EXHIBIT 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE APPEAL BOARD

In The Matter Of PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Units Nos. 1 and 2)	Docket Nos. 50-275 O.L. 50-323 O.L.

AFFIDAVIT OF RICHARD B. HUBBARD

RICHARD B. HUBBARD, being duly sworn, deposes and says as follows:

1. The purpose of this affidavit is threefold. First, to estimate the elapsed time which is likely to be required after issuance of a low power operating license to load fuel and to complete the special low power tests at or below 5% of Rated Thermal Power as Pacific Gas and Electric Company has proposed for the Diablo Canyon Unit 1; second, to describe the substantial fission product inventory that would be created in less than one month of 5 percent power operation; and third, to identify the technical difficulties and increased costs associated with modifying the structures, systems, and components of the plant should further modifications be required after fuel has been loaded and operation commenced. A recent statement of my professional qualifications and experience is attached hereto as Appendix A.

2. In preparing this affidavit, I have reviewed PG&E's proposed special low power test program as set forth in the low power license application and as further described in PG&E's safety analysis report provided to the NRC Staff on February 6, 1981. I also attended, as a consultant to Governor Brown's counsel, all sessions of the recent low power test proceedings which were held in San Luis Obispo from May 19 to May 22, 1981. Thus, I am familiar with the duration of the low power tests as postulated by PG&E and Staff witnesses. Further, I have reviewed the actual schedule for fuel loading, initial criticality and zero power testing, and low power testing of large pressurized water reactors (PWR's) which have occurred in the post-TMI period, particularly North Anna-2, Salem-2, and Sequoyah-1. In addition, on July 10, 1981, I accompanied NRC Commissioner Gilinsky on his tour of the Diablo Canyon facility. The results of my review are summarized in the following paragraphs.

A. <u>INITIAL CRITICALITY AND DURATION OF LOW POWER TEST PROGRAM</u>

3. During Commissioner Gilinsky's tour of the Diablo Canyon facility, both NRC and PG&E personnel emphasized PG&E's readiness to load fuel. The necessary fuel is presently on site in a building immediately adjacent to the Containment Building. Further, due to the duration of the licensing process, PG&E has had sufficient time to conduct, and in some cases reconduct,

its pre-operational tests as set forth in Section 14.1 of the Final Safety Analysis Report ("FSAR"). Thus, I conclude that Diablo Canyon Unit 1 equipment is in an advanced state of readiness to load fuel, and that virtually all preliminary testing such as that described in the FSAR Table 14.1-1 possible prior to fuel loading has been completed. */ Further, I conclude that PG&E should be able to promptly load fuel once such authorization is received from the NRC.

4. I estimate that the fuel loading task should be completed in less than one week elapsed time. For example, at Salem-2, a Westinghouse-designed PWR similar in design and rating to Diablo Canyon, fuel loading began on May 23, 1980 and was completed on May 27, 1980. Following fuel loading, the Precritical Test Program of eleven tests, as set forth by PG&E in Table 14.1-2 of the Diablo Canyon FSAR, should require no more than two weeks to complete. Thus, there is no technical reason that initial criticality could not be achieved within two weeks after fuel loading is completed. Therefore, I conclude that it is reasonable to expect that the fuel loading and precritical test program could be completed in no more than 30 days after the issuance of a low power test license. The reactor could be made critical immediately thereafter.

---

*/ A recent <u>Nucleonics Week</u> article indicated that all steps prior to fuel load will be completed by approximately August 12, 1981 (p. 4, July 23, 1981). In general, all pre-operational testing will be completed before fuel loading (FSAR, p. 14.1-8).

5. The next phase of startup and testing includes initial criticality (i.e., commencement of the nuclear reaction) and testing (of the reactor at power levels up to 5 percent of rated capacity). FSAR Table 14.1-2 summarizes the normal tests which will be performed. In addition, the scope and duration of the special low power tests were described in detail during the recent low power proceedings in San Luis Obispo. The Licensing Board, in the Partial Initial Decision dated July 17, 1981, noted at page 24, paragraph 61, that PG&E has proposed a series of eight special low power tests. The proposed tests would probably last for no more than one month and in actuality, as cited by the Board, would perhaps only take about eighteen days (Tr. 10,826-10,728). Other references to the "relatively few days" encompassed by the proposed low power test program are set forth in the recent decision by the Board at page 25 (paragraph 65), page 32 (paragraph 82), and page 33 (paragraph 83). Therefore, I believe that it is reasonable to expect that, absent major problems or absent discretionary delay by PG&E (for instance, to conduct some other tests), initial criticality can be achieved and low power testing can be conducted in an elapsed time of less than 30 days. Thus, assuming a 30-day period for fuel loading and precritical testing, the entire fuel load and testing program can readily be completed in no more than 60 days.

6. The reasonableness of a 60-day cycle from license issuance to completion of the special low power tests was further confirmed during Commissioner Gilinsky's tour of the Diablo Canyon facility. In response to a question, the Diablo Canyon Plant Manager, Robert C. Thornberry, stated in my presence that PG&E's current schedules forecast that fuel loading, zero power testing, and the special low power test program will be completed approximately 58 days after receipt of a low power license. Mr. Thornberry added that the schedule might need to be increased if major unanticipated problems were encountered during the test program.

7. In order to be conservative, I believe it may be appropriate to add 15 to 30 days to the fuel loading and low power testing schedule to allow time for resolution of any routine unanticipated events. Thus, at the outside, I would expect the entire low power program at Diablo Canyon to take no more than 90 days. I understand that the NRC Staff recently indicated that the entire program would be completed in 101 days, which I feel is consistent with the schedule set forth herein. */

8. The post-TMI experience and the current schedules for startup testing lend further support to the preceding conclusions. The first plant granted an operating license in the post-TMI period was Sequoyah-1, which received a low power

---

*/ See Attachment to Transcript of NRC Commissioner Briefing of August 27, 1981.

license on February 29, 1980. Fuel loading commenced on March 2, 1980 and was completed on March 8, 1980. Two major problems thereafter seriously delayed the initial criticality of Sequoyah-1. First, in response to I&E Bulletin 79-14, TVA required approximately 60 days to inspect and rework pipe hangers and supports. Second, in parallel with the hanger reinspection, TVA conducted a base line inspection of the turbine blades. The turbine reinspection required 4-5 weeks of elapsed time. Routine maintenance problems and pre-operational testing resulted in further delays. Initial criticality was achieved on July 5, 1980. Following zero power testing, the special low power testing program began on July 12 and was completed on July 18, 1980.

