UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454
	)	50-455
(Byron Station, Units 1 and 2)	)	

TESTIMONY OF L. G. HULMAN, MILLARD L. WOHL, SCOTT NEWBERRY AND EDWARD E. BRANAGAN ON LEAGUE CONTENTIONS 8 AND 62 AND DAARE/SAFE CONTENTION 2A

Summary

The following testimony addresses League Contentions 8 and 62 and DAARE/SAFE Contention 2a which relate generally to the subject of risk and accident impacts. The principal points made in the testimony are as follows:

1. The Final Environmental Statement for Byron Station contains a reasoned consideration of environmental risks from the plant, including risks resulting from postulated accidents.

2. The overall assessment of environmental risk of accidents shows that it is roughly comparable to the risk from normal plant operation.

3. The probabilistic risk assessment methodology of WASH-1400 has been used by the Staff in the preparation of the FES and is sound for the purposes for which used.

4. The Precursor Study results do not necessarily imply that WASH-1400 estimates do not currently apply to a large class of plants and do not invalidate those estimates with respect to their use in the Byron FES.

5. Adequate protection against potential accidents has been provided at Byron Station through the Commission's licensing requirements and additional measures.

6. The possibility of cumulative doses to residents of the Illinois area from accidents at more than one nuclear power plant does not create undue risk to public health and safety.

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COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454
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NRC STAFF TESTIMONY OF L. G. HULMAN, MILLARD L. WOHL, SCOTT NEWBERRY AND EDWARD F. BRANAGAN, JR. ON LEAGUE CONTENTIONS 8 AND 62 AND DAARE/SAFE CONTENTION 2A

Q.1 Please state your names and positions with the NRC?

A.1 (Panel)

I, L. G. Hulman, am Branch Chief, Accident Evaluation Branch, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.

I, Millard L. Wohl, am a nuclear engineer in the Accident Evaluation Branch, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.

I, Scott Newberry, am a Risk Analyst in the Reliability and Risk Assessment Branch, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.

I, Edward F. Branagan, Jr., am a Health Physicist in the Radiological Assessment Branch, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.

Q.2 What is the purpose of your testimony?

A.2 (Panel)

The purpose of this testimony is to provide the Staff position in response to League Contentions 8 and 62 and DAARE/SAFE Contention 2A relating generally to Class 9 accident analysis. (Copies of those contentions are provided as Attachment A to this testimony.)

Q.3 With respect to League Contention 8, has the risk from operation of Byron Station been assessed by the Staff?

A.3 (Wohl, Hulman)

Yes, the Final Environmental Statement for Byron Station (NUREG-0848), in Section 5.9.4, contains a reasoned consideration of environmental risks from the plant, including risks resulting from postulated accidents. That section of the FES was prepared by the Accident Evaluation Branch and we adopt it as part of our testimony here. Attention is given there both to the probability of occurence of radioactive releases and to the probability of occurrence of the environmental consequences of those releases via atmospheric and groundwater pathways, as required by the Commission's Statement of Interim Policy, dated June 13, 1980, on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969." (Attachment B)

Q.4 What has the Staff concluded with respect to the overall assessment of risk of accidents at Byron Station?

A.4 (Wohl)

The overall assessment of environmental risk of accidents, assuming protective action, shows that it is roughly comparable to the risk from normal operation although accidents have a potential early fatalities and economic costs that cannot arise from normal operations. The risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a compratively sized population. FES § 5.9.4.6.

Q.5 In preparing the FES, did the Staff consider accident risks that could be caused by external natural and man-caused events such as tornadoes, fires, earthquakes and sabotage?

A.5 (Hulman, Wohl)

Yes, but only qualitatively.

Q.6 Please explain.

A.6 (Hulman, Wohl)

In Section 5.9.4.5(2) of the FES, reference is made to natural phenomena and sabotage, but no reference is made to other man-caused risks such as from explosions or airplane crashes. All natural and man-caused events, including fires, are referred to by the Staff as external events.

With respect to this case, no quantitative assessment of accident risks from external events has been made. The only cases for which external natural events have been assessed in detail are for the Zion and Indian Point reactors. For Zion, the licensee has submitted a Probabilistic Risk Assessment which indicates external events can be significant contributors to risk. For Indian Point, evaluations by the Staff also indicate significant risks due to external events. By significant, we mean that the best estimates of the additional risk from external events were shown to be as much as about a factor of 30 higher compared to the best estimate risks from internal events at Indian Point, but about 10 times the best estimate risk from internal events at Zion.

In preparing the FES for this case, the Staff made no numerical assessment of accident risks from external events at Byron, but did draw upon information obtained from the Zion and Indian Point studies for estimates in the Byron FES. That is, the Staff's best estimate of accident risks from external causes, based upon what has been learned at Zion and Indian Point, could be higher than what has been presented in the FES, but may be in the range predicted for Indian Point and Zion.

Q.7 To what extent is the generic subject of external events under consideration by the Staff?

A.7 (Hulman, Wohl)

The Staff has long recognized that the accident risks from external events can be significant. In developing criteria for the design of nuclear power plants, the Staff has developed considerable guidance for the treatment of the subject within design bases in order to reduce substantially the risk from external events. However, the current Staff assessment of the state of the art of consideration of external event PRA methodology is that it is not sufficiently mature to produce reliable absolute estimates of risk. In other words, there are many uncertainties associated with absolute estimates obtained using current methodology; however, the estimates can often yield valuable insights if used in a relative sense. The Staff is undertaking the development of a program plan for improving the capability of external events PRA methodology. This plan is expected to be completed by early summer, 1983 and is expected to be implemented over the next 2 to 3 years. The plan is directly related to Commission planning guidance presented in NUREG-0885, Issue 2 "U.S. Nuclear Regulatory Commission Policy and Planning Guidance - 1983."

Q.8 How does this compare with the guidance promulgated in the June 13, 1980 Statement of Interim Policy?

A.8 (Hulman, Wohl)

We consider this responsive in view of the state-of-the-art in quantatively assessing accident risks from external events.

Specifically, we conclude that external events can be contributors to risk, but that the state-of-the-art in quantifying the likelihood of such events, and associated uncertainty, is not well developed.

Q.9 Was the methodology of the Reactor Safety Study, WASH-1400, used in the preparation of Section 5.9.4 of the FES?

A.9 (Wohl)

Yes, the probabilistic risk assessment methodology of WASH-1400 was used by the Staff in the preparation of Section 5.9.4 of the FES. Probabilistic discussion of the environmental risks attributable to accidents at nuclear power reactor facilities is called for by the Commission's June 13, 1980 Statement of Interim Policy.

Q.10 Has the methodology of WASH-1400 been called into question since publication of that document?

A.10 (Wohl)

No. The Independent Risk Assessment Review Group stated in the Lewis Report (NUREG/CR-0400) that it was unable to determine whether the overall core-melt probability given in WASH-1400 was high or low, and concluded that the error bands were understated. It also stated that it was difficult to follow the detailed thread of calculations through the WASH-1400.

The group also determined, however, that the probabilistic methodology employed was an important advance over earlier methodologies that had been applied to reactor risk, and was sound. It stated that the fault-tree/event tree approach, coupled with an adequate data base, is the best available tool with which to quantify the accident probabilities associated with nuclear reactors. This approach was applied to a prototype pressurized water reactor (Surry) and led to the establishment of probabilities for core melt accidents and resulting release of large amounts of radioactive materials which were used as surrogates in the Byron FES.

With respect to the findings of the WASH-1400, the Commission has recently stated that it accepts the Review Group Report's conclusion that "absolute values of the risks presented by WASH-1400 should not be used uncritically either in the regulatory process or for public policy purposes and has taken and will continue to take steps to assure that any such use in the past will be corrected as appropriate." Letter, dated December 27, 1982, from Acting Chairman Ahearne to Congressman Udall (Attachment C). The letter also states that "Taking due account of the reservations expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulatory decisionmaking."

