NRC FORM 464 Part I (6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

UNITED STATES NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA	RESPONSE NUMBER
2001-0323	1

RESPONSE TYPE ☑ FINAL ☐ PARTIAL

REQUESTER: Gerardo M. Caballero

DATE: AUG 29 2001

PART I. -- INFORMATION RELEASED

☐ No additional agency records subject to the request have been located.

☐ Requested records are available through another public distribution program. See Comments section.

☑ APPENDICES A — Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.

☐ APPENDICES — Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.

☑ Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room. ~~2120 L Street, NW, Washington, DC.~~

☑ APPENDICES A* — Agency records subject to the request are enclosed.

☐ Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.

☐ We are continuing to process your request.

☐ See Comments.

PART I.A -- FEES

AMOUNT * $

* See comments for details

☐ You will be billed by NRC for the amount listed.

☐ You will receive a refund for the amount listed.

☑ None. Minimum fee threshold not met.

☐ Fees waived.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

☐ No agency records subject to the request have been located.

☐ Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.

☐ This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

With regard to category A of your request, many records exist regarding the Three Mile Island accident that are already publicly available. We have enclosed a Fact Sheet entitled, "The Accident at Three Mile Island," which provides a summary of events, information on the health effects and impact of the accident, and provides the current status. The fact sheet also contains a list of pertinent records regarding the accident all of which are publicly available. Records subject to categories B & D of the request are identified on enclosed Appendix A and are already publicly available. Records with an ML Accession No. are publicly available in the NRC's Public Electronic Reading Room at http://www.nrc.gov/NRC/ADAMS/index.html. If you need assistance in obtaining these records or any other publicly available records, please contact the NRC Public Document Room (PDR) at (301) 415-4737, or 1-800-397-4209. A notice is enclosed which provides procedures for obtaining records from the PDR.

*We have enclosed documents A/4, A/6 and A/7 to provide you the 100 pages you are entitled to free of charge. We have also enclosed a Fact Sheet entitled "Next-Generation Reactors" for your information.

No records were located with regard to category C of your request.

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Carol Ann Reed

Mary Jean Pool (for)

FOIA No. 2001-0323

APPENDIX A
RECORDS ALREADY AVAILABLE IN THE PDR

NO.	DATE	ACCESSION NO.	DESCRIPTION/(PAGE COUNT)
1.	12/05/00	ML003776078	Ltr J. Muntz. Exelon Generation Co., to NRC Document Control Desk, Re: PBMR Review Requirements (3 pages)
2.	01/12/01	ML010110527	Ltr W. Travers, NRC, to J. Muntz, Exelon Generation Co., Re: Response to 12/05/00 letter (2 pages)
3.	02/12/01	ML010440070	Memo T. King, RES, to A. Thadani, RES, Re: January 31, 2001, Meeting Summary w/Exelon & Other Interested Stakeholders (48 pages)
4.	04/25/01	ML011010424	Memo W. Travers, EDO, to Commission Re: SECY-01-0070 - Plan for preapplication activities on the PBMR (18 pages)
5.	05/10/01	ML011420393	Ltr J. Muntz, Exelon, to T. King, RES, Re: Regulatory Issues Related to PBMR (44 pages)
6.	05/14/01	ML011340114	Memo T. King, RES, to A. Thadani, RES, Re: April 30, 2001, Meeting Summary w/Exelon, et al. (66 pages)
7.	05/25/01	ML011520314	Ltr J. Muntz, Exelon, to NRC Document Control Desk, Re: PMBR 10 CFR Part 52 Application and Licensing Plans (15 pages)
8.	06/01/01	ML011620360	Ltr K. Borton, Exelon, to NRC Document Control Desk, Re: Upcoming Pre-application Meeting on Proposed Licensing Approach (3 pages)
9.	06/19/01	ML011700453	Staff Requirements Memorandum - SECY-01-0070 re: Plan for PBMR preapplication activities (29 pages)
10.	06/25/01	ML011770055	Ltr T. King, RES, to K. Borton, Exelon, Re: Upcoming July 17-18, 2001 meeting (4 pages)
11.	07/09/01	ML012140094	Memo T. King, RES, to A. Thadani, RES, Re: July 17-18, 2001 Meeting Summary (6 pages)
12.	07/23/01	ML012040355	Memo T. King, RES, to A. Thadani, RES, Re: June 12-13, 2001, Meeting Summary (102 pages)
13.	08/01/01	ML012140094	Memo T. King, RES, to A. Thadani, RES, Re: July 17-18, 2001, Meeting Summary (6 pages)

