probability, per demand	1.3×10^{-5} (Appendix II, Sec. 6.2)	1.4×10^{-4} (Appendix A of this report)
3. Recirculating loop pump failure probability, per demand	Negligible	0.5 (Sec. III-C.4.)
4. Failure to manually introduce liquid poison	0.1 (Appendix V, Sec. 4.3)	0.1 (Same as WASH-1400 value)

b. Anticipated transient followed by failure to remove decay heat. The second accident sequence which was determined in WASH-1400 to contribute significantly (see Appendix A) to the BWR core melt accident probability was a transient-initiated accident followed by failure to remove core decay heat (described on page V-52 of WASH-1400). After reviewing the accident sequence described in WASH-1400 for Peach Bottom II and investigating decay heat removal systems in Oyster Creek, it was determined that the WASH-1400 analysis does not apply to Oyster Creek. Several reasons exist for this, but the main one is the existence of the isolation condensor system (ICS) in Oyster Creek. There is no comparable system in Peach Bottom II. In order to determine the core melt probability in Oyster Creek from a transient accident followed by failure to remove decay heat, an independent assessment was performed. This is described next.

Following an anticipated transient with successful core power reduction (scram), the ICS would be used in Oyster Creek to remove decay heat. This system consists of a closed loop heat removal system which is connected to the primary cooling system. The ICS is normally isolated from the primary system by two valves in parallel. When the ICS is needed (such as for removal of decay heat), the valves are opened and the primary system cooling fluid from the core (mostly steam) flows through the ICS to a heat exchanger, is condensed, and the resulting water flows back into the bottom of the reactor vessel via the main recirculation lines. The ICS is arranged so that the cooling cycle occurs by natural convection; special pumping power is not required. In addition, the ICS valves are operated by DC power available from within the plant so that the loss of all AC power (e.g., from off-site) does not disable the system. The valves are automatically activated following an anticipated transient. Heat is removed from the ICS heat exchanger by steam production; the steam is released to the outside atmosphere. The water supply for this heat removal is provided by a supply tank which can supply water for 1 3/4 hours following initial operation of the ICS. If the ICS fails (by failure of the valves to receive an opening signal when required, by failure of the DC power supply, or by mechanical failure of the valves), two other mechanisms are available

to remove heat. One is removal via the power conversion system (PCS); this requires that the main steam isolation valves (MSIV) be manually opened, the turbine bypass valves opened, and the feedwater pumps started. The MSIVs automatically close for most anticipated transients, and the feedwater pumps trip off, that is, close. If both the ICS and the PCS fail, an automatic depressurization system (ADS), designed to automatically activate, should come into play, reducing primary system pressure. When the pressure drops to a pre-set level, the core spray injection systems activate automatically to maintain water in the reactor vessel for core cooling. The steam produced from the core is vented through the ADS relief valves to the water in the "torus", a water-holding chamber beneath the reactor. Heat is removed form this water by the torus heat removal systems (two redundant systems). Heat is transferred by these systems into a heat exchanger fed by water from the emergency service water system.

Since the ICS self-contained water supply lasts only 1 hour and 45 minutes, it is necessary, after this time has elapsed, to add additional water to the ICS. This can be accomplished by either pumping water from a storage tank or from the fire water system. The storage tank capacity is sufficient to supply adequate water to the ICS heat exchanger for at least one day. The fire water supply is essentially unlimited, coming from sources external to the plant.

Given that the ICS self-contained water supply is adequate for 1 3/4 hours, it is convenient to consider the decay heat systems for two time periods during the accident, those decay heat removal systems available at times less than 1 hour and 45 minutes and those systems available after 1 hours and 45 minutes. Fault trees were constructed for both time periods and are shown in Figures 2 and 3; the failure probabilities for each system are also shown. These probabilities were derived by various means as shown in Table IV. It should be recognized that the failure probability of some of the systems, notably the ICS, have not been determined with the full rigor used in WASH-1400 since such quantification is beyond the scope of this effort. Instead, in these cases, failure probabilities were assessed based on values for similar systems in WASH-1400. The explanation for each system failure probability is given in the right hand column of Table IV.

Figure 2 shows the probability of failure to remove core decay heat within 1 3/4 hours following an anticipated transient in the Oyster Creek reactor. As can be seen, this failure probability, 1×10^{-11} (or about one in one hundred million) is negligible. Figure 3 shows the probability of failure to remove decay after 1 3/4 hours (when the self-contained water supply for the ICS is exhausted). The result is 1.4×10^{-6} (about 1.4 times in one million). Obviously, this second time period dominates the decay heat removal failure following anticipated transient.