The use of probabilistic risk assessment techniques used in generating the estimates of environmental consequences of radioactive releases (FES Section 5.9.4) fulfills the requirements of the Commissions Statement of Interim Policy of June 13, 1980 with respect to NEPA accident review. The methods employed in the analyses performed for the Byron Station FES based upon WASH-1400 methodology have uncertainties associated with them. These are discussed in Section 5.9.4.5(7) of the Byron Station FES. The environmental consequences estimation in the FES takes into account significant site-specific features such as sector-dependent population, meteorology, and land fraction data surrounding the site.

Q.11 What is the Precursor Study?

A.11 (Newberry)

The "Precursor Study", or more accurately, "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report," (NUREG/CR-2497) is a report which presents the initial results of a program performed at Oak Ridge National Laboratory and administered by the Nuclear Regulatory Commission. The program uses operational data in Licensee Event Reports to evaluate potential accident precursors occurring at operating reactors. These precursors are then summarized to derive a probability for severe core damage.

Q.12 Does the Precursor Study imply that WASH-1400 estimates may not currently apply generically to a large class of plants?

A.12 (Newberry)

The Precursor Study estimated the frequency of severe core damage accidents (averaged over all domestic light water power reactors in the decade of the 1970's) to have been between 1.7 x 10^{-3} and 4.5 x 10^{-3} per reactor year. In WASH-1400, the core melt frequency for the Surry plant (taken to represent pressurized water reactors) was estimated to be 5 x 10^{-5} per year. We do not differentiate between severe core damage and core melt in this testimony since analyses have not been refined to differentiate the fraction of core melt events that may terminate at severe core damage. While this difference appears to be substantial, it does not necessarily imply that the WASH-1400 results do not currently apply to a large class of plants.

Q.13 What are the reasons for this difference in frequency estimates?

A.13 (Newberry)

As stated in the Precursor Study, 82% of the precursor estimate of severe accident frequency comes from three events: Three Mile Island accident, the Browns Ferry fire and the Rancho Seco power supply failure. These events were not explicitly addressed in WASH-1400.

While WASH-1400 did treat most elements of the TMI accident, it did not treat the possibility that the reactor operators might misdiagnose an accident in progress and turn off the safety systems that were necessary to cool the core. This event (TMI) is the most important of the three and it is the only actual instance of severe core damage.

Fires were not included among the accident initiators in WASH-1400.

The Rancho Seco event was caused by a power supply fault. A comprehensive analysis of the fault effects and systems interactions originating in power supplies for control and instrumentation was not done in WASH-1400.

Q.14 Why do these omissions in WASH-1400 not invalidate the severe core damage frequency estimates today with respect to their use in the Byron FES?

A.14 (Newberry)

Since the Three Mile Island accident, regulatory requirements have been implemented to reduce the likelihood that operators might fail to diagnose inadequate core cooling. These requirements include training procedures and new and improved instruments to aid in event diagnosis. Therefore, operator errors of this type are less likely today than they were before the TMI accident. In addition, the accident initiator (transient induced LOCA) that occurred at TMI is less likely at a Westinghouse plant like Byron because the pressurizer power operated relief valve(s) is not likely to open during feedwater transients.

Following the Browns Ferry fire, fire protection requirements were developed in a new rule, Appendix R to 10 CFR 50. Byron is being is being reviewed against the requirements of this rule. See SER § 9.5.1.

The Rancho Seco power supply failure was significant from the standpoint that the power fault caused a loss of main feedwater, affected the auxiliary feedwater controls and caused erroneous information to be sent to the operator regarding the need to manually initiate auxiliary feedwater or the emergency core cooling system. Plants studied in WASH-1400 and Byron do not appear to be as vulnerable to such faults as Rancho Seco. Additionally, Byron will have safety-related actuation for the emergency feedwater system (as well as for other engineered safety features) so that a fault in the nonsafety-related feedwater control system should not defeat the autostart of the auxiliary feedwater system. Byron will also have safety-related auxiliary feedwater flow indication and steam generator level indication in the control room, so that failures like that at Rancho Seco should not impair the operator's ability to monitor plant status.

Loss of feedwater events were the fourth dominant contributor to severe core damage in the precursor study. Auxiliary feedwater system reliability was found to be poor and no credit was given for feed-and-bleed cooling.[1/] This is a possible source of conservatism, but there were no procedures in place for feed and bleed cooling, and the staff has not yet made a complete evaluation of this mode of cooling.

WASH-1400 did not give credit for feed and bleed; however, there is some likelihood that it could be used to prevent severe core damage. Since Three Mile Island additional requirements have been implemented on all reactor plants to improve auxiliary feedwater system reliability. These requirements and the Staff evaluation can be found in Section 10.4.9 of the Byron Safety Evaluation Report.

In summary, the use of WASH-1400 core melt frequency estimates is not invalidated by the precursor study.

Q.15 Does probabilistic risk assessment provide the basis for decisions concerning safety in the licensing of Byron Station?

1/ "Feed and bleed" refers to a mode of core cooling in which all feedwater (main and auxiliary) is not available, and decay heat removal is accomplished by adding coolant inventory with the high pressure injection system and removing decay heat energy through the safety or relief valves.

A.15 (Wohl)

No. The probabilistic risk assessment approach is used by the Staff in assessing environmental impact of power reactor operation under the June 1980 Statement of Interim Policy. Licensing considerations have rested, and continue to rest, upon an applicant's compliance with the Commission's determinstic licensing criteria. Performance of a plant-specific PRA is not a licensing requirement for Byron Station.

Q.16 What is the meaning of the term "Class 9" accident or event as used in League Contention 62?

A.16 (Wohl)

The term "Class 9" event is derived from a proposed rule change published by the AEC in 1971. The proposed rule change, which has now been withdrawn by the NRC, set forth a system of classification of potential accidents for use in Staff NEPA assessments. It set forth a spectrum of accidents consisting of nine classes ranging from the most trivial to the most severe for purposes of evaluating environmental risk.

Class 9 events were characterized as ". . . involv(ing) se[illegible] postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is so small that their environmental risk is extremely low." Defense in depth, constituted by such multiple physical barriers as fuel clad, pressure vessel and containment, is an important design philosophy instituted to provide and maintain the required high degree of assurance that the environmental risk is extremely low.

Since the mitigation features of nuclear power plants have been designed to avoid breach of containment and core melt accidents, occurrences of these accidents involve sequences of failures and have been designated Class 9 events. The term "Class 9" has often been considered synonymous with accidents involving severe release of radioactive material to the environment, but such use is imprecise since the term "Class 9" is much more inclusive. Class 9 events could have radiological consequences ranging from benign to severe. For example, core damage events not involving loss of containment integrity would have fairly limited radiological consequences.

Q.17 Have there been any examples of beyond design basis, or "Class 9" accidents?

A.17 (Wohl)

In considering the facts available regarding the accident at Three Mile Island, the Staff concluded[2/] that the Three Mile Island accident ". . . involved a sequence of successive failures (i.e., small break loss-of-coolant accident and failure of emergency core cooling system) more severe than those postulated for the design basis of the plant" and thus judged that the occurrence at Three Mile Island was a Class 9 accident.

On the other hand, measurements have shown that at no time during or following the accident at Three Mile Island were the radiological consequences to the public severe.[3/] The radioactive material actually released to the environment during the accident at Three Mile Island represented a minimal risk to the public health and safety.

Q.18 What, if any, measures have been taken at Byron to protect the public health and safety against "Class 9" accidents?

2/ NRC Staff response to Board Question No. 4 regarding the Occurrence of a Class 9 Accident at Three Mile Island, in the Matter of Public Service Electric and Gas Company, August 24, 1979.

3/ Ad Hoc Interagency Assessment Group, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station. NUREG-0559, May 1979.

A.18 (Wohl)

The Byron plant and its various safety systems are analytically tested for adequacy of performance against a series of design-basis events (DBE). Each of these events imposes severe performance demands on the various safety systems which must function in response to such events to enable the plant design to satisfy regulatory requirements. Each of the events is analyzed using conservative assumptions regarding equipment availability and performance capability which are described in detail in the Staff's Standard Review Plan. Thus, the plant is tested not only against a set of challenges to its safety but under additional conservative assumptions regarding plant conditions before and during these challenges. This results in a design capability with multiple and redundant systems for coping with very severe performance demands, and provides substantial protection against unforeseen events involving multiple equipment failures and operator errors.