9. The second plant to receive a post-TMI license to load fuel and conduct special low power tests was North Anna-2. The authorization to load fuel was issued on April 11, 1980 and the low power testing was completed by July 1, 1980, an elapsed time of less than 80 days.

10. The Salem-2 low power license was issued on April 18, 1980. As set forth in paragraph 4, fuel loading was completed on May 27, 1980. Initial criticality was achieved on August 2, 1980. The two months delay between fuel loading and initial criticality was largely due to the need to conduct routine pre-operational maintenance testing and surveillance testing (such

as valve operability) which could have been accomplished prior to fuel load. As presented in paragraph 3, I believe that these pre-operational tests will be accomplished at Diablo Canyon prior to fuel loading. Thus, I conclude that the actual duration of the Salem-2, North Anna-2, and Sequoyah-1 fuel loading and low power testing programs is not inconsistent with my conclusions for Diablo Canyon as set forth herein.

B. FISSION PRODUCT HAZARD

11. There is sufficient evidence in the record of the recent low power test proceeding to show that the consequences of a severe accidental release during low power operation would be serious. The basis for my views are as follows: First, Table I of the testimony of Applicant's witness, Dr. Brunot, sets forth the fission product inventories which will be produced in the core during the proposed Diablo Canyon LPTP. The inventory of iodine-131, one of the radionuclides which is a significant contributor to the dominant exposure modes for accidents requiring off-site emergency preparedness, is estimated by Dr. Brunot as 4,500,000 curies (approximately 1/20th the full power value as set forth in FSAR Table 11.1-4). In contrast, for the design basis LOCA addressed by the Applicant in the FSAR, only 192 curies of iodine-131 were postulated to be released to the environment in the first two hours. The corresponding two-hour thyroid doses cited in the FSAR are as follows:

Nuclide	Activity Released*/ (Curies)	Thyroid Doses (Rem)**/ 800 (Meters)	10,000 (Meters)
I-131	27.0	7.3	0.3
I-131 ORG	73.4	19.9	0.8
I-131 PAR	91.8	24.9	1.0
TOTALS:	192.2	52.1	2.1

12. Furthermore, in the Diablo Canyon Emergency Plan***/ the Applicant has calculated that if the equivalent of 1000 curies of iodine-131 were to be released during a "Site Emergency" class****/ accident, and assuming the design basis meteorological conditions, then the thyroid dose at the plume centerline would be as follows:

Nuclide	Activity Released (Curies)	Thyroid Doses (Rem) 800 (Meters)	10,000 (Meters)
I-131	1000	270	12

The preceding relationships between releases and exposures are all based on numbers in the record in the low power proceeding. By observation, it can be inferred that the thyroid doses can

*/ FSAR Table 15.5-12 (attached hereto as Appendix B).
**/ FSAR Table 15.5-14 (attached hereto as Appendix C).
***/ Emergency Plan, p. 4-5 (attached hereto as Appendix D).
****/ The release potential and significance for a larger class of accidents, the "General Emergency," were not quantified by the Applicant in the Diablo Canyon Emergency Plan.

be scaled approximately linearly with fission product releases. This relationship is not surprising in that Dr. Brunot stated in his testimony that estimated exposure is directly proportional to the core inventory which could contribute to that exposure.*/ (We believe he must be assuming a constant release fraction). Brunot further estimated exposure levels by scaling exposures linearly based on the reduced fission product inventories at LP as compared to the FP operation.**/ Thus, using the Brunot scaling methodology, and assuming release fractions of 1.0 percent or 0.1 percent, the exposures for an accident during the Diablo Canyon LPTP can reasonably be extrapolated approximately as follows:

Nuclide	Activity Released (Curies)	Thyroid Doses (Rem) 800 (Meters)	10,000 (Meters)
I-131	4,500 (0.1%)	1,221	49
I-131	45,000 (1.0%)	12,211	492

In either of the preceding cases, the potential thyroid exposures appear to be of significant magnitude. Thus, the next question is whether the postulated release fractions are reasonable.

13. The probabilities for nine major PWR release categories (PWR-1 to PWR-9) were developed in the NRC's Reactor Safety Study (WASH-1400).***/ The event sequences in PWR-1-7 lead to

---

*/ Brunot Testimony, p. 11.

**/ Brunot Testimony, p. 12.

***/ The dominant PWR accident sequences from WASH-1400 for each of the release categories are set forth in Appendix E which is attached hereto.

partial or complete melting of the reactor core while those in the last two categories do not involve melting of the core. These severe accidents can be distinguished from design basis accidents in that they involve deterioration of the capability of the containment structure to perform its intended function of limiting the release of radioactive materials to the environment. In release categories 1 to 3, the event sequences include containment failure by steam explosion, hydrogen burning, or overpressure. In release categories 6 and 7, the dominant containment failure mode is by melt-through of the containment base mat. The other release categories contain event sequences in which the systems intended to isolate the containment fail to act properly. The uncertainties in the absolute values of the probabilities are significant. The error band for the probabilities of some of the event sequences could be as great as a factor of 100 as discussed by Staff witness Lauben in the low power proceeding. The containment releases postulated in WASH-1400 are described in more detail in Appendix F which is attached hereto. It is important to note that the magnitudes (curies) of radioactive releases for each PWR category are obtained by multiplying the release fractions shown in Table VI 2-1 of Appendix F by the amounts of radionuclides that would be present in the core at the time of the hypothetical accident (for Diablo Canyon LP inventory, see Table I of Brunot testimony). For

example, if one started with the iodine-131 inventory of 4,500,000 curies calculated by Brunot and the release fractions set forth by the WASH-1400 authors, the magnitude of the iodine releases for each of the nine PWR accidents, if it occurred during the proposed Diablo Canyon LPTP, would be as follows:

PWR Release Category	Release Fractions	Activity Released (Curies)
1	0.70	3,150,000
2	0.70	3,150,000
3	0.20	900,000
4	0.09	405,000
5	0.03	135,000
6	8×10^{-4}	3,600
7	2×10^{-5}	90
8	1×10^{-4}	450
9	1×10^{-7}	0.45