Fact Sheet

United States Nuclear Regulatory Commission
Office of Public Affairs
Washington, D.C. 20555
Telephone: 301/415-8200 E-mail: opa@nrc.gov

Next-Generation Reactors

Background

The NRC has long sought standardization of nuclear power plant designs, and the enhanced safety and licensing reform which standardization could make possible. The NRC's regulation (Part 52 to Title 10 of the Code of Federal Regulations) provides a predictable licensing process including certification of next-generation reactor designs. The design certification process is the key for early public participation and resolution of safety issues prior to an application to construct a nuclear power plant.

Pre-Application Reviews

The NRC's "Statement of Policy for Regulation of Advanced Nuclear Power Plants," July 8, 1986, encourages early discussions (prior to a license application) between NRC and reactor designers to provide licensing guidance. In June 1988, the NRC issued NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants." This document provides guidance on the implementation of the policy and describes the approach used by NRC in its review of advanced reactor concepts.

The NRC has conducted pre-application reviews of advanced reactor designs to identify (1) major safety issues that could require Commission policy guidance to the staff, (2) major technical issues that the staff could resolve under existing regulations or NRC policy, and (3) the research needed to resolve identified issues.

Final Design Approvals

The review process for next-generation reactor designs involves the certification of standard reactor designs by a rulemaking process (Subpart B of Part 52). The design certification process requires an applicant to provide the technical information necessary to demonstrate compliance with the safety standards set forth in NRC regulations (10 CFR Parts 20, 50, 73, and 100). Applicants for design certification must also provide information related to the resolution of unresolved and generic safety issues, issues that arose after the accident at the Three Mile Island plant, a detailed analysis of the design's vulnerability to certain accidents or events, and inspections, tests, analyses, and acceptance criteria.

The NRC provided final design approvals following rulemaking for three reactor designs that can be referenced in an application for a nuclear power plant application. These are:

1. **Advanced Boiling Water Reactor** design by GE nuclear Energy (August 1997);
2. **System 80+** design by Westinghouse (formerly ABB-Combustion Engineering) (June 1998); and
3. **AP600** design by Westinghouse (March 2000).

Status of Other Design Reviews

Nine other reactor designs have been submitted for NRC review, however only two--the AP1000 and the Pebble Bed Module Reactor-- are being actively pursued at this time. The status of each application is provided below in alphabetical order.

Reactor Design Status

- **AP1000** - Westinghouse requested review of its AP1000 design, by letter dated May 4, 2000, in order to determine the scope of a future design certification review. The NRC expects to complete its pre-application review in 2001.
- **CANDU 3U** - NRC terminated its review at the request of Atomic Energy of Canada, Limited, in March 1995.
- **MHTGR** - NRC discontinued its review in early 1996 at the Department of Energy's request.
- **PBMR** - Exelon Generation Company, by letter dated December 5, 2000, has requested to meet with NRC to discuss issues associated with the potential to license a Pebble Bed Modular Reactor design. An initial public meeting was held January 31, 2001.
- **PIUS** - The NRC documented its pre-application review of ABB-CE's Process Inherent Ultimate Safety design in April 1994 and terminated all other activities until an application for design certification is submitted.
- **PRISM** - The Department of Energy submitted the conceptual design for the Power Reactor Innovative Small Module to NRC for pre-application review in November 1986. DOE amended their design document in 1990 and NRC completed its review in February 1994.
- **RESAR SP/90** - The NRC published its final safety evaluation report (NUREG-1413) for Westinghouse's advanced pressurized water reactor design in April 1991 and issued a preliminary design approval. RESAR SP/90 was the first "evolutionary" light-water-reactor.
- **SAFR** - The NRC's pre-application safety evaluation report (NUREG-1369) for the Sodium Advanced Fast Reactor (SAFR) design, sponsored by DOE, was published in December 1991.
- **SBWRGE** - Nuclear Energy submitted an application for final design approval and design certification in August 1992. The NRC, in May 1993, determined that it was acceptable for review. In response to some NRC concerns, GE sponsored testing which continued into 1996. However, in March 1996, GE announced the cancellation of the design certification application with an intent to shift the focus of its SBWR programs to plants of 1000 MWe or larger. At GE's request, NRC closed out its review activities in early 1997.