The Applicant is developing Emergency Response Guidelines which will consider multiple failure events. In addition to the design basis events, analyses assuming various event sequences (including multiple failures) that could occur and fall outside of the required design envelope have been utilized in the preparation of the emergency operating procedures. This approach for the operators is a result of the lessons learned from the TMI-2 accident. Its objective is to further assure that the operator is able to respond to the complete spectrum of possible events.

A margin for overall safe response to unforeseen events is provided by the flexibility incorporated in many systems and in the multiplicity of installed systems in a nuclear power plant. The plant is designed to tolerate unforeseen event sequences by appropriate use of installed dedicated emergency safety features and other equipment not considered in analysis of the DBE's. For example, alternative systems configurations may be employed or equipment may be manually actuated if automatic logic circuits do not trigger actuation.

The source terms used in offsite radiological consequence analyses for many of the DBE's for Byron are based on the conservative assumptions that 100 percent of the core noble gas inventory and 25 percent of the core iodine inventory are available for release to the containment atmosphere. During the TMI-2 accident, for example, analyses of air samples indicated that a whole body dose of about 100 mrem and thyroid dose of about 15 mrem, both very small fractions of the 10 CFR Part 100 offsite radiological consequence guidelines, would have been received by a hypothetical individual at the site boundary. There is, therefore, a spectrum of severe core damage scenarios for which it can be inferred that adequate radiological protection has been provided, as long as containment integrity is maintained.

Thus, the Byron design provides protection for a wide range of Class 9 events.

Q.19 Have steps been taken since the TMI-2 accident to reduce the likelihood of Class 9 events?

A.19 (Wohl)

Yes. Immediately following the TMI-2 accident, the Staff recognized the need for improvements. A number of bulletins and orders were issued, followed by the systematic formulation of a Task Action Plan containing extensive recommendations related to operator training and procedures, instrumentation, equipment reliability, and additional hardware.

Requirements for licensee review of operating experience, operational quality assurance, verification of management and technical capability, verification of capability for safety review and operational advice, training of operators, review of facility procedures, review of plant maintenance capability, requirement for shift turnover procedures, requirements related to shift manning, requirements for an onsite safety engineering group, systematic assessment of licensee safety programs, requirements for a shift technical advisor all contribute to a reduction in the probability of systems failure and increased capability to take corrective actions to prevent accidents from becoming more severe.

The effect of these changes is, first, to enhance the maintenance and operation of the systems involved in each step of identified event sequences, thus diminishing malfunction probabilities for the components of these systems. Secondly, they serve to upgrade significantly the ability of the operators and the operating organization to recognize and take the proper remedial action to cope with a malfunction should it occur. There is a combined effect from improvement in both these aspects on each and every step in the event sequence. Thus, the combined impact on the overall chance for successful safe termination of the initiating event is enhanced, and the likelihood of event sequences leading to core melt with concomitant containment failure resulting in 10 CFR 100 guidelines being exceeded is substantially reduced.

In sum, the deterministic licensing requirements, based upon design basis event considerations, knowledge acquired from the TMI-2 accident, mitigative engineered safety features, multiple barriers against post-accident release of radioactivity, and additional measures, such as emergency operator guidelines which allow risk-reducing human intervention in reactor accident situations provide, in the Staff's judgment, reasonable assurance that the Byron plant can be operated with no undue risk to the public health and safety.

Q.20 With respect to DAARE/SAFE Contention 2A, has the Staff considered the potential radiological impacts of accidents at the Byron Station?

A.20 (Wohl, Branagan)

Yes. The staff has considered the potential radiological impacts on the environment of certain postulated accidents at the Byron station. Calculated population exposures for these events range from a small fraction of a person-rem to about 450 person-rems for the population within 80 km (50 mi) of the Byron station. These calculations for both individual and population exposures indicate that the risk of incurring any adverse health effects as a consequence of these events is exceedingly small. FES § 5.9.4.5(1). The staff also concludes that radiation exposures from design-basis accidents are roughly comparable to the exposures to individuals and the population from normal station operations over the expected lifetime of the plant.

As stated earlier, the overall assessment of environmental risk of accidents, assuming protective action, shows that it is roughly comparable to the risk from normal operation although accidents have a potential for early fatalities and economic costs that cannot arise from normal operations. The risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a comparatively sized population. FES § 5.9.4.6.

Q.21 Have accidents at nuclear power plants in the area of northern Illinois caused a radiological dose burden to residents in that area?

A.21 (Wohl)

There has been no measured offsite radiological dose burden to Northern Illinois residents due to accidents at the nuclear power plants in Northern Illinois, either of a discrete or cumulative nature. The likelihood of a severe accident occurring at any of the nuclear power plants in Northern Illinois is sufficiently small that the addition of the Byron plants will not raise this likelihood to a significant level, even in the case of a hypothetical accident induced by an external event.

Further, the likelihood of more than one severe accident at more than one plant with resultant cumulative significant radiological consequences to residents of a specific area is obviously much smaller. Its upper bound is the product of three terms: 1) the already low probability of a severe accident at one plant over its lifetime, 2) the similarly low probability of a severe accident at another plant, and 3) the probability that in each case the radioactive plume will travel over the specific area of concern, such as the DeKalb-Sycamore or Rockford areas.

Q22. Does the possibility of cumulative doses to residents of the northern Illinois area from accidents at more than one nuclear power plant create undue risk to public health and safety?

A22. (Wohl)

No, for the reasons discussed in the foregoing answers to Questions 20 and 21 relating to DAARE/SAFE Contention 2A.

Q23. Do the Precursor Study results cause a change in the population dose estimates made by the Staff in the FES?

A23. (Wohl)

No, for the reasons discussed in the answers to questions 12-14 above.

MILLARD L. WOHL

PROFESSIONAL QUALIFICATIONS

ACCIDENT EVALUATION BRANCH

DIVISION OF SYSTEMS INTEGRATION

I am employed as a nuclear engineer in the Accident Evaluation Branch, Division of Systems Integration, U. S. Nuclear Regulatory Commission, Washington, D. C. My duties are to conduct site and accident analyses and various other safety-related studies for nuclear power and non-power reactor facilities.

I attended Case Western Reserve University (formerly Case Institute of Technology) and received a B. S. degree in Physics in 1956. I received a M. S. degree in Physics from Indiana University in 1958. I did graduate work in Nuclear Engineering at Columbia University and Case Western Reserve University from 1962 through 1964. I was a teaching assistant in Physics at Indiana University from 1956 - 1958. I have taught physics and mathematics in the evening divisions of Baldwin-Wallace College, the Ohio State University and Cuyahoga Community College from 1958 - 1973.

In 1958, I joined the NASA Lewis Research Center in Cleveland, Ohio. My initial duties involved the writing of Monte Carlo computer codes for the determination of radiation shielding requirements and propellant heating for proposed nuclear-powered rocket designs. Other assignments involved methods development and shielding and nuclear safety analyses for numerous proposed mobile nuclear vehicle applications. Numerous technical publications evolved in the course of this work. Additionally, during the period 1958 - 1973, I had substantial research contract management responsibilities.

In 1973, I joined the General Atomic Company in La Jolla, California, as a nuclear engineer. At General Atomic I performed a variety of nuclear safety-related analyses for the High-Temperature Gas-Cooled Reactor (HTGR). These included the analysis of depressurization accidents and containment integrity studies, as well as computer code upgrading and modification.

In 1975, I joined the Accident Analysis Branch in the Division of Technical Review, U. S. Nuclear Regulatory Commission. My responsibilities involved site characteristic studies and accident analyses. Presently, my responsibilities in the Accident Evaluation Branch involve evaluation of the radiological consequences of accidents postulated in connection with safety evaluations for operating reactors, and preparation of accident risk sections of Environmental Statements.