14. Several conclusions are obvious. First, the 1.0% release fraction postulated herein is exceeded by a factor of 3 to 70 for WASH-1400 release Categories 1 through 5. The 0.1% release is consistent with a Category 6 release occurring during LP operation. Thus, I conclude that the proposed 1.0% and 0.1% release fractions are conservative representations of the potential releases.*/ Therefore, because of the relatively rapid buildup (half-life of hours to days) of the radioactive isotopes

*/ Indeed, the NRC indicated recently that the possession of as little as 3.3 curies of I-131 constitutes a sufficient amount to be "of potential significant concern in the event of a major accident....." 46 Federal Register 29714 (June 3, 1981). The I-131 inventory after one month of low power operation of Diablo Canyon will be 4.5 million curies, or more than one million times greater than the NRC's recently stated threshold level of concern.

listed in Table 3 of NUREG-0654*/ which dominate prompt health consequences resulting from postulated accidental releases, I conclude that even at 5% power after less than 30 days the fission products available for release pose a significant potential hazard.

C. PLANT CONTAMINATION

15. Operation at low power will not only cause a buildup of fission products within the reactor core, making it inaccessible for contact repair and/or modification, but will also cause a spread of radioactive contaminants throughout the primary portion of the steam supply system. It will also contaminate certain auxiliary systems such as the Chemical and Volume Control System, Equipment and Floor Drainage Systems, and the Liquid Radioactive Waste System. If fuel failures and/or steam generator tube failures or leaks are experienced, a large number of other systems, including the turbine, condensate, and other components within the Steam and Power Conversion System could become contaminated. Contamination and irradiation of such equipment greatly increases the care required and the time and cost of future modifications that could be required at Diablo Canyon. I conclude, therefore, that it is important that power operation, including low power testing, not be permitted until reviews and evaluations that could lead to required plant modifications have been completed.

---

*/ NUREG-0654, Rev. 1 (FEMA-REP-1), Criteria for Preparation and and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, November, 1980.

D. <u>CONCLUSION</u>

16. Based on the foregoing, I conclude: (a) that fuel loading, initial criticality, and low power testing, including the special low power tests, can be accomplished at Diablo Canyon Unit 1 within approximately 60 days, with an outside maximum elapsed time of approximately 90 days, after issuance of the low power operating license; (b) that it is feasible for fuel loading to be completed within one week after issuance of the low power license; and (c) that the fuel loading and pre-critical testing portion of the startup schedule should be completed within less than 30 days following issuance of the low power license and that immediately thereafter initial criticality could be achieved. Further, I conclude that because of the relatively rapid buildup of the radioactive isotopes which dominate health consequences, even at 5% power the fission products such as iodine-131 available for release pose a significant hazard. Finally, I conclude that operation at low power will contaminate some of the facility's components and systems. This unnecessary commitment of resources creates technical difficulties and increased costs associated with modifying the reactor, should further modification be required <u>after</u> fuel has been loaded and power operation commenced.

I have read the foregoing and swear that it is true and accurate to the best of my knowledge.

Richard B Hubbard

RICHARD B. HUBBARD

Subscribed and sworn to before me this 9 day of September, 1981.

James F. Lehman

NOTARY PUBLIC

OFFICIAL SEAL
JAMES F LEHMAN
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires AUG 21, 1984

My commission expires 8/21/84

APPENDIX A

PROFESSIONAL QUALIFICATIONS OF RICHARD B. HUBBARD

RICHARD B. HUBBARD
MHB Technical Associates
1723 Hamilton Avenue
Suite K
San Jose, California 95125
(408) 266-2716

EXPERIENCE:

9/76 - PRESENT

Vice-President - MHB Technical Associates, San Jose, California. Founder, and Vice-President of technical consulting firm. Specialists in independent energy assessments for government agencies, particularly technical and economic evaluation of nuclear power facilities. Consultant in this capacity to Oklahoma and Illinois Attorney Generals, Minnesota Pollution Control Agency, German Ministry for Research and Technology, Governor of Colorado, Swedish Energy Commission, Swedish Nuclear Inspectorate, and the U.S. Department of Energy. Also provided studies and testimony for various public interest groups including the Center for Law in the Public Interest, Los Angeles; Public Law Utility Group, Baton Rouge, Louisiana; Friends of the Earth (FOE), Italy; and the Union of Concerned Scientists, Cambridge, Massachusetts. Provided testimony to the U.S. Senate/House Joint Committee on Atomic Energy, the U.S. House Committee on Interior and Insular Affairs, the California Assembly, Land Use, and Energy Committee, the Advisory Committee on Reactor Safeguards, and the Atomic Safety and Licensing Board. Performed comprehensive risk analysis of the accident probabilities and consequences at the Barseback Nuclear Plant for the Swedish Energy Commission and edited, as well as contributed to, the Union of Concerned Scientist's technical review of the NRC's Reactor Safety Study (WASH-1400).

2/76 - 9/76

Consultant, Project Survival, Palo Alto, California. Volunteer work on Nuclear Safeguards Initiative campaigns in California, Oregon, Washington, Arizona, and Colorado. Numerous presentations on nuclear power and alternative energy options to civic, government, and college groups. Also resource person for public service presentations on radio and television.

5/75 - 1/76

Manager - Quality Assurance Section, Nuclear Energy Control and Instrumentation Department, General Electric Company, San Jose, California.
Report to the Department General Manager. Develop and implement quality plans, programs, methods, and equipment which assure that products produced by the Department meet quality requirements as defined in NRC regulation 10 CFR 50, Appendix B, ASME Boiler and Pressure Vessel Code, customer contracts, and GE Corporate policies and procedures. Product areas include radiation sensors, reactor vessel internals, fuel handling and servicing tools, nuclear plant control and protection instrumentation systems, and nuclear steam supply and Balance of Plant control room panels. Responsible for approximately 45 exempt personnel, 22 non-exempt personnel, and 129 hourly personnel with an expense budget of nearly 4 million dollars and equipment investment budget of approximately 1.2 million dollars.