Design Descriptions

ABWR: The U.S. Advanced Boiling Water Reactor design uses a single-cycle, forced circulation, boiling water reactor with a rated power of 1300 megawatts electric (MWe). The design incorporates features of the BWR designs in Europe, Japan, and the United States, and uses improved electronics, computer, turbine, and fuel technology. The design is expected to show improvement in plant availability, operating capacity, safety, and reliability. Improvements include the use of internal recirculation pumps, control rod drives that can be controlled by a screw mechanism rather than a step process, microprocessor-based digital control and logic systems, and digital safety systems. The design also includes safety enhancements such as containment over pressure protection, passive core debris flooding capability, an independent water makeup system, three emergency diesels, and a combustion turbine as an alternate power source.

AP600: This is a 600 MWe advanced pressurized water reactor that incorporates passive safety systems and simplified system designs. The passive systems use natural driving forces without active pumps, diesels, and other support systems after actuation. Use of redundant, non-safety-related, active equipment and systems minimizes unnecessary use of safety-related systems.

AP1000: This is a larger version of the previously approved AP600 design. It is a 1000 MWe advanced pressurized water reactor that incorporates passive safety systems and simplified system designs. It is similar to the AP600 design but uses a longer reactor vessel to accommodate longer fuel, larger steam generators, and a larger pressurizer.

CANDU 3U: This is a single-loop, pressurized heavy water reactor rated at 450 MWe with two steam generators and four heat transport pumps. The design utilizes natural uranium fuel, separate heavy water moderator and reactor coolant, computer-controlled operation, and on-line refueling. The reactor has 232 horizontal pressure tubes supported in a tank filled with the heavy water moderator. The tank also supports the reactivity regulating and safety devices which are inserted between and among the pressure tubes. Except for its smaller size and evolutionary design improvements, the CANDU 3U is similar in design to a number of CANDU reactors operating in Canada and other countries.

MHTGR: The Modular High Temperature Gas-Cooled Reactor is a helium-cooled and graphite-moderated thermal power reactor. The fuel is millions of ceramic coated microspheres distributed in cylindrical rods which are inserted in large hexagonal graphite blocks. The blocks are stacked vertically within the reactor vessel through which pressurized helium coolant is circulated. The plant design consists of four identical reactor modules, each with a thermal output of 350 MW, which are coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MWe. The design includes passive reactor shutdown and decay heat removal features to minimize required reactor operator actions.

PIUS: The Process Inherent Ultimate Safe reactor is a 640 MWe advanced pressurized water reactor designed by ABB-Atom of Sweden that utilizes natural physical phenomena to accomplish control and safety functions. The PIUS design consists of a vertical pipe, called a reactor module, which contains the reactor core and is submerged in a large pool of highly borated water. The reactor core is comprised of fuel elements that are similar to current PWR fuel elements. The borated pool water is provided to shut down the reactor and to cool the core by natural circulation. Unlike most reactors, PIUS does not use control rods for controlling the nuclear chain reaction. The reaction is controlled by the boron concentration and temperature of the primary loop reactor water. The steam generating equipment of the PIUS design is similar to that of.a typical pressurized light water reactor plant. One important difference in plant design is the very large, by current standards, prestressed concrete reactor vessel. This vessel holds both the reactor module and the borated pool.

PRISM: The Power Reactor Innovative Small Module design uses a modular, pool-type, liquid-sodium cooled reactor producing 471 MWt. The reactor fuel elements are cylindrical tubes containing pellets of uranium-plutonium-zirconium metal alloy. The reactor size permits use of passive shutdown and decay heat removal features. The standard plant consists of nine reactor modules arranged in power blocks of three reactor modules of 465 MWe. Each module is located in its own below-grade silo and is connected to its own intermediate heat transport system and steam generator system. The steam generator and secondary system hardware are located in a separate building and are connected by a below-grade pipe-way. All the reactors on the site share a common control center, reactor maintenance facility, remote shutdown and radwaste facility, and assembly facility. Each reactor module has its own steam generator which is combined with the two other steam generators in each power block. Total electrical power output would be 1395 MWe.