Professional Qualifications

Scott F. Newberry

Reliability and Risk Assessment Branch
Division of Safety Technology
U.S. Nuclear Regulatory Commission

My name is Scott F. Newberry. I am employed as a Risk Analyst in the Reliability and Risk Assessment Branch, Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C.

I attended the United States Naval Academy, Annapolis, Maryland, and received a B.S. degree in 1970. I received a Masters degree in Mechanical Engineering from the Catholic University of America in Washington, D.C. in 1980.

From 1970 to 1971 I attended the Navy Nuclear Power Training Program which consisted of training at the Nuclear Power Training School, Bainbridge, Maryland, and the S3G submarine reactor prototype in West Milton, New York.

From 1972 until 1974 I worked as Engineering Officer of the Watch aboard the USS Daniel Boone SSBN 629 (Blue), a nuclear fleet ballistic missile submarine. My primary assignment was to serve as the ship's Main Propulsion Assistant and Radiological Controls Officer during this period. I was responsible for the ship's reactor coolant system and steam system propulsion machinery and the control of all radioactive material on board.

In 1974 I qualified as Nuclear Engineering Officer in the Naval Reactors Program.

From 1974 to 1976 I served as Weapons Officer, USS Nathan Hale SSBN 623 (GOLD). During this period I was involved in the ship's precritical and power range testing program during the nuclear refueling overhaul as a Command Duty Officer.

In December 1976, I started working for the Reactor Systems Branch, Division of Systems Safety, U.S. Nuclear Regulatory Commission, as a reactor engineer. I have reviewed construction and operating license safety analyses in the reactor systems areas for compliance with NRC regulations. The reactor systems areas included:

1. Structures, systems, and components to be protected from internally generated missiles inside containments.

2. Overpressure protection systems and the steam generator safety valves.

3. Reactor coolant pressure boundary leakage detection systems.

4. Residual heat removal systems.

5. Reactivity control systems.

6. Emergency core cooling systems.

7. Configuration and process design parameters of the reactor coolant pumps, steam generators (PWR); reactor coolant piping.

In 1979 I joined the Three Mile Island Program Office. My responsibilities included:

1. Analysis of plant conditions and proposed changes in system design or operation mode.

2. Review of proposed operating plans and system modifications, and procedures to accomplish major operations such as long-term cooling.

3. Preparation of Technical Specifications appropriate to the plant conditions and activities.

In October 1981 I joined the Reliability and Risk Assessment Branch. My responsibilities include performance of reliability and risk assessment reviews pertaining to the functional capability of nuclear power plant safety systems, equipment and procedures needed for safe plant operation and shutdown.

EDWARD F. BRANAGAN, JR.
OFFICE OF NUCLEAR REACTOR REGULATION

PROFESSIONAL QUALIFICATIONS

From April 1979 to the present, I have been employed in the Radiological Assessment Branch in the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC). As a Health Physicist with the Radiological Assessment Branch, I am responsible for evaluating the environmental radiological impacts resulting from the operation of nuclear power reactors. In particular, I am responsible for evaluating radioecological models and health effect models for use in reactor licensing.

In addition to my duties involving the evaluation of radiological impacts from nuclear reactors, my duties in the Radiological Assessment Branch have included the following: (1) I managed and was the principal author of a report entitled "Staff Review of 'Radioecological Assessment of the Wyhl Nuclear Power Plant'" (NUREG-0668); (2) I served as a technical contact on an NRC contract with Argonne National Laboratory involving development of a computer program to calculate health effects from radiation; (3) I served as the project manager on an NRC contract with Idaho National Engineering Laboratory involving estimated and measured concentrations of radionuclides in the environment; (4) I served as the project manager on an NRC contract with Lawrence Livermore Laboratory concerning a literature review of values for parameters in terrestrial radionuclide transport models; and (5) I served as the project manager on an NRC contract with Oak Ridge National Laboratory concerning a statistical analysis of dose estimates via food pathways.

From 1976 to April 1979, I was employed by the NRC's Office of Nuclear Materials Safety and Safeguards, where I was involved in project management and technical work. I served as the project manager for the NRC in connection with the NRC's estimation of radiation doses from radon-222 and radium-226 releases from uranium mills, in coordination with Oak Ridge National Laboratory which served as the NRC contractor. As part of my work on NRC's Generic Environmental Impact Statement on Uranium Milling (GEIS), I estimated health effects from uranium mill tailings. Upon publication of the GEIS, I presented a paper entitled "Health Effects of Uranium Mining and Milling for Commercial Nuclear Power" at a Conference on Health Implications of New Energy Technologies.

I received a B.A. in Physics from Catholic University in 1969, a M.A. in Science Teaching from Catholic University in 1970, and a Ph.D. in Radiation Biophysics from Kansas University in 1976. While completing my course work for my Ph.D., I was an instructor of Radiation Technology at Haskell Junior College in Lawrence, Kansas. My doctoral research work was in the area of DNA base damage, and was supported by a U.S. Public Health Service traineeship; my doctoral dissertation was entitled "Nuclear Magnetic Resonance Spectroscopy of Gamma-Irradiated DNA Bases."

I am a member of the Health Physics Society.

Attachment A

LEAGUE CONTENTION 8

Neither C.E. nor the Staff has presented a meaningful assessment of the risks associated with the operation of the proposed Byron nuclear facility, contrary to the requirements of 10 C.F.R. § 51.20(a) and § 51.20(d). Studies carried out by the NRC have identified accident mechanisms, considered credible, which would lead to uncontrolable accidents and release to the environment of appreciable fractions of a reactor's inventory of radioactive materials. Traditionally, these accident potentials have been downplayed or ignored on the basis of the Rasmussen Report. However, the Lewis Committee has now called into serious question the entire methodology, as well as the findings and conclusions, of the Rasmussen Report, which led the NRC to withdraw official reliance on the Rasmussen Report, yet the Staff still regulates upon the validity of the basic conclusions therein. In addition, NRC Staff studies, which are not common public knowledge, have cast doubt upon numberous of the specific conclusions of the Rasmussen Report. For example, in one secret NRC study, estimates of the "killing distance" were made, referring to the range over which lethal injuries would be received under varying weather conditions from the release of radioactive material in a nuclear power plant accident. Depending upon prevailing weather conditions, this "killing distance" was estimated to be up to several dozen miles from the accident-damaged reactor. Unpublished document from Brookhaven National Laboratory, USAEC. In addition, the Liquid Pathways Study, NUREG-0440 (February, 1978), highlights the incomplete safety assessment currently performed by the NRC, particularly with respect to incomplete review of all credible accident sequences. A General Accounting Office report pertaining to that study criticizes the NRC's failure to consider core-melt accidents in assessments of relative differences in Class 9 risks. The March 7, 1978 letter from the NRC's Mr. Case to the Commissioners (Secy-78-137) also urges the inclusion of core-melt considerations in site comparisons in the case of sites involving high population density, such as Byron and the surrounding area in which live now (or at time of proposed operation) upwards of 500,000 persons. Moreover, neither C.E. nor the NRC Staff has presented an accurate assessment of the risks posed by operation of Byron, contrary to the requirements of 10 C.F.R. § 51.20(a) and § 51.20(d). The decision to issue the Byron construction permit did not, and the presently filed analysis of C.E. and the Staff do not, consider the consequences of so-called Class 9 accidents, particularly core meltdown with breach of containment. These accidents were deemed to have a low probability of occurrence. The Reactor Safety Study, WASH-1400, was an attempt to demonstrate that the actual risk from Class 9 accidents is very low. However, the Commission has stated that it "does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident." (NRC Statement of Risk Assessment and the Reactor Safety Study Report (WASH-1400) in Light of the Risk Assessment Review Group Report, January 18, 1970). The withdrawal of NRC's endorsement of the Reactor Sfaety Study and its findings leaves no technical basis for concluding that the actual risk is low enough to justify operation of Byron.

LEAGUE CONTENTION 62

The design of Byron does not provide protection against so-called "Class 9" accidents. There is no basis for concluding that such accidents are not credible. Indeed, the staff has conceded that the accident at TMI falls within that classification. Therefore, there is no reasonable assurance that Byron can be operated without endangering the health and safety of the public. See also Contention 8, supra."