Assuming, as discussed previously, that the frequency of anticipated transients is 4 per year, the probability of a core melt in the Oyster Creek reactor from an anticipated transient followed by failure to remove decay heat is assessed to be 4 multiplied by (1.4×10^{-6}) or 5.6×10^{-6} per year (about 5.6 chances in a million). Thus, over the remaining 30-year life of the reactor, the likelihood of a core melt by this mchanism is thirty times larger this value per year or, 1.68×10^{-4} (about one chance in 6,000).

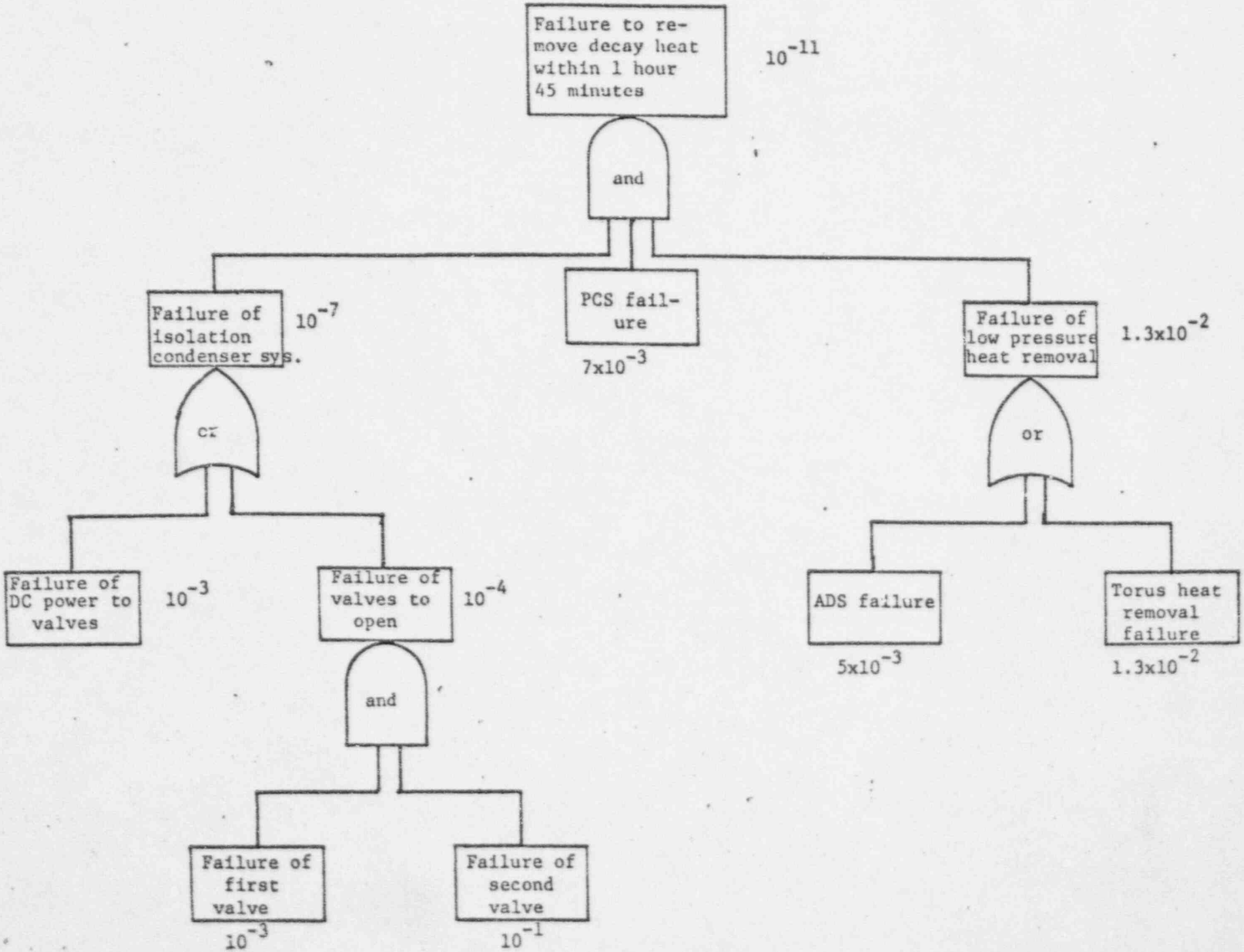

FIG. 2 - Computation of Probability of Failure to Remove Decay Heat

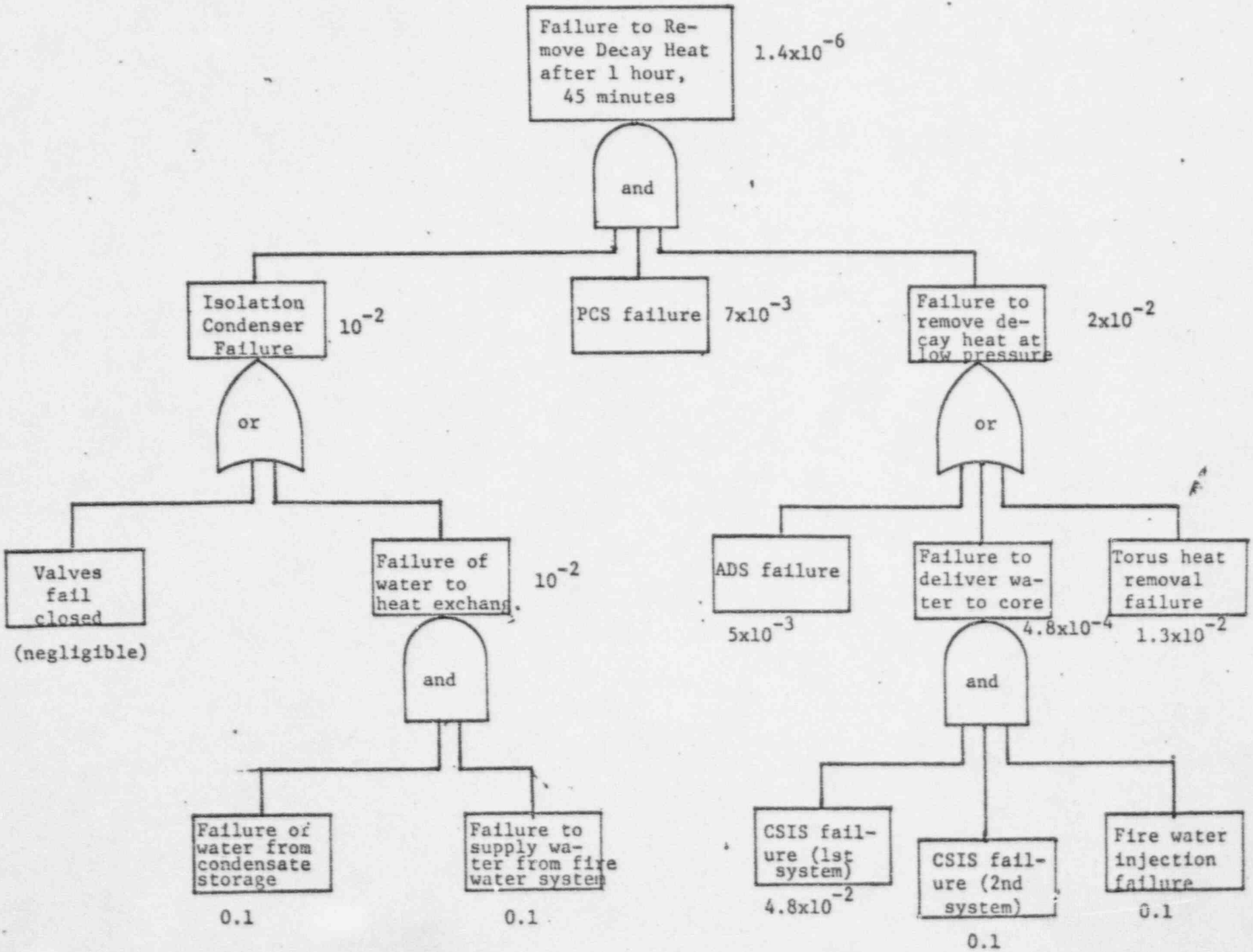

FIG. 3 - Computation of Probability of Failure to Remove Decay Heat after 1 hour 45 minutes

3. Numerical Result of Oyster Creek Core Melt Probability. Based on the analyses presented in previous sections of this report, the Oyster Creek core melt accident probability is dominated by an anticipated transient followed by failure to shut down core power generation. The total core melt probability per year from the two dominant accidents considered, as illustrated in Table V, is 3.46×10^{-4}, about 3 1/2 chances in ten thousand per year.

TABLE IV

Assessed Failure Probability for Oyster Creek Systems