11/71 - 5/75

Manager - Quality Assurance Subsection, Manufacturing Section of Atomic Power Equipment Department, General Electric Company, San Jose, California.
Report to the Manager of Manufacturing. Same functional and product responsibilities as in Engagement #1, except at a lower organizational report level. Developed a quality system which received NRC certification in 1975. The system was also successfully surveyed for ASME "N" and "NPT" symbol authorization in 1972 and 1975, plus ASME "U" and "S" symbol authorizations in 1975. Responsible for from 23 to 39 exempt personnel, 7 to 14 non-exempt personnel, and 53 to 97 hourly personnel.

3/70 - 11/71

Manager - Application Engineering Subsection, Nuclear Instrumentation Department, General Electric Company, San Jose, California.
Responsible for the post order technical interface with architect engineers and power plant owners to define and schedule the instrumentation and control systems for the Nuclear Steam Supply and Balance of Plant portion of nuclear power generating stations. Responsibilities included preparation of the plant instrument list with approximate location, review of interface drawings to define functional design requirements, and release of functional requirements for detailed equipment designs. Personnel supervised included 17 engineers and 5 non-exempt personnel.

12/69 - 3/70

Chairman - Equipment Room Task Force, Nuclear Instrumentation Department, General Electric Company, San Jose, California. Responsible for a special task force reporting to the Department General Manager to define methods to improve the quality and reduce the installation time and cost of nuclear power plant control rooms. Study resulted in the conception of a factory-fabricated control room consisting of signal conditioning and operator control panels mounted on modular floor sections which are completely assembled in the factory and thoroughly tested for proper operation of interacting devices. Personnel supervised included 10 exempt personnel.

12/65 - 12/69

Manager - Proposal Engineering Subsection, Nuclear Instrumentation Department, General Electric Company, San Jose, California. Responsible for the application of instrumentation systems for nuclear power reactors during the proposal and pre-order period. Responsible for technical review of bid specifications, preparation of technical bid clarifications and exceptions, definition of material list for cost estimating, and the "as sold" review of contracts prior to turnover to Application Engineering. Personnel supervised varied from 2 to 9 engineers.

8/64 - 12/65

Sales Engineer, Nuclear Electronics Business Section of Atomic Power Equipment Department, General Electric Company, San Jose, California. Responsible for the bid review, contract negotiation, and sale of instrumentation systems and components for nuclear power plants, test reactors, and radiation hot cells. Also responsible for industrial sales of radiation sensing systems for measurement of chemical properties, level, and density.

10/61 - 8/64

Application Engineer, Low Voltage Switchgear Department, General Electric Company, Philadelphia, Pennsylvania. Responsible for the application and design of advanced diode and silicon-controlled rectifier constant voltage DC power systems and variable voltage DC power systems for industrial applications. Designed, followed manufacturing and personally tested an advanced SCR power supply for product introduction at the Iron and Steel Show. Project Engineer for a DC power system for an aluminum pot line sold to Anaconda beginning at the 161KV switchyard and encompassing all the equipment to convert the power to 700 volts DC at 160,000 amperes.

9/60 - 10/61

GE Rotational Training Program

Four 3-month assignments on the GE Rotational Training Program for college technical graduates as follows:

a. Installation and Service Eng. - Detroit, Michigan. Installation and startup testing of the world's largest automated hot strip steel mill.

b. Tester - Industry Control - Roanoke, Virginia. Factory testing of control panels for control of steel, paper, pulp, and utility mills and power plants.

c. Engineer - Light Military Electronics - Johnson City, New York. Design of ground support equipment for testing the auto pilots on the F-105.

d. Sales Engineer - Morrison, Illinois. Sale of appliance controls including range timers and refrigerator cold controls.

EDUCATION:

Bachelor of Science Electrical Engineering, University of Arizona, 1960.

Master of Business Administration, University of Santa Clara, 1969.

PROFESSIONAL AFFILIATION:

Registered Quality Engineer, License No. QU805, State of California.

Member of Subcommittee 8 of the Nuclear Power Engineering Committee of the IEEE Power Engineering Society responsible for the preparation and revision of the following 3 national Q.A. Standards:

a. IEEE 498 (ANSI N45.2.16): Requirements for the Calibration and Control of Measuring and Test Equipment used in the Construction and Maintenance of Nuclear Power Generating Stations.

PROFESSIONAL AFFILIATION: (Contd)

b. IEEE 336 (ANSI N45.2.4): Installation, Inspection, and Testing Requirements for Class 1E Instrumentation and Electric Equipment at Nuclear Power Generating Stations.

c. IEEE 467 : Quality Assurance Program Requirements for the Design and Manufacture of Class IE Instrumentation and lectric Equipment for Nuclear Power Generating Stations.

I am currently a member of the IEEE Ad Hoc Committee which recommended the issues to be addressed in the development of a standard relating to the selection and utilization of replacement parts for Class IE equipment during the construction and operation phase. I am also a member of the work group which ll prepare this proposed standard.

PERSONAL DATA:

Birth Date: 7/08/37
Married; three children
Health: Excellent

PUBLICATIONS AND TESTIMONY:

1. In-Core System Provides Continuous Flux Map of Reactor Cores, R.B. Hubbard and C.E. Foreman, Power, November, 1967.

2. Quality Assurance: Providing It, Proving It, R.B. Hubbard, Power, May, 1972.

3. Testimony of R.B. Hubbard, D.G. Bridenbaugh, and G.C. Minor before the United States Congress, Joint Committee on Atomic Energy, February 18, 1976, Washington, DC. (Published by the Union of Concerned Scientists, Cambridge, Massachusetts.) Excerpts from testimony published in Quote Without Comment, Chemtech, May, 1976.

4. Testimony of R.B. Hubbard, D.G. Bridenbaugh, and G.C. Minor to the California State Assembly Committee on Resources, Land Use, and Energy, Sacramento, California, March 8, 1976.

5. Testimony of R. B. Hubbard and G.C. Minor before California State Senate Committee on Public Utilities, Transit, and Energy, Sacramento, California, March 23, 1976.

6. Testimony or R.B. Hubbard and G.C. Minor, Judicial Hearings Regarding Grafenrheinfeld Nuclear Plant, March 16 & 17, 1977, Wurzburg, Germany.