DARRE/SAFE CONTENTION 2A

"Due to the concentration of nuclear power plants already in Northern Illinois; the Applicant's record of incidents and violations in existing plants which have emerged since the granting of a Construction License for Byron; and the credibility which must now be given to large scale accident scenarios since TMI, Intervenors contend that the addition of Byron Station operations places an undue and unfair burden of risk from exposure to radioactive materials from accidental releases on DeKalb-Sycamore and Rockford area residents. With the addition of two more nuclear power units in operation at Byron, the potential for cumulative dose effects from discrete accident events at plants in Northern Illinois under unfavorable meteorological conditions poses an unreasonable level of risk to the health and safety of DeKalb-Sycamore and Rockford area residents."

precise condition of the reactor core is not known at this time and cannot be known until the containment has been entered and the reactor vessel has been opened. For this reason, it is unrealistic to expect that the programmatic impact statement will serve as a blueprint, detailing each and every step to be taken over the coming months and years with their likely impacts. That the planned programmatic statement inevitably will have gaps and will not be a complete guide for all future actions does not invalidate its usefulness as a planning tool. As more information becomes available it will be incorporated into the decision-making process, and where appropriate supplements to the programmatic environmental impact statement will be issued. As the decontamination of TMI-2 progresses the Commission will make any new information available to the public and to the extent necessary will also prepare separate environmental statements or assessments for individual portions of the overall clean-up effort.

The development of a programmatic impact statement will not preclude prompt Commission action when needed. The Commission does recognize, however, that as with its Epicor-II approval action, any action taken in the absence of an overall impact statement will lead to arguments that there has been an inadequate environmental analysis, even where the Commission's action itself is supported by an environmental assessment. As in settling upon the scope of the programmatic impact statement, CEQ can lend assistance here. For example should the Commission before completing its programmatic statement decide that it is in the best interest of the public health and safety to decontaminate the high level waste water now in the containment building, or to purge that building of its radioactive gases, the Commission will consider CEQ's advice as to the Commission's NEPA responsibilities. Moreover, as stated in the Commission's May 25 statement, any action of this kind will not be taken until it has undergone an environmental review, and furthermore with opportunity for public comment provided.

However, consistent with our May 25 Statement, we recognize that there may be emergency situations, not now foreseen, which should they occur would require rapid action. To the extent practicable the Commission will consult with CEQ in these situations as well.

With the help of the public's comments on our proposals we intend to assure, pursuant to NEPA and the Atomic Energy Act, that the clean-up of TMI-2 is done consistently with the public health and safety, and with awareness of the choices ahead. We are directing our staff to include in the programmatic environmental impact statement on the decontamination and disposal of TMI-2 wastes an overall description of the planned activities and a schedule for their completion along with a discussion of alternatives considered and the rationale for choices made. We are also directing our staff to keep us advised of their progress in these matters.

45 FR 2893
Published 1/15/80

EPA Policy Statement; Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents

Purpose

This is a statement of policy with regard to an Environmental Protection Agency (EPA) and Nuclear Regulatory Commission (NRC) task force report on guidance for use in State and local radiological emergency response plans at nuclear power plants.

Background

The NRC received a request from the Conference of Radiation Control Program Directors, an organization of State officials, to "make a determination of the most severe accident basis for which radiological emergency response plans should be developed by offsite agencies." In response, an EPA and NRC task force was established which prepared a report entitled "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396, EPA 520/1-78-016, dated December 1978. Single copies of the report can be obtained by writing to the Director, Division of Technical Information and Document Control, Nuclear Regulatory Commission, Washington, D.C. 20555.

Planning Basis

The major recommendation of the report is that Emergency Planning Zones (EPZ's) should be established around light water nuclear power plants. The EPZ for airborne exposure has a radius of about 10 miles; the EPZ for contaminated food has a radius of about 50 miles. Predetermined protective action plans are needed for the EPZ's. The exact size and shape of each EPZ will be decided by emergency planning officials after they consider the specific conditions at each site.

The report indicates that officials may have from one-half hour to several hours warning in which to implement protective actions before a release of radioactivity to the atmosphere.

The chemical and physical characteristics of those radionuclides which contribute most significantly to human exposure are presented.

EPA Policy

EPA concurs in and endorses for use the guidance contained in the task force report. It will be EPA's policy to incorporate its recommendations into all EPA emergency response guidance to State and local officials.

45 FR 40101
Published 6/13/80
Comment period expires 9/11/80

10 CFR Parts 50 and 51

Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969

AGENCY: U.S Nuclear Regulatory Commission

ACTION: Statement of Interim Policy.

SUMMARY: The Nuclear Regulatory Commission (NRC) is revising its policy for considering the more severe kinds of very low probability accidents that are physically possible in environmental impact assessments required by the National Environmental Policy Act (NEPA). Such accidents are commonly referred to as Class 9 accidents, following an accident classification scheme proposed by the Atomic Energy Commission (predecessor to NRC) in 1971 for purposes of implementing NEPA.[1] The March 28, 1979 accident at Unit 2 of the Three Mile Island nuclear plant has emphasized the need for changes in NRC policies regarding the considerations to be given to serious accidents from an environmental as well as a safety point of view.

This statement of interim policy announces the withdrawal of the proposed Annex to Appendix D of 10 CFR Part 50 and the suspension of the rulemaking proceeding that began with the publication of that proposed Annex on December 1, 1971. It is the Commission's position that its Environmental Impact Statements shall include considerations of the site-specific environmental impacts attributable to accident sequences that

[1] Proposed as an Annex to 10 CFR Part 50, Appendix D. 36 FR 22851. The Commission's NEPA-implementing regulations were subsequently (July 18, 1974) revised and recast as 10 CFR Part 51 but at that time the Commission noted that "The Proposed Annex is still under consideration * * *" 39 FR 26279.

Attachment B

Public Document Room.

Persons with questions may call Dr. Harry J. Watters in the Office of Management and Program Analysis, telephone 301-492-7721.

Written comments or questions should be addressed to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Comments must be received by December 10, 1979.

44 FR 61123
Published 10/23/79

Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents

AGENCY: Nuclear Regulatory Commission.

ACTION: NRC Policy Statement.

Purpose

This is a statement of policy with regard to an Environmental Protection Agency (EPA) and Nuclear Regulatory Commission (NRC) task force report on guidance for use in state and local radiological emergency response plans at nuclear power plants.

Background

The NRC received a request from the Conference of Radiation Control Program Directors, an organization of State officials, to "make a determination of the most servere accident basis for which radiological emergency response plans should be developed by offsite agencies." In response, an EPA and NRC task force was established which prepared a report entitled "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants."

NUREG-0396, EPA 520/1-78-016, dated December 1978. Single copies of the report can be obtained by writing to the Director, Division of Technical Information and Document Control, Nuclear Regulatory Commission, Washington, D.C. 20555. The task force report was published for public comment in the Federal Register on December 15, 1978 and the comment period was extended to May 15, 1979 to allow additional comments resulting from the accident at Three Mile Island. A synopsis of the comments received and the task force consideration of these comments is available from the Assistant Director for Emergency Preparedness, Office of State Programs, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

Planning Basis

The major recommendation of the report is that two Emergency Planning Zones (EPZs) should be established around light water nuclear power plants. The EPZ for airborne exposure has a radius of about 10 miles; the EPZ for contaminated food has a radius of about 50 miles. Predetermined protective action plans are needed for the EPZs. The exact size and shape of each EPZ will be decided by emergency planning officials after they consider the specific conditions at each site. These distances are considered large enough to provide a response base which would support activity outside the planning zone should this ever be needed.

The report also provides planning basis guidance in the form of a range of time values in which emergency response officials should be prepared to implement protective action. The report indicates that, depending on such factors as the specific sequence of events during an accident which results in the release of radioactivity to the atmoshpere and the prevailing meteorological conditions, protective action may be required from perhaps one-half hour to one day after the intiation of the accident. Development and periodic testing of procedures for rapid notification of emergency response officials is encouraged, since the time available for action is strongly affected by the time consumed in notification.