System or Component	Failure Probability	Rationale for Failure Probability
DC power failure	10^{-3}	pg. II-88*
First valve failure	10^{-3}	Appendix III*
Second valve failure	10^{-1}	Appendix III* (common mode considerations)
PCS failure	7×10^{-3}	pg. V-41*
ADS failure	5×10^{-3}	pg. II-404*
Torus heat removal failure	1.3×10^{-2}	pg. II-177 (similar to PWR LPRS)*
Valve fails closed	Negligible	After being opened, valve failing closed is very unlikely, especially compared to failure of water to ICS heat exchanger
CSIS failure (first system)	4.8×10^{-2}	pg. II-387*
CSIS failure (second system)	10^{-1}	Appendix II* (common mode consideration)

* from WASH-1400[2]

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TABLE V

Core Melt Probability for
the Oyster Creek Reactor

Accident	Core Melt Probability (per year)
Anticipated transient without scram	3.4×10^{-4}
Anticipated transient with failure to remove core decay heat after 1 3/4 hours	5.6×10^{-6}
TOTAL	3.4×10^{-4}

Taking into account both of these dominant sequences, this probability means that, over the remaining 30-year design lifetime of the Oyster Creek reactor, the overall probability of a core melt accident is 30 times larger than the annual value, 1×10^{-2}, or about 1 chance in 100.