PUBLICATIONS AND TESTIMONY: (Contd)

7. Testimony of R.B. Hubbard to United States House of Representatives, Subcommittee on Energy and the Environment, June 30, 1977, Washington, DC, entitled, Effectiveness of NRC Regulations - Modifications to Diablo Canyon Nuclear Units.

8. Testimony of R.B. Hubbard to the Advisory Committee on Reactor Safeguards, August 12, 1977, Washington, DC, entitled, Risk Uncertainty Due to Deficiencies in Diablo Canyon Quality Assurance Program and Failure to Implement Current NRC Practices.

9. The Risks of Nuclear Power Reactors: A Review of the NRC Reactor Safety Study WASH-1400, Kendall, et al, edited by R.B. Hubbard and G.C. Minor for the Union of Concerned Scientists, August, 1977.

10. Swedish Reactor Safety Study: Barsebäck Risk Assessment, MHB Technical Associates, January 1978 (Published by Swedish Department of Industry as Document DSI 1978:1).

11. Testimony of R.B. Hubbard before the Energy Facility Siting Council, March 31, 1978, in the matter of Pebble Springs Nuclear Power Plant, Risk Assessment: Pebble Springs Nuclear Plant, Portland, Oregon.

12. Presentation by R.B. Hubbard before the Federal Ministry for Research and Technology (BMFT), August 31 and September 1, 1978, Meeting on Reactor Safety Research, Risk Analysis, Bonn, Germany.

13. Testimony by R.B. Hubbard, D.G. Bridenbaugh, and G.C. Minor before the Atomic Safety and Licensing Board, September 25, 1978, in the matter of the Black Fox Nuclear Power Station Construction Permit hearings, Tulsa, Oklahoma.

14. Testimony of R.B. Hubbard before the Atomic Safety and Licensing Board, November 17, 1978, in the matter of Diablo Canyon Nuclear Power Plant Operating License Hearings, Operating Basis Earthquake and Seismic Analysis of Structures, Systems, and Components, Avila Beach, California.

15. Testimony of R.B. Hubbard and D.G. Bridenbaugh before the Louisiana Public Service Commission, November 19, 1978, Nuclear Plant and Power Generation Costs, Baton Rouge, Louisiana.

16. Testimony of R.B. Hubbard before the California Legislature, Subcommittee on Energy, Los Angeles, April 12, 1979.

PUBLICATIONS AND TESTIMONY: (Contd)

17. Testimony of R.B. Hubbard and G.C. Minor before the Federal Trade Commission, on behalf of the Union of Concerned Scientists, Standards and Certification Proposed Rule 16 CFR Part 457, May 18, 1979.

18. ALO-62, Improving the Safety of LWR Power Plants, MHB Technical Associates, prepared for U.S. Department of Energy, Sandia National Laboratories, September, 1979, available from NTIS.

19. Testimony by R.B. Hubbard before the Arizona State Legislature, Special Interim House Committee on Atomic Energy, Overview of Nuclear Safety, Phoenix, AZ, September 20, 1979.

20. "The Role of the Technical Consultant," Practising Law Institute program on "Nuclear Litigation," New York City and Chicago, November, 1979. Available from PLI, New York City.

21. Uncertainty in Nuclear Risk Assessment Methodology, MHB Technical Associates, January, 1980, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

22. Italian Reactor Safety Study: Caorso Risk Assessment, MHB Technical Associates, March, 1980, prepared for and available from Friends of the Earth, Rome, Italy.

23. Development of Study Plans: Safety Assessment of Monticello and Prairie Island Nuclear Stations, MHB Technical Associates, August, 1980, prepared for and available from the Minnesota Pollution Control Agency.

24. Affidavit of Richard B. Hubbard and Gregory C. Minor before the Illinois Commerce Commission, In the Matter of an Investigation of the Plant Construction Program of the Commonwealth Edison Company, prepared for the League of Woman Voters of Rockford, Illinois, November 12, 1980, ICC Case No. 78-0646.

25. Systems Interaction and Single Failure Criterion, MHB Technical Associates, January, 1981, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

APPENDIX B

(Source: Diablo Canyon FSAR)

TABLE 15.5-12

CALCULATED ACTIVITY RELEASES FROM LOCA- DESIGN BASIS CASE (CURIES)

NUCLIDE	0-2 Hrs	2-8 Hrs	8-24 Hrs	24-96 Hrs	4-30 Days
I-131	0.2702E 02	0.0	0.0	0.0	0.0
I-132	0.3985E 02	0.0	0.0	0.0	0.0
I-133	0.6207E 02	0.0	0.0	0.0	0.0
I-134	0.7663E 02	0.0	0.0	0.0	0.0
I-135	0.5712E 02	0.0	0.0	0.0	0.0
I-131ORG	0.7340E 02	0.2170E 03	0.5561E 03	0.1070E 04	0.3227E 04
I-132ORG	0.8325E 02	0.6763E 02	0.1862E 02	0.9240E-01	0.6557E-10
I-133ORG	0.1639E 03	0.4314E 03	0.8078E 03	0.5263E 03	0.5383E 02
I-134ORG	0.9847E 02	0.2469E 02	0.2045E 00	0.2811E-06	0.2665E-31
I-135ORG	0.1411E 03	0.2838E 03	0.2668E 03	0.3148E 02	0.1834E-01
I-131PAR	0.9175E 02	0.2713E 03	0.6951E 03	0.1338E 04	0.4033E 04
I-132PAR	0.1041E 03	0.1095E 03	0.2327E 02	0.1155E 00	0.1070E-09
I-133PAR	0.2048E 03	0.5392E 03	0.1010E 04	0.6579E 03	0.6728E 02
I-134PAR	0.1231E 03	0.3086E 02	0.2557E 00	0.3514E-06	0.3331E-31
I-135PAR	0.1764E 03	0.3548E 03	0.3335E 03	0.3935E 02	0.2293E-01
KR-83M	0.9260E 03	0.7487E 03	0.8940E 02	0.1154E 00	0.2578E-12
KR-85	0.6279E 02	0.1913E 03	0.5097E 03	0.1145E 04	0.9827E 04
KR-85M	0.2623E 04	0.4660E 04	0.2723E 04	0.1191E 03	0.1413E-02
KR-87	0.3847E 04	0.1864E 04	0.7273E 02	0.5752E-02	0.4530E-19
KR-88	0.7090E 04	0.8484E 04	0.2388E 04	0.2220E 02	0.3333E-06
XE-133	0.1664E 05	0.4942E 05	0.1241E 06	0.2205E 06	0.4392E 06
XE-133M	0.4780E 03	0.1212E 04	0.2819E 04	0.3766E 04	0.2556E 04
XE-135	0.7402E 04	0.1655E 05	0.2028E 05	0.4316E 04	0.1910E 02
XE-135M	0.8506E 03	0.4137E 01	0.4690E-06	0.7061E-25	0.0
XE-138	0.2564E 04	0.6552E 01	0.1164E-06	0.1252E-27	0.0