The chemical and physical characteristics of those radionuclides which contribute most significantly to human exposure are presented.

NRC Policy

NRC concurs in and endorses for use the guidance contained in the task force report. In endorsing this guidance, the Commission recognizes that it is appropriate and prudent for emergency planning guidance to take into consideration the principal characteristics (such as nuclides released and distances likely to be involved) of a spectrum of design basis and core melt accidents. While the Commission recognizes that the guidance may have significant response in parts for many local jurisdictions, it believes that implementation of the guidance is nevetheless needed to improve emergency response planning and preparedness around nuclear power reactors.

The Commission is direction its staff to incorporate the planning basis guidance into existing documents used in the evaluation of state an local emergency response plans to the extent practicable. The NRC has recently published and Advance Notice of Proposed Rulemaking concerning additional regulations on emergency plans, 44 FR 41484, Tuesday, July 17, 1979. Additional guidance will be provided following this rulemaking. This additional guidance can be expected to consider how local conditions such as demography, land use, and meteorology can influence the size and shape of the EPZs and to address other issues, such as evacuation planning.

Specific implementation dates for full implementation of the task force recommendations and any others that are developed will be established as part of the ongoing rulemaking effort. The Commission also expects the staff to assist state and local governments in improving their emergency response capabilities at existing sites in the immediate future.

44 FR 67738
Published 11/27/79

Statement of Policy and Notice of Intent To Prepare a Programmatic Environmental Impact Statement

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Statement of Policy.

SUMMARY: The Nuclear Regulatory Commission has decided to prepare a programmatic environmental impact statement on the decontamination and disposal of radioactive wastes resulting from the March 28, 1979 accident at Three Mile Island Unit 2. For some time the Commission's staff has been moving in this direction. In the Commission's judgment an overall study of the decontamination and disposal process will assist the Commission in carrying out its regulatory responsibilities under the Atomic Energy Act to protect the public health and safety as decontamination progresses. It will also be in keeping with the purposes of the National Environmental Policy Act to engage the public in the Commission's decision-making process, and to focus on environmental issues and alternatives before commitments to specific clean-up choices are made. Additionally, in light of the extraordinary nature of this action and the expressed interest of the President's Council on Environmental Quality in the TMI-2 clean-up, the Commission intends to co-ordinate its action with CEQ. In particular, before determining the scope of the programmatic environmental impact statement the Commission will consult with CEQ.

The Commission recognizes that there are still areas of uncertainty regarding the clean-up operation. For example, the

lead to releases of radiation and/or radioactive materials, including sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core. In this regard, attention shall be given both to the probability of occurrence of such releases and to the environmental consequences of such releases. This statement of interim policy is taken in coordination with other ongoing safety-related activities that are directly related to accident considerations in the areas of plant design, operational safety, siting policy, and emergency planning. The Commission intends to continue the rulemaking on this matter when new siting requirements and other safety related requirements incorporating accident considerations are in place.

DATES: This statement of interim policy is effective June 13, 1980 Comment period expires September 11, 1980.

ADDRESSES: The Commission intends the interim policy guidance contained herein to be immediately effective. However, all interested persons who desire to submit written comments or suggestions for consideration in connection with this statement should send them to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

FOR FURTHER INFORMATION CONTACT: R. Wayne Houston, Chief, Accident Evaluation Branch Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Telephone: (301) 492-7323.

SUPPLEMENTARY INFORMATION:

Accident Considerations in Past NEPA Reviews

The proposed Annex to Appendix D of 10 CFR Part 50 (hereafter the "Annex") was published for comment on December 1, 1971 by the (former) Atomic Energy Commission. It proposed to specify a set of standardized accident assumptions to be used in Environmental Reports submitted by applicants for construction permits or operating licenses for nuclear power reactors. It also included a system for classifying accidents according to a graded scale of severity and probability of occurrence. Nine classes of accidents were defined, ranging from trivial to very serious. It directed that "for each class, except classes 1 and 9, the environmental consequences shall be evaluated as indicated." Class 1 events were not to be considered because of their trivial consequences, whereas in regard to Class 9 events, the Annex stated as follows:

> The occurrences in Class 9 involve sequences of postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is so small that their environmental risk is extremely low. Defense in depth (multiple physical barriers), quality assurance for design, manufacture, and operation, continued surveillance and testing, and conservative design are all applied to provide and maintain the required high degree of assurance that potential accidents in this class are, and will remain, sufficiently remote in probability that the environmental risk is extremely low. For these reasons, it is not necessary to discuss such events in applicants' Environmental Reports.

A footnote to the Annex stated:

> Although this annex refers to applicant's Environmental Reports, the current assumptions and other provisions thereof are applicable, except as the content may otherwise require, to AEC draft and final Detailed Statements.

During the public comment period that followed publication of the Annex a number of criticisms of the Annex were received. Principal among these were the following:

(1) The philosophy of prescribing assumptions does not lead to objective analysis.

(2) It failed to treat the probabilities of accidents in any but the most general way.

(3) No supporting analysis was given to show that Class 9 accidents are sufficiently low in probability that their consequences in terms of environmental risks need not be discussed.

(4) No guidance was given as to how accident and normal releases of radioactive effluents during plant operation should be factored into the cost-benefit analysis.

(5) The accident assumptions are not generally applicable to gas cooled or liquid metal cooled reactors.

(6) Safety and environmental risks are not essentially different considerations.

Neither the Atomic Energy Commission nor the NRC took any further action on this rulemaking except in 1974 when 10 CFR Part 51 was promulgated. Over the intervening years the accident considerations discussed in Environmental Impact Statements for proposed nuclear power plants reflected the guidance of the Annex with few exceptions. Typically, the discussions of accident consequences through Class 8 (design basis accidents) for each case have reflected specific site characteristics associated with meteorology (the dispersion of releases of radioactive material into the atmosphere), the actual population within a 50-mile radius of the plant and some differences between boiling water reactors (BWR) and pressurized water reactors (PWR). Beyond these few specifics, the discussions have reiterated the guidance of the Annex and have relied upon the Annex's conclusion that the probability of occurrence of a Class 9 event is too low to warrant consideration, a conclusion based upon generally stated safety considerations.

With the publication of the Reactor Safety Study (WASH-1400), in draft form in August 1974 and final form in October 1975, the accident discussions in Environmental Impact Statements began to refer to this first detailed study of the risks associated with nuclear power plant accidents, particularly events which can lead to the melting of the fuel inside a reactor.[2] The references to this study were in keeping with the intent and spirit of NEPA "to disclose" relevant information, but it is obvious that WASH-1400 did not form the basis for the conclusion expressed in the Annex in 1971 that the probability of occurrence of Class 9 events was too low to warrant their (site-specific) consideration under NEPA.

The Commission's staff has, however, identified in certain cases unique circumstances which it felt warranted more extensive and detailed consideration of Class 9 events. One of these was the proposed Clinch River Breeder Reactor Plant (CRBRP), a liquid metal cooled fast breeder reactor very different from the more conventional light water reactor plants for which the safety experience base is much broader. In the Final Environmental Statement for the CRBRP,[3] the staff included a discussion of the consideration it had given to Class 9 events.

In the early site review for the Perryman site, the staff performed an informal assessment of the relative differences in Class 9 accident consequences among the alternative sites. (SECY-78-137)

In the case of the application by Offshore Power Systems to manufacture floating nuclear power plants, the staff judged that the environmental risks of some Class 9 events warranted special consideration. The special circumstances were the potentially serious consequences associated with water (liquid) pathways leading to radiological exposures if a molten reactor core were to fall into the water

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[2] It is of interest that the Reactor Safety Study never refers to nor uses the term "Class 9 accident" although this term is commonly used as loosely equivalent to a core melt accident.

[3] NUREG-0139, February 1977.

body on which the plant floats. Here the staff emphasized its focus on risk to the environment but did not find that the probability of a core melt event occurring in the first place was essentially any different than for land-based plant. In its Memorandum and Order In the Matter of Offshore Power Systems,[4] the Commission consurred in the staff's judgment. Thus, the Reactor Safety Study and NRC experience with these cases has served to refocus attention on the need to reemphasize that environmental risk entails both probabilities and consequences, a point that was made in the publication of the Annex, but was not given adequate emphasis.