SBWR: The Simplified Boiling Water Reactor design uses a 600 MWe boiling water reactor with simplified power generation, safety, and heat removal systems to reduce power generation costs, simplify plant safety, and reduce construction times. The SBWR uses natural circulation for coolant flow through the reactor. The emergency core cooling is provided by a gravity-driven core cooling system that reduces piping and eliminates pumps and the need for safety related diesel generators. Because there are no large pipes attached to the vessel near or below the core elevation, the design is intended to ensure full core coverage for all design basis accidents. The isolation condenser is designed as a safety-related system to remove decay heat from the reactor core, by natural circulation and with minimal loss of reactor coolant inventory, following reactor isolation and shutdown.

System 80+: This standard plant design uses a 1300 MWe pressurized water reactor. It is based upon evolutionary improvements to the standard CE System 80 nuclear steam supply system and a balance-of-plant design developed by Duke Power Co. The System 80+ design has safety systems that provide emergency core cooling, feedwater and decay heat removal. The new design also has a safety depressurization system for the reactor, a combustion turbine as an alternate AC power source, and an in-containment refueling water storage tank to enhance the safety and reliability of the reactor system.

February 2001

Fact Sheet
United States Nuclear Regulatory Commission
Office of Public Affairs
Washington, D.C. 20555
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The Accident at Three Mile Island

Additional Information Sources | Glossary | Plant Diagram

The accident at the Three Mile Island Unit 2 (TMI-2) nuclear power plant near Middletown, Pennsylvania, on March 28, 1979, was the most serious in U.S. commercial nuclear power plant operating history[(1)], even though it led to no deaths or injuries to plant workers or members of the nearby community. But it brought about sweeping changes involving emergency response planning, reactor operator training, human factors engineering, radiation protection, and many other areas of nuclear power plant operations. It also caused the U.S. Nuclear Regulatory Commission to tighten and heighten its regulatory oversight. Resultant changes in the nuclear power industry and at the NRC had the effect of enhancing safety.

The sequence of certain events - - equipment malfunctions, design related problems and worker errors - - led to significant damage to the TMI-2 reactor core but only very small off-site releases of radioactivity.

Summary of Events

The accident began about 4:00 a.m. on March 28, 1979, when the plant experienced a failure in the secondary, non-nuclear section of the plant. The main feedwater pumps stopped running, caused by either a mechanical or electrical failure, which prevented the steam generators from removing heat. First the turbine, then the reactor automatically shut down. Immediately, the pressure in the primary system (the nuclear portion of the plant) began to increase. In order to prevent that pressure from becoming excessive, the pressurizer relief valve (a valve located at the top of the pressurizer) opened. The valve should have closed when the pressure decreased by a certain amount, but it did not. Signals available to the operator failed to show that the valve was still open. As a result, the stuck-open valve caused the pressure to continue to decrease in the system.

Meanwhile, another problem appeared elsewhere in the plant. The emergency feedwater system (backup to main feedwater) was tested 42 hours prior to the accident. As part of the test, a valve is closed and then reopened at the end of the test. But this time, through either an administrative or human error, the valve was not reopened - - preventing the emergency feedwater system from functioning. The valve was discovered closed about eight minutes into the accident. Once it was reopened, the emergency feedwater system began to work correctly, allowing cooling water to flow into the steam generators.

As the system pressure in the primary system continued to decrease, voids (areas where no water is present) began to form in portions of the system other than the pressurizer. Because of these voids, the water in the system was redistributed and the pressurizer became full of water. The level indicator, which tells the operator the amount of coolant capable of heat removal, incorrectly indicated the system was full of water. Thus, the operator stopped adding water. He was unaware that, because of the stuck valve, the indicator can, and in this instance did, provide false readings.

Because adequate cooling was not available, the nuclear fuel overheated to the point where some of the zirconium cladding (the long metal tubes or jackets which hold the nuclear fuel pellets) reacted with the water and generated hydrogen. This hydrogen was released into the reactor containment building. By March 30, two days after the start of the chain of events, some hydrogen remained within the primary coolant system in the vessel surrounding the reactor, forming a "hydrogen bubble" above the reactor core.

The concern was that if reactor pressure decreased, the hydrogen bubble would expand and thus interfere with the flow of cooling water through the core. Over the next few days, the bubble was reduced by "degassing" the pressurizer -- adjusting air and water pressure.