TABLE 15.5-14

THYROID DOSE - TWO HOUR - CONTAINMENT LEAKAGE - DESIGN BASIS CASE (REM)

DISTANCE FROM RELEASE POINT

NUCLIDE	800M	1200M	2000M	4000M	7000M	10000M	20000M
I-131	0.7342E 01	0.4715E 01	0.2555E 01	0.1080E 01	0.4983E 00	0.3053E 00	0.1728E 00
I-132	0.3913E 00	0.2515E 00	0.1383E 00	0.5755E-01	0.2656E-01	0.1627E-01	0.6547E-02
I-133	0.4556E 01	0.2929E 01	0.1611E 01	0.6703E 00	0.3093E 00	0.1895E 00	0.7625E-01
I-134	0.3241E 00	0.2083E 00	0.1146E 00	0.4767E-01	0.2200E-01	0.1348E-01	0.5422E-02
I-135	0.1300E 01	0.8356E 00	0.4596E 00	0.1912E 00	0.8823E-01	0.5407E-01	0.2175E-01
I-131ORG	0.1994E 02	0.1282E 02	0.7049E 01	0.2933E 01	0.1353E 01	0.8292E 00	0.3336E 00
I-132ORG	0.8176E 00	0.5255E 00	0.2890E 00	0.1202E 00	0.5548E-01	0.3400E-01	0.1368E-01
I-133ORG	0.1203E 02	0.7734E 01	0.4253E 01	0.1770E 01	0.8166E 00	0.5004E 00	0.2013E 00
I-134ORG	0.4519E 00	0.2904E 00	0.1597E 00	0.6646E-01	0.3067E-01	0.1879E-01	0.7560E-02
I-135ORG	0.3211E 01	0.2064E 01	0.1135E 01	0.4723E 00	0.2179E 00	0.1336E 00	0.5372E-01
I-131PAR	0.2492E 02	0.1602E 02	0.8811E 01	0.3666E 01	0.1692E 01	0.1037E 01	0.4170E 00
I-132PAR	0.1022E 01	0.6568E 00	0.3613E 00	0.1503E 00	0.6936E-01	0.4250E-01	0.1710E-01
I-133PAR	0.1504E 02	0.9667E 01	0.5317E 01	0.2212E 01	0.1021E 01	0.6255E 00	0.2518E 00
I-134PAR	0.5648E 00	0.3630E 00	0.1997E 00	0.8307E-01	0.3833E-01	0.2349E-01	0.9450E-02
I-135PAR	0.4014E 01	0.2580E 01	0.1419E 01	0.5904E 00	0.2724E 00	0.1669E 00	0.6715E-01
TOTAL	0.9593E 02	0.6166E 02	0.3391E 02	0.1411E 02	0.6511E 01	0.3990E 01	0.1605E 01

(Source: Diablo Canyon FSAR)

APPENDIX D

(Source: Diablo Canyon Emergency Plan)

4.1.3 Site Emergency

4.1.3.1 Description

The Site Emergency action level reflects conditions where there is a clear potential for significant releases, such releases are likely, or they are occurring, but in all cases where a core meltdown situation is not indicated based on current information. Because the possible release associated with a Site Emergency is significant, care must be taken in alerting offsite authorities to distinguish whether the release is merely potential, likely, or actually occurring. Response of offsite authorities will be guided initially by this determination.

4.1.3.2 Release Potential and Significance

The Site Emergency class includes releases up to 1000 Ci of I-131 equivalent and/or up to 10^6 Ci of Xe-133 equivalent.

Assuming design basis meteorological conditions, the maximum Site Emergency release would produce the following doses due to direct exposure to the plume centerline:

DOWNWIND DISTANCE (m)	ASSUMED $(\chi/Q)(sec/m^3)$	WHOLE BODY DOSE FROM Xe-133 (mrem)	THYROID DOSE FROM I-131 (rem)
800 (site boundary)	5.3×10^{-4}	6000	270
10000 (edge of LPZ)	2.2×10^{-5}	250	12
16000 (10 mile zone)	1.2×10^{-5}	140	7

As can be seen, such a release occurring with unfavorable meteorological conditions would certainly require that protective measures be taken on the site and in the downwind sectors throughout the plume exposure Emergency Planning Zone. However, even in the case of a maximum release, it is likely that offsite doses would be much lower than those tabulated above due to such factors as more favorable meteorology and the effects of sheltering. The appropriate near term response for such an occurrence is to make an assessment of conditions as they actually exist and take action based on this assessment, as discussed below.

APPENDIX E

(Source: WASH-1400, Main Report)

TABLE 5-2 PWR DOMINANT ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	RELEASE CATEGORIES						Core Melt	No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AHF-γ 2×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ε 1×10^{-9} AHF-ε 1×10^{-10} ADF-ε 2×10^{-10}	AD-ε 2×10^{-6} AH-ε 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 3×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 5×10^{-9} S_1D-β 6×10^{-9}	S_1DF-ε 3×10^{-10} S_1B-ε 2×10^{-9} S_1HF-ε 4×10^{-10}	S_1D-ε 3×10^{-6} S_1H-ε 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-5} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-6}	S_2DG-β 1×10^{-12}	S_2D-β 2×10^{-8} S_2H-β 1×10^{-8}	S_2B-ε 8×10^{-9} S_2CD-ε 2×10^{-8} S_2HF-ε 1×10^{-9}	S_2D-ε 9×10^{-6} S_2H-ε 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10	3×10^{-7}	2×10^{-6}	2×10^{-5}		
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-δ 1×10^{-12}	R-α 1×10^{-9}				R-ε 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-6}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 3×10^{-10}	TMLB'-ε 6×10^{-7}	TML-ε 6×10^{-6} TKQ-ε 3×10^{-6} TKMQ-ε 1×10^{-6}		
T Probabilities	3×10^{-7}	3×10^{-6}	4×10^{-7}	7×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

KEY TO TABLE 5-2 ON FOLLOWING PAGE

KEY TO PWR ACCIDENT SEQUENCE SYMBOLS

A - Intermediate to large LOCA.