In July 1977 the NRC commissioned a Risk Assessment Review Group "to clarify the achievements and limitations of the Reactor Safety Study." One of the conclusions of this study, published in September 1978, as NUREG/CR-0400, "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission," was that "The Review Group was unable to determine whether the absolute probabilities of accident sequences in WASH-1400 are high or low, but believes that the error bounds on those estimates are in general, greatly understated." This and other findings of the Review Group have also subsequently been referred to in Environmental Impact Statements, along with a reference to the Commission's policy statement on the Reactor Safety Study in light of the Risk Assessment Review Group Report, published on January 18, 1979. The Commission's statement accepted the findings of the Review Group, both as to the Reactor Safety Study's achievements and as to its limitations.

A few Draft Environmental Statements have been published subsequent to the Three Mile Island accident. These were for conventional land-based light water reactor plants and continued to reflect the past practice with respect to accidents at such plants, but noted that the experience gained from the Three Mile Island accident was not factored into the discussion.

Our experience with past NEPA reviews of accidents and the TMI accident clearly leads us to believe that a change is needed.

Accordingly, the proposed Annex to Appendix D of 10 CFR Part 50, published on December 1, 1971, is hereby withdrawn and shall not hereafter be used by applicants nor by the staff. The reasons for the withdrawal are as follows:

1. The Annex proscribes consideration of the kinds of accidents (Class 9) that, according to the Reactor Safety Study, dominate the accident risk.

2. The definition of Class 9 accidents in the Annex is not sufficiently precise to warrant its further use in Commission policy, rules, and regulations, nor as a decision criterion in agency practice.

3. The Annex's prescription of assumptions to be used in the analysis of the environmental consequences of accidents does not contribute to objective consideration.

4. The Annex does not give adequate consideration to the detailed treatment of measures taken to prevent and to mitigate the consequences of accidents in the safety review of each application.

The classification of accidents proposed in that Annex shall no longer be used. In its place the following interim guidance is given for the treatment of accident risk considerations in NEPA reviews.

Accident Considerations in Future NEPA Reviews

It is the position of the Commission that its Environmental Impact Statements, pursuant to Section 102(c)(i) of the National Environmental Policy Act of 1969, shall include a reasoned consideration of the environmental risks (impacts) attributable to accidents at the particular facility or facilities within the scope of each such statement. In the analysis and discussion of such risks, approximately equal attention shall be given to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences of those releases. Releases refer to radiation and/or radioactive materials entering environmental exposure pathways, including air, water, and ground water.

Events or accident sequences that lead to releases shall include but not be limited to those that can reasonably be expected to occur. In-plant accident sequences that can lead to a spectrum of releases shall be discussed and shall include sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core. The extent to which events arising from causes external to the plant which are considered possible contributors to the risk associated with the particular plant shall also be discussed. Detailed quantitative considerations that form the basis of probabilistic estimates of releases need not be incorporated in the Environmental Impact Statements but shall be referenced therein. Such references shall include, as applicable, reports on safety evaluations.

The environmental consequences of releases whose probability of occurence has been estimated shall also be discussed in probabilistic terms. Such consequences shall be characterized in terms of potential radiological exposures to individuals, to population groups and, where applicable, to biota. Health and safety risks that may be associated with exposures to people shall be discussed in a manner that fairly reflects the current state of knowledge regarding such risks. Socioeconomic impacts that might be associated with emergency measures during or following an accident should also be discussed. The environmental risk of accidents should also be compared to and contrasted with radiological risks associated with normal and anticipated operational releases.

In promulgating this interim guidance, the Commission is aware that there are and will likely remain for some time to come many uncertainties in the application of risk assessment methods, and it expects that its Environmental Impact Statements will identify major uncertainties in its probabilistic estimates. On the other hand the Commission believes that the state of the art is sufficiently advanced that a beginning should now be made in the use of these methodologies in the regulatory process, and that such use will represent a constructive and rational forward step in the discharge of its reponsibilities.

It is the intent of the Commission in issuing this Statement of Interim Policy that the staff will initiate treatments of accident considerations, in accordance with the foregoing guidance, in its ongoing NEPA reviews, i.e., for any proceeding at a licensing stage where a Final Environmental Impact Statement has not yet been issued. These new treatments, which will take into account significant site- and plant-specific features, will result in more detailed discussions of accident risks than in previous environmental statements, particularly for those related to conventional light water plants at land-based sites. It is expected that these revised treatments will lead to conclusions regarding the environmental risks of accidents similar to those that would be reached by a continuation of current practices, particularly for cases involving special circumstances where Class 9 risks have been considered by the staff, as described above. Thus, this change in policy is not to be construed as any lack of confidence in conclusions regarding the environmental risks of accidents expressed in any previously

[4] Docket No. STN 50-437, September 14, 1979.

issued Statements, nor, absent a showing of similar special circumstances, as a basis for opening, reopening, or expanding any previous or ongoing proceeding.[5]

However, it is also the intent of the Commission that the staff take steps to identify additional cases that might warrant early consideration of either additional features or other actions which would prevent or mitigate the consequences of serious accidents. Cases for such consideration are those for which a Final Environmental Statement has already been issued at the Construction Permit stage but for which the Operating License review stage has not yet been reached. In carrying out this directive, the staff should consider relevant site features, including population density, associated with accident risk in comparison to such features at presently operating plants. Staff should also consider the likelihood that substantive changes in plant design features which may compensate further for adverse site features may be more easily incorporated in plants when construction has not yet progressed very far.

Environmental Reports submitted by applicants for construction permits and for operating licenses on or after July 1, 1980 should include a discussion of the environmental risks associated with accidents that follows the guidance given herein.

Related Policy Matters Under Consideration

In addition to its responsibilities under NEPA, the NRC also bears responsibility under the Atomic Energy Act for the protection of the public health and safety from the hazards associated with the use of nuclear energy. Pursuant to this responsibility the Commission notes that there are currently a number of ongoing activities being considered by the Commission and its staff which intimately relate to the "Class 9 accident" question and which are either the subject of current rulemaking or are candidate subjects for rulemaking.

On December 19, 1979 the Commission issued for public comment[6] a proposed rule which would significantly revise its requirements in 10 CFR Part 50 for emergency planning for nuclear power plants. One of the considerations in this rulemaking was the potential consequences of Class 9 accidents in a generic sense.[7]

In August 1979, pursuant to the Commission's request, a Siting Policy Task Force made recommendations with respect to possible changes in NRC reactor siting policy and criteria,[8] currently set forth in 10 CFR Part 100. As stated therein, its recommendations were made to accomplish (among others) the following goal:

> To take into consideration in siting the risk associated with accidents beyond the design basis (Class 9) by establishing population density and distribution criteria.

This matter is currently before the Commission.

This and other recommendations that have been made as a result of the investigations into the Three Mile Island accident are currently being brought together by the Commission's staff in the form of proposed Action Plans.[9] Among other matters, these incorporate recommendations for rulemaking related to degraded core cooling and core melt accidents. The Commission expects to issue decisions on these Action Plans in the near future. It is the Commission's policy and intent to devote NRC's major resources to matters which the Commission believes will make existing and future nuclear power plants safer, and to prevent a recurrence of the kind of accident that occurred at Three Mile Island. In the interim, however, and pending completion of rulemaking activities in the areas of emergency planning, siting criteria, and design and operational safety, all of which involve considerations of serious accident potential, the Commission finds it essential to improve its procedures for describing and disclosing to the public the basis for arriving at conclusions regarding the environmental risks due to accidents at nuclear power plants. On completion of the rulemaking activities in these areas, and based also upon the experience gained with this statement of interim policy and guidance, the Commission intends to pursue possible changes or additions to 10 CFR Part 51 to codify its position on the role of accident risk under NEPA.

[5] Commissioners Gilinsky and Bradford disagree with the inclusion of the preceding two sentences. They feel that they are absolutely inconsistent with an even-handed reappraisal of the former, erroneous position on Class 9 accidents.