It would be of considerable value to establish the uncertainty limits for the Oyster Creek core melt probability as assessed here. However, any computation of uncertainty limits would involve substantial further effort requiring detailed knowledge of the failure uncertainty associated with each component of each system, the test and maintenance schedules, possible common mode contributions and their related uncertainties, etc. This information would have to be combined in a complex statistical manner in order to establish uncertainty limits and confidence intervals. This full quantification of these uncertainties is beyond the scope of the present effort. As an alternative, a sensitivity study was undertaken to determine the relative significance of factors involved in the computation of the Oyster Creek core melt probability. The details of this study are presented in Appendix B. Table VI is a composite table showing the results of the study. The first column identifies the contributor used in the core melt probability computation, and the second column gives the decrease factor in the core melt probability for a factor of two decrease in the value used above for the contributor listed in the first column. Only those factors associated with the anticipated transient without scram accident are considered since no other sequences are significant by comparison.

TABLE VI

Core Melt Sensitivity Study

Core Melt Probability Contributor	Factor of Change in Core Melt Probability for Factor of 2 Changes in the Core Melt Probability Contributor
1. Anticipated Transient Frequency	2
2. Scram Failure Probabilities	2
3. Recirculation Pump Trip Failure	1.7
4. Manual Poison Injection Failure Probability	1.1

As shown in the table, the core melt probability is most sensitive to the frequency of anticipated transients and the scram failure probability. In both cases, a reduction of a factor of 2 would result in a similar reduction in core melt probability. The results in Appendix B shows that if the probability of recirculation pump trips were to be made negligible (as was assumed in WASH-1400 for Peach Bottom II), the core melt probability would be reduced to about 2×10^{-3} or 1 chance in 500, rather than 1 chance in 100, over the remaining design operating life of the Oyster Creek reactor.

The largest uncertainty, by far, is in the foregong analysis of scram failure probability. The simplified analysis given in Appendix A for scram failure at Oyster Creek is undoubtedly inadequate in view of the importance of this factor. However, it is not clear that techniques readily exist at this time which would allow the derivation of a value that could be applied with greater confidence. In view of these uncertainties, a survey was undertaken to determine what scram failure probabilities have been derived by various investigators. The results of this survey are summarized in Table VII. The table shows that values ranging from 4×10^{-4} to 1.5×10^{-6} have been quoted for scram failure probability. The value used in this study (1.4×10^{-4}) is at the high end of this range, and therefore tends to produce a high value for the core melt probability. Table VII also includes a core power shutdown failure probability which includes the reliability of the combination of recirculation pump trip and manual poison injection as a backup to scram failure (evaluated only for WASH-1400 and this study).

VII also includes a core power shutdown failure probability which includes the reliability of the combination of recirculation pump trip and manual poison injection as a backup to scram failure (evaluated only for WASH-1400 and this study).

It should be noted that, as a result of the U.S. Atomic Energy Commission's investigation of anticipated transients without scram[22], General Electric Co. has proposed to incorporate automatic recirculation pump trip and automatic poison injection as a backup to scram failure[24]. It is not known what the current NRC requirements are for the incorporation or retrofitting of such a system in BWRs. If such a system were employed on Oyster Creek, and assuming a failure probability of 10^{-2} (which should be easily attainable), the core melt probability, based on the results of this study, could be reduced to one chance in over 3000 for the remaining life of the Oyster Creek reactor.

TABLE VII

Survey of Core Power Shutdown Failure Probability

Source	Reference	Scram Failure Probability	Core Power Shutdown Failure Probability
AEC	22	1×10^{-4} and 4×10^{-4} (1)	--
WASH-1400	2	1.3×10^{-5}	1.3×10^{-6}
Fullwood	23	1.9×10^{-5}, 8.8×10^{-6}, and 1.5×10^{-6} (2)	--
This Study	--	1.4×10^{-4}	8.4×10^{-5}

(1) These values are based on data from a variety of reactors.
(2) Based on different statistical treatments of data from light water reactors.

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    Dr. Peter Preuss, State of Ne Jersey, DEP
    Donald A. Ross, Jersey Central Power and Light
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