Without water to cool it, and with the top of the reactor core uncovered, the primary damage to the reactor occurred two to three hours into the accident. Although no "meltdown" occurred in the classic sense of the word, in that fuel did not "melt" through the floor beneath the containment or through the steel reactor vessel, a significant amount of fuel did in fact melt. Radioactivity in the reactor coolant increased dramatically, and there were small leaks in the reactor coolant system which caused high radiation levels in other parts of the plant and small releases into the environment. Shortly after the accident began, some of the water, carrying fuel debris and fission products, escaped from the reactor coolant system and flowed into the reactor building basement. By the time the accident had ended, the water in the basement had been heated by residual heat from the reactor vessel, evaporated, condensed on the walls, and drained down onto the floors and back into the basement. The radionuclides then permeated into the porous surfaces of concrete and layers of iron which later became corroded (this area of the plant became a major focus of the subsequent clean-up and decontamination).

Response to the accident was swift. The NRC's regional office in King of Prussia, Pennsylvania, was notified at 7: 45 a.m. on March 28. By 8: 00, the NRC headquarters in Washington, D.C. was alerted and the NRC Operations Center in Bethesda, Maryland, was activated. The regional office promptly dispatched the first team of inspectors to the site and other agencies, such as the Department of Energy, and the Environmental Protection Agency, also mobilized their response teams. Helicopters hired by TMI's owner, General Public Utilities Nuclear, and the Department of Energy were sampling radioactivity in the atmosphere above the plant by midday. A team from the Brookhaven National Laboratory was also sent to assist in radiation monitoring. At 9: 15 a.m., the White House was notified and at 11: 00 a.m., all non-essential personnel were ordered off the plant's premises.

From the early stages of the accident, low levels of radioactive gas, mostly in the form of xenon, continued to be released to the environment. At the time, efforts to halt the releases were unsuccessful and there was some fear of an explosion from the buildup of hydrogen - -fortunately, this did not occur. However, on Friday, March 30, Governor Thornburgh of Pennsylvania ordered a precautionary evacuation of preschool children and pregnant women from within the 5-mile zone nearest the plant, and suggested that people living within 10 miles of the plant stay inside and keep their windows closed. Most evacuees had returned to their homes by April 4 -- by that time, the situation at the reactor had been brought under control.

The American Nuclear Insurers, an organization made up of nuclear insurance firms, had already begun distributing checks to evacuees to cover hotel and meal expenses, and was beginning to handle claims for property and liability losses.

Health Effects

Detailed studies of the radiological consequences of the accident have been conducted by the NRC, the Environmental Protection Agency, the Department of Health, Education and Welfare (now Health and Human Services), the Department of Energy, and the State of Pennsylvania. Several independent studies have also been conducted. Estimates are that the average dose to about 2 million people in the area was about only about 1 millirem. To put this into context, exposure from a full set of chest x-rays is about 6 millirem. Compared to the natural radioactive background dose of about 100-125 millirem per year for the area, the collective dose to the community from the accident was very small. The maximum dose to a person at the site boundary would have been less than 100 millirem.

In the months following the accident, although questions were raised about possible adverse effects from radiation on human, animal, and plant life in the TMI area, none could be directly correlated to the accident. Thousands of environmental samples of air, water, milk, vegetation, soil, and foodstuffs were collected by various groups monitoring the area. Very low levels of radionuclides could be attributed to releases from the accident. However, comprehensive investigations and assessments by several well-respected organizations have concluded that in spite of serious damage to the reactor, most of the radiation was contained and that the actual release had negligible effects on the physical health of individuals or the environment.

Impact of the Accident

The causes of the accident continue to be debated to this day. However, based on a series of investigations, the main factors appear to have been a combination of personnel error, design deficiencies, and component failures. There is no doubt that the accident at Three Mile Island permanently changed both the nuclear industry and the NRC. Public fear and distrust increased, NRC's regulations and oversight became broader and more robust, and management of the plants was scrutinized more carefully. The problems identified from careful analysis of the events during those days have led to permanent and sweeping changes in how NRC regulates its licensees - - which, in turn, has strengthened public health and safety.