B - Failure of electric power to ESFs.

B' - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system.

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system.

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - A small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - LPIS check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ε - Containment vessel melt-throug[illegible]

KEY TO TABLE 5-2

APPENDIX F

(Source: WASH-1400, Appendix VI)

Section 2

Releases from Containment

2.1 GENERAL REMARKS

A large portion of the work of the Reactor Safety Study was expended in determining the probability and magnitude of various radioactive releases. This work is described in detail in the preceding appendices as well as Appendices VII, and VIII. In order to define the various releases that might occur, a series of release categories were identified for the postulated types of containment failure in both BWRs and PWRs. The probability of each release category and the associated magnitude of radioactive releases (as fractions of the initial core radioactivity that might leak from the containment structure) are used as input data to the consequence model.

In addition to probability and release magnitude, the parameters that characterize the various hypothetical accident sequences are time of release, duration of release, warning time for evacuation, height of release, and energy content of the released plume.

The time of release refers to the time interval between the start of the hypothetical accident and the release of radioactive material from the containment building to the atmosphere; it is used to calculate the initial decay of radioactivity. The duration of release is the total time during which radioactive material is emitted into the atmosphere; it is used to account for continuous releases by adjusting for horizontal dispersion due to wind meander. These parameters, time and duration of release, represent the temporal behavior of the release in the dispersion model. They are used to model a "puff" release from the calculations of release versus time presented in Appendix V.

The warning time for evacuation (see section 11.1.1) is the interval between awareness of impending core melt and the release of radioactive material from the containment building. Finally, the height of release and the energy content of the released plume gas affect the manner in which the plume would be dispersed in the atmosphere.

Table VI 2-1 lists the leakage parameters that characterize the PWR and BWR release categories. It should be understood that these categories are composites of numerous event tree sequences with similar characteristics, as discussed in Appendix V.

2.2 ACCIDENT DESCRIPTIONS

To help the reader understand the postulated containment releases, this section presents brief descriptions of the various physical processes that define each release category. For more detailed information on the release categories and the techniques employed to compute the radioactive releases to the atmosphere, the reader is referred to Appendices V, VII, and VIII. The dominant event tree sequences in each release category are discussed in detail in section 4.6 of Appendix V.

PWR 1

This release category can be characterized by a core meltdown followed by a steam explosion on contact of molten fuel with the residual water in the reactor vessel. The containment spray and heat removal systems are also assumed to have failed and, therefore, the containment could be at a pressure above ambient at the time of the steam explosion. It is assumed that the steam explosion would rupture the upper portion of the reactor vessel and breach the containment barrier, with the result that a substantial amount of radioactivity might be released from the containment in a puff over a period of about 10 minutes. Due to the sweeping action of gases generated during containment-vessel meltthrough, the release of radioactive materials would continue at a relatively low rate thereafter. The total release would contain

approximately 70% of the iodines and 40% of the alkali metals present in the core at the time of release.[1] Because the containment would contain hot pressurized gases at the time of failure, a relatively high release rate of sensible energy from the containment could be associated with this category. This category also includes certain potential accident sequences that would involve the occurrence of core melting and a steam explosion after containment rupture due to overpressure. In these sequences, the rate of energy release would be lower, although still relatively high.

PWR 2

This category is associated with the failure of core-cooling systems and core melting concurrent with the failure of containment spray and heat-removal systems. Failure of the containment barrier would occur through overpressure, causing a substantial fraction of the containment atmosphere to be released in a puff over a period of about 30 minutes. Due to the sweeping action of gases generated during containment vessel meltthrough, the release of radioactive material would continue at a relatively low rate thereafter. The total release would contain approximately 70% of the iodines and 50% of the alkali metals present in the core at the time of release. As in PWR release category 1, the high temperature and pressure within containment at the time of containment failure would result in a relatively high release rate of sensible energy from the containment.

PWR 3

This category involves an overpressure failure of the containment due to failure of containment heat removal. Containment failure would occur prior to the commencement of core melting. Core melting then would cause radioactive materials to be released through a ruptured containment barrier. Approximately 20% of the iodines and 20% of the alkali metals present in the core at the time of release would be released to the atmosphere. Most of the release would occur over a period of about 1.5 hours. The release of radioactive material from containment would be caused by the sweeping action of gases generated by the reaction of the molten fuel with concrete. Since these gases would be initially heated by contact with the melt, the rate of sensible energy release to the atmosphere would be moderately high.

PWR 4

This category involves failure of the core-cooling system and the containment spray injection system after a loss-of-coolant accident, together with a concurrent failure of the containment system to properly isolate. This would result in the release of 9% of the iodines and 4% of the alkali metals present in the core at the time of release. Most of the release would occur continuously over a period of 2 to 3 hours. Because the containment recirculation spray and heat-removal systems would operate to remove heat from the containment atmosphere during core melting, a relatively low rate of release of sensible energy would be associated with this category.

PWR 5

This category involves failure of the core cooling systems and is similar to PWR release category 4, except that the containment spray injection system would operate to further reduce the quantity of airborne radioactive material and to initially suppress containment temperature and pressure. The containment barrier would have a large leakage rate due to a concurrent failure of the containment system to properly isolate, and most of the radioactive material would be released continuously over a period of several hours. Approximately 3% of the iodines and 0.9% of the alkali metals present in the core would be released. Because of the operation of the containment heat-removal systems, the energy release rate would be low.

[1]The release fractions of all the chemical species are listed in Table VI 2-1. The release fractions of iodine and alkali metals are indicated here to illustrate the variations in release with release category.