[6] 44 FR 75167.

[7] Cf. NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," November 1978.

[8] NUREG-0625, "Report of the Siting Policy Task Force," August 1979.

[9] Draft NUREG-0660, "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the TMI-2 Accident," December 10, 1979.

45 FR 41738
Published 6/20/80

Further Commission Guidance for Power Reactor Operating Licenses; Statement of Policy[1]

I. Background

After the March 1979 accident at Three Mile Island, Unit 2, the Commission directed its technical review resources to assuring the safety of operating power reactors rather than to the issuance of new licenses. Furthermore, the Commission decided that power reactor licensing should not continue until the assessment of the TMI accident had been substantially completed and comprehensive improvements in both the operation and regulation of nuclear power plants had been set in motion.

At a meeting on May 30, 1979, the Nuclear Regulatory Commission decided to issue policy guidance addressing general principles for reaching licensing decisions and to provide specific guidance for near-term operating license cases.[2] In November 1979, the Nuclear Regulatory Commission issued the policy guidance in the form of an amendment to 10 CFR Part 2 of its regulations,[3] describing the approach to be taken by the Commission regarding licensing of power reactors. In particular, the Commission noted that it would "be providing case-by-case guidance on changes in regulatory policies." The Commission has now acted on three operating licenses, has given extensive consideration to issues arising as a result of the Three Mile Island accident, and is able to provide general guidance.

Following the accident at Three Mile Island 2, the President established a Commission to make recommendations regarding changes necessary to improve nuclear safety. In May 1979, the Nuclear Regulatory Commission established a Lessons Learned Task Force,[4] to determine what actions were required for new operating licenses and chartered a Special Inquiry Group to examine all facets of the accident and its causes. These groups have published

[1] All footnotes for this statement of policy appear at end of text.

[2] "Staff Requirements—Discussion of Options Regarding Deferral of Licenses," memorandum from Samuel J. Chilk, Secretary to Lee V. Gossick, Executive Director for Operations, May 31, 1979.

[3] "Suspension of 10 CFR 2.764 and Statement of Policy on Conduct of Adjudicatory Proceedings," 44 FR 65050 (November 9, 1979).

[4] "Lessons Learned from TMI-2 Accident," Roger Mattson to NRR staff, May 31, 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

December 27, 1982

The Honorable Morris K. Udall
Chairman, Committee on Interior
and Insular Affairs
U.S. House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

Your letter to me dated October 1, 1982 cited Mr. Bender's recent comments concerning the use of probabilistic risk assessment (PRA) and asked for answers to three questions. Before responding to your questions, I would like to comment on the statements made in your letter.

I would first like to note that the section you have quoted from the January 18, 1979, Commission's statement on the use of risk assessment is substantially less than the Commission's response to the Lewis Committee Review. A few additional quotes will serve to amplify this. The Commission commented on the findings of the Lewis Report and said:

> "The Commission accepts these findings and takes the following actions:
>
> .....
>
> Accident Probabilities: The Commission accepts the Review Group Report's conclusion that absolute values of the risks presented by WASH 1400 should not be used uncritically either in the regulatory process or for public policy purposes and has taken and will continue to take steps to assure that any such use in the past will be corrected as appropriate. In particular, in light of the Review Group conclusions on accident probabilities, the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident.
>
> .....
>
> With respect to the component parts of the Study, the Commission expects the staff to make use of them as appropriate, that is, where the data base is adequate and analytical techniques permit. Taking due account of the reservations expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulatory decisionmaking."

The Commission also approved a directive which was sent from the Secretary of the Commission to the Executive Director for Operations on January 18, 1979. Some sections are particularly germane to answering your questions:

Attachment C

"Quantitative risk assessment techniques and results can be used in the licensing process if proper consideration is given to the results of the Review Group. The staff should use the following procedures regarding the use of quantitative risk assessment techniques and results pending development of further guidance:

....

Quantitative risk assessment techniques may be used to estimate the relative importance of potential nuclear power plant accident sequences or other features where sufficient similarity exists so that the comparisons are not invalidated by lack of an adequate data base....

The quantitative estimates of event probabilities in the RSS should not be used as the principal basis for any regulatory decision. However, these estimates may be used for relative comparisons of alternative designs or requirements provided that explicit considerations are given to the criticisms of those estimates as set forth in the Report of the Risk Assessment Review Group.

The RSS consequence model shall not be used as the basis for licensing decisions regarding individual nuclear power plant sites until significant refinements and sensitivity tests are accomplished. However, the consequence model may be used for relative comparisons provided that such estimates are not the primary basis for such reviews and provided that explicit consideration is given to the criticisms of the various elements of that model as set forth in the Report of the Risk Assessment Review Group."

The Commission went on in this memo to direct the staff to expand its use of probabilistic risk assessment:

"The staff shall give special attention to those activities identified by the Review Group as being especially amenable to risk assessment, i.e., dealing with generic safety issues, formulating new regulatory requirements, assessing and re-validating existing regulatory requirements, evaluating new designs, and formulating reactor safety research and inspection priorities."

Given the content of the Commission's statement on the Lewis Report and the directive to the Executive Director for Operations, the Commission believes that it holds essentially the same position on the use of PRA now as it had on January 18, 1979. *

With regard to Mr. Bender's remarks appended to the September 15, 1982 ACRS letter, we agree with Mr. Bender that there are large uncertainties in the quantitative assessments of risk from nuclear power plant accidents. These uncertainties arise from several areas, including: (1) inadequacies in the data base; (2) incomplete present knowledge of core melt phenomena, in-plant fission product transport, and containment performance; (3) the effect of unidentified systems interactions; (4) difficulties in quantitatively modeling human behavior; and (5) large uncertainties in the risk from external initiators. However, we believe that the data base is not as poor as implied by Mr. Bender; there are programs underway to develop a better understanding of core melt phenomena, containment performance, and fission product transport, and to improve the probabilistic assessment of external events.

Commissioner Gilinsky adds:

"My own views on the usefulness and the limitations of 'probabilistic risk assessment' and its use in the Reactor Safety Study are still pretty much as expressed in the (unanimously adopted) Commission statement of January 18, 1979. I am not at all in agreement with the current Commission's increasing tendency to view probabilistic risk assessment together with a quantitative 'safety goal' as a shortcut to regulatory decisionmaking. I am particularly concerned about resort to these calculational techniques in combination with sparse data to explain away the need for the traditional independent safety barriers which have been chosen on the basis of experience and engineering judgment. I have the impression that Mr. Bender and I are in philosophical agreement on these points. To cite one example that I find especially telling on the paucity of equipment reliability data, it was not until last year that full-scale tests were run on the large safety valves used to protect against excessive pressures in reactor coolant systems. And even these tests did not cover the full range of conditions to which such valves might be subject."

The majority of the Commissioners do not agree with his statement that the Commission is tending "to view probabilistic risk assessment together with a quantitative 'safety goal' as a shortcut to regulatory decisionmaking." *

Commissioner Asselstine adds:

"Since I did not participate in the development of the Commission's view on the usefulness of the PRA methodology as given in the January 18, 1979 statement, I defer to my colleagues as to whether there has been a change in that view since then. I do believe that, with this Commission's consideration of a safety goal containing quantitative benchmarks for judging an acceptable level of risk, there is necessarily a greater emphasis on the use of the PRA methodology than would otherwise exist. Because of the wide spectrum of expert views on the ability of the PRA methodology to provide reliable estimates of the risk associated with the operation of nuclear reactors, I believe the basis for safety must continue to depend on compliance with our regulations and on the judgment of responsible individuals. On the latter, judgment is aided significantly through systematic reviews and careful analyses of available information. I believe the PRA methodology has a role to play here, provided that the Commission adheres to its view of January 18, 1979, and provided that the concerns expressed by Mr. Bender and others are properly accounted for."

I trust that this has been responsive to your concerns.

Sincerely,

Original Signed By
John F. Ahearne

John F. Ahearne
Acting

cc: Rep. Manuel Lujan

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