Here are some of the major changes which have occurred since the accident:

- Expansion of NRC's resident inspector program - first authorized in 1977 - whereby at least two inspectors live nearby and work exclusively at each plant in the U.S to provide daily surveillance of licensee adherence to NRC regulations;
- Establishment of a program to integrate NRC observations, findings, and conclusions about licensee performance and management effectiveness into a periodic, public report;
- Regular analysis of plant performance by senior NRC managers who identify those plants needing additional regulatory attention;
- Expansion of performance-oriented as well as safety-oriented inspections, and the use of risk assessment to identify vulnerabilities of any plant to severe accidents;
- Strengthening and reorganization of enforcement as a separate office within the NRC;
- Upgrading and strengthening of plant design and equipment requirements. This includes fire protection, piping systems, auxiliary feedwater systems, containment building isolation, reliability of individual components (pressure relief valves and electrical circuit breakers), and the ability of plants to shut down automatically;
- Identifying human performance as a critical part of plant safety, revamping operator training and staffing requirements, followed by improved instrumentation and controls for operating the plant, and establishment of fitness for duty programs for plant workers to guard against alcohol or drug abuse;
- Enhancement of emergency preparedness to include immediate NRC notification requirements for plant events and an NRC operations center which is now staffed 24 hours a day. Drills and response plans are now tested by licensees several times a year, and state and local agencies participate in drills with the Federal Emergency Management Agency and NRC;
- The installing of additional equipment by licensees to mitigate accident conditions, and monitor radiation levels and plant status;
- The establishment of the Institute of Nuclear Power Operations (INPO), the industry's own "policing" group, and formation of what is now the Nuclear Energy Institute to provide a unified industry approach to generic nuclear regulatory issues, and interaction with NRC and other government agencies;
- Employment of major initiatives by licensees in early identification of important safety-related problems, and in collecting and assessing relevant data so lessons of experience can be shared and quickly acted upon;
- Expansion of NRC's international activities to share enhanced knowledge of nuclear safety with other countries in a

number of important technical areas.

Current Status

Today, the TMI-2 reactor is permanently shut down and defueled, with the reactor coolant systemdrained, the radioactive water decontaminated and evaporated, radioactive waste shipped off-site to an approppropriate disposal site, reactor fuel and core debris shipped off-site to a Department of Energy facility, and the remainder of the site being monitored. The owner, General Public Utilities Nuclear Corporation, says it will keep the facility in long-term, monitored storage until the operating license for the TMI-1 plant expires in 2014, at which time both plants will be decommissioned.

Below is a chronology of highlights of the TMI-2 cleanup from 1980 through 1993.

Date	Event
July 1980	Approximately 43,000 curies of krypton were vented from the reactor building.
July 1980	The first manned entry into the reactor building took place.
Nov. 1980	An Advisory Panel for the Decontamination of TMI-2, composed of citizens, scientists, and State and local officials, held its first meeting in Harrisburg, PA.
July 1984	The reactor vessel head (top) was removed.
Oct. 1985	Defueling began.
July 1986	The off-site shipment of reactor core debris began.
Aug. 1988	GPU submitted a request for a proposal to amend the TMI-2 license to a "possession-only" license and to allow the facility to enter long-term monitoring storage.
Jan. 1990	Defueling was completed.
July 1990	GPU submitted its funding plan for placing $229 million in escrow for radiological decommissioning of the plant.
Jan. 1991	The evaporation of accident-generated water began.
April 1991	NRC published a notice of opportunity for a hearing on GPU's request for a license amendment.
Feb. 1992	NRC issued a safety evaluation report and granted the license amendment.
Aug. 1993	The processing of accident-generated water was completed involving 2.23 million gallons.
Sept. 1993	NRC issued a possession-only license.
Sept. 1993	The Advisory Panel for Decontamination of TMI-2 held its last meeting.
Dec. 1993	Post-Defueling Monitoring Storage began.

Additional Information

Further information on the TMI-2 accident can be obtained from sources listed below. The documents can be ordered through the Government Printing Office, at 202-512-1800 or the NRC's Public Document Room at 301-415-4737 or 1-800-397-4209; e-mail pdr@nrc.gov. The PDR is located at 1555 Rockville Pike, Rockville, Maryland; however the mailing address is: U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. A glossary is also provided at the end.