PWR 6

This category involves a core meltdown due to failure in the core cooling systems. The containment sprays would not operate, but the containment barrier would retain its integrity until the molten core proceeded to melt through the concrete containment base mat. The radioactive materials would be released into the ground, with some leakage to the atmosphere occurring upward through the ground. Direct leakage to the atmosphere would also occur at a low rate prior to containment-vessel meltthrough. Most of the release would occur continuously over a period of about 10 hours. The release would include approximately 0.08% of the iodines and alkali metals present in the core at the time of release. Because leakage from containment to the atmosphere would be low and gases escaping through the ground would be cooled by contact with the soil, the energy release rate would be very low.

PWR 7

This category is similar to PWR release category 6, except that containment sprays would operate to reduce the containment temperature and pressure as well as the amount of airborne radioactivity. The release would involve 0.002% of the iodines and 0.001% of the alkali metals present in the core at the time of release. Most of the release would occur over a period of 10 hours. As in PWR release category 6, the energy release rate would be very low.

PWR 8

This category approximates a PWR design basis accident (large pipe break), except that the containment would fail to isolate properly on demand. The other engineered safeguards are assumed to function properly. The core would not melt. The release would involve approximately 0.01% of the iodines and 0.05% of the alkali metals. Most of the release would occur in the 0.5-hour period during which containment pressure would be above ambient. Because containment sprays would operate and core melting would not occur, the energy release rate would also be low.

PWR 9

This category approximates a PWR design basis accident (large pipe break), in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt. It is assumed that the minimum required engineered safeguards would function satisfactorily to remove heat from the core and containment. The release would occur over the 0.5-hour period during which the containment pressure would be above ambient. Approximately 0.00001% of the iodines and 0.00006% of the alkali metals would be released. As in PWR release category 8, the energy release rate would be very low.

BWR 1

This release category is representative of a core meltdown followed by a steam explosion in the reactor vessel. The latter would cause the release of a substantial quantity of radioactive material to the atmosphere. The total release would contain approximately 40% of the iodines and alkali metals present in the core at the time of containment failure. Most of the release would occur over a 1/2 hour period. Because of the energy generated in the steam explosion, this category would be characterized by a relatively high rate of energy release to the atmosphere. This category also includes certain sequences that involve overpressure failure of the containment prior to the occurrence of core melting and a steam explosion. In these sequences, the rate of energy release would be somewhat smaller than for those discussed above, although it would still be relatively high.

BWR 2

This release category is representative of a core meltdown resulting from a transient event in which decay-heat-removal systems are assumed to fail. Containment overpressure failure would result, and core melting would follow. Most of the release would occur over a period of about 3 hours. The containment failure would be such that radioactivity would be released directly to the atmosphere without significant retention of fission products. This category involves a relatively high rate of energy release due to the sweeping action of the gases generated by the molten mass. Approximately 90% of the iodines and 50% of the alkali metals present in the core would be released to the atmosphere.

BWR 3

This release category represents a core meltdown caused by a transient event accompanied by a failure to scram or failure to remove decay heat. Containment failure would occur either before core melt or as a result of gases generated during the interaction of the molten fuel with concrete after reactor-vessel meltthrough. Some fission-product retention would occur either in the suppression pool or the reactor building prior to release to the atmosphere. Most of the release would occur over a period of about 3 hours and would involve 10% of the iodines and 10% of the alkali metals. For those sequences in which the containment would fail due to overpressure after core melt, the rate of energy release to the atmosphere would be relatively high. For those sequences in which overpressure failure would occur before core melt, the energy release rate would be somewhat smaller, although still moderately high.

BWR 4

This release category is representative of a core meltdown with enough containment leakage to the reactor building to prevent containment failure by overpressure. The quantity of radioactivity released to the atmosphere would be significantly reduced by normal ventilation paths in the reactor building and potential mitigation by the secondary containment filter systems. Condensation in the containment and the action of the standby gas treatment system on the releases would also lead to a low rate of energy release. The radioactive material would be released from the reactor building or the stack at an elevated level. Most of the release would occur over a 2-hour period and would involve approximately 0.08% of the iodines and 0.5% of the alkali metals.

BWR 5

This category approximates a BWR design basis accident (large pipe break) in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into containment. The core would not melt, and containment leakage would be small. It is assumed that the minimum required engineered safeguards would function satisfactorily. The release would be filtered and pass through the elevated stack. It would occur over a period of about 5 hours while the containment is pressurized above ambient and would involve approximately 6×10^{-9} % of the iodines and 4×10^{-7} % of the alkali metals. Since core melt would not occur and containment heat-removal systems would operate, the release to the atmosphere would involve a negligibly small amount of thermal energy.

TABLE VI 2-1 SUMMARY OF RELEASE CATEGORIES REPRESENTING HYPOTHETICAL ACCIDENTS

Release Category	Probability (reactor-yr^{-1})	Time of Release (hr)	Duration of Release (hr)	Warning Time for Evacuation (hr)	Elevation of Release[g] (meters)	Energy Release (10^6 Btu/hr)	Xe-Kr	Fraction of Core Inventory Released[a]: Organic I[b]	I[b]	Cs-Rb	Te-Sb	Ba-Sr	Ru[c]	La[d]
PWR 1	9×10^{-7}[e]	2.5	0.5	1.0	25	20 and 520[e]	0.9	[illegible] $\times 10^{-3}$	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	[illegible] $\times 10^{-3}$	0.7	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.6	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	[illegible] $\times 10^{-4}$	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	[illegible] $\times 10^{-3}$	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A[f]	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	0.5	1.5	25	130	1.0	7×10^{-3}	.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	4×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) Background on the isotope groups and release mechanisms is presented in Appendix VII.
(b) Organic iodine is combined with elemental iodines in the calculations. Any error is negligible since its release fraction is relatively small for all large release categories.
(c) Includes Ru, Rh, Co, Mo, Tc.
(d) Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.
(e) Accident sequences within PWR 1 category have two distinct energy releases that affect consequences. PWR 1 category is subdivided into PWR 1A with a probability of 4×10^{-7} per reactor-year and 20×10^6 Btu/hr and PWR 1B with a probability of 5×10^{-7} per reactor-year and 520×10^6 Btu/hr.
(f) Not applicable.
(g) A 10 meter elevation is used in place of zero representing the mid-point of a potential containment break. Any impact on the results would be slight and conservative.