ADDITIONAL SOURCES FOR INFORMATION ON THREE MILE ISLAND

NRC Annual Report - 1979, NUREG-0690;

"Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," NUREG-0558;

"Environmental Assessment of Radiological Effluents from Data Gathering and Maintenance Operation on Three Mile Island Unit 2," NUREG-0681;

"Report of The President's Commission on The Accident at Three Mile Island," October, 1979;

"Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600;

"Three Mile Island; A Report to the Commissioners and to the Public," by Mitchell Rogovin and George T. Frampton, Vols. I-II, 1980;

"Lessons learned From the Three Mile Island - Unit 2 Advisory Panel," NUREG-6252;

"The Status of Recommendations of the President's Commission on the Accident at Three Mile Island," (A ten-year review), NUREG-1355;

"NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," NUREG-0632;

"Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28, 1979 accident Three Mile Island Nuclear Station,

Unit 2," NUREG-0683

"Answers to Questions About Updated Estimates of Occupational Radiation Doses at Three Mile Island, Unit 2," NUREG-1060;

"Answers to Frequently Asked Questions About Cleanup Activities at Three Mile Island, Unit 2," NUREG-0732;

"Status of Safety Issues at Licensed Power Plants" (TMI Action Plan Reqmts.), NUREG-1435.

Other Organizations to Contact:

GPU Nuclear Corp, One Upper Pond Road, Parsippany, NJ, 07054, telephone 201-316-7249;

Three Mile Island Public Health Fund, 1622 Locust Street, Philadelphia, PA, 19103, telephone 215-875-3026;

Pennsylvania Bureau of Radiation Protection, Department of Environmental Protection, Rachel Carson State Office Building, P.O. Box 8469, Harrisburg, PA, 17105-8469, telephone 717-787-2480.

Nuclear Energy Institute, 1776 Eye Street, N.W., Suite 400, Washington, D.C. 20006, telephone 202-739-8000.

Glossary

Auxiliary feedwater(see emergency feedwater)

Background radiation The radiation in the natural environment, including cosmic rays and radiation from the naturally radioactive elements, both outside and inside the bodies of humans and animals. The usually quoted average individual exposure from background radiation is 360 millirem per year.

CladdingThe thin-walled metal tube that forms the outer jacket of a nuclear fuel rod. It prevents the corrosion of the fuel by the coolant and the release of fission products int the coolants. Aluminum, stainless steel and zirconium alloys are common cladding materials.

Emergency feedwater

systemBackup feedwater supply used during nuclear plant startup and shutdown; also known as auxiliary feedwater.

Fuel rodA long, slender rube that holds fuel (fissionable material) for nuclear reactor use. Fuel rods are assembled into bundles called fuel elements or fuel assemblies, which are loaded individually into the reactor core.

ContainmentThe gas-tight shell or other enclosure around a reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.

CoolantA substance circulated through a nuclear reactor to remove or transfer heat. The most commonly used coolant in the U.S. is water. Other coolants include air, carbon dioxide, and helium.

Core - The central portion of a nuclear reactor containing the fuel elements, and control rods.

Decay heat - The heat produced by the decay of radioactive fission products after the reactor has been shut down.

Decontamination - The reduction or removal of contaminating radioactive material from a structure, area, object, or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination; (2) letting the material stand so that the radioactivity is decreased by natural decay; and (3) covering the contamination to shield the radiation emitted.

Feedwater - Water supplied to the steam generator that removes heat from the fuel rods by boiling and becoming steam. The steam then becomes the driving force for the turbine generator.

Nuclear Reactor - A device in which nuclear fission may be sustained and controlled in a self-supporting nuclear reaction. There are several varieties, but all incorporate certain features, such as fissionable material or fuel, a moderating material (to control the reaction), a reflector to conserve escaping neutrons, provisions for removal of heat, measuring and controlling instruments, and protective devices

Pressure VesselA strong-walled container housing the core of most types of power reactors.

PressurizerA tank or vessel that controls the pressure in a certain type of nuclear reactor.

Primary System The cooling system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine.

RadiationParticles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of an unstable atom as a result of radioactive decay.

Reactor Coolant System(see primary system)

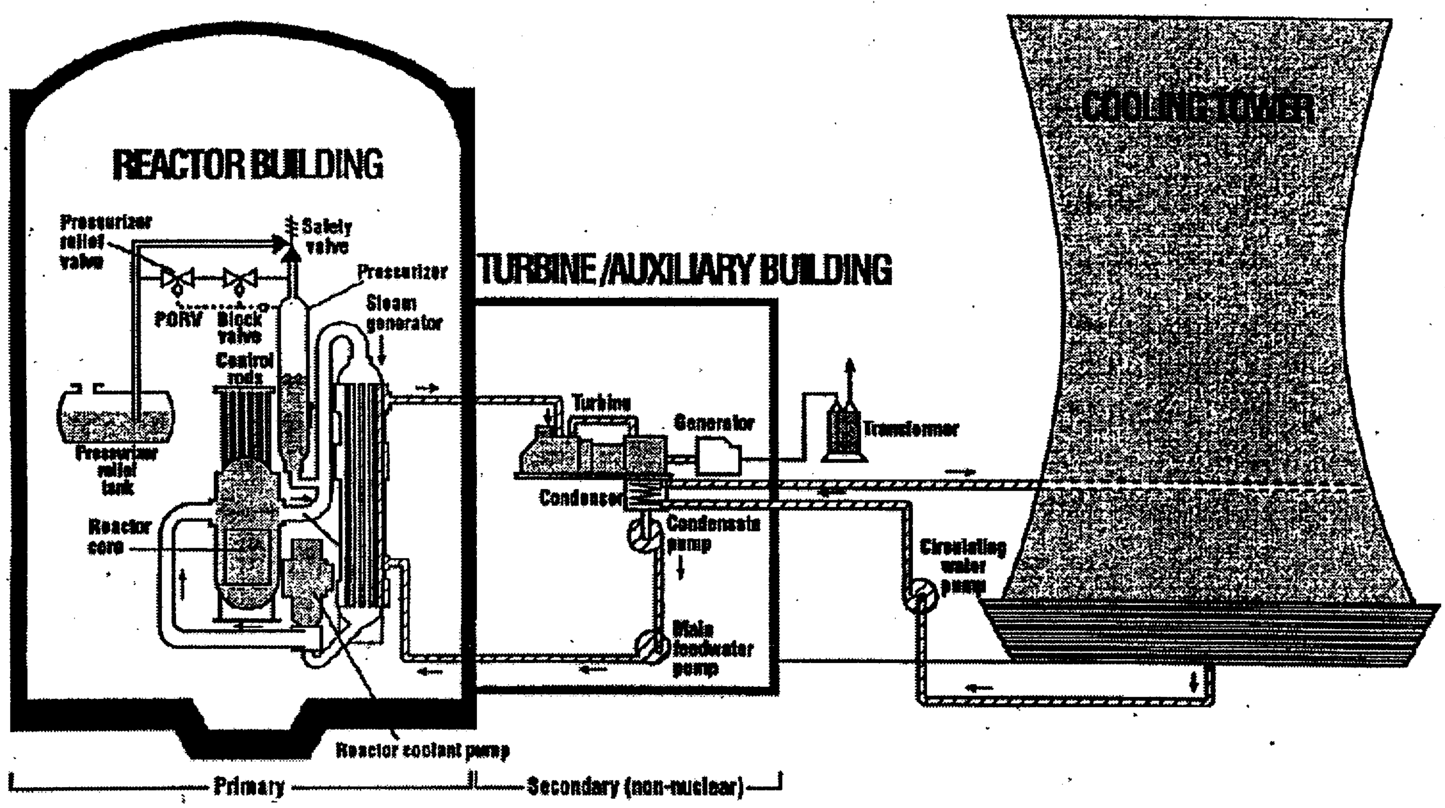
Secondary System The steam generator tubes, steam turbine, condenser and associated pipes, pumps, and heaters used to convert the heat energy of the reactor coolant system into mechanical energy for electrical generation.

Steam Generator The heat exchanger used in some reactor designs to transfer heat from the primary (reactor coolant) system to the secondary (steam) system. This design permits heat exchange with little or no contamination of the secondary system equipment.

TurbineA rotary engine made with a series of curved vanes on a rotating shaft. Usually turned by water or steam. Turbines are considered to be the most economical means to turn large electrical generators.

January 2001

TMI-2



1. The catastrophic Chernobyl accident in the former Soviet Union, in 1986, was by far the most severe nuclear reactor accident to occur in any country; it is widely believed an accident of that type could not occur in U.S.-designed plants. Chernobyl is discussed in more detail on the internet, at http://www.nrc.gov/OPA/gmo/tip/fschernobyl.html