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April 15, 1971

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U. S. Atomic Energy Commission
Washington, D. C. 20545

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In the Matter of Consolidated Edison Company of New York, Inc.
Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247

Gentlemen:

The responses of the AEC regulatory staff to the questions asked by the presiding atomic safety and licensing board at the hearing session on January 19, 1971 are attached herewith.

Sincerely,

Myron Karman

Myron Karman
Counsel for AEC Regulatory Staff

Enclosure
As stated

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RESPONSES OF THE DIVISION OF REACTOR LICENSING TO THE QUESTIONS OF THE ATOMIC SAFETY AND LICENSING BOARD AT THE HEARING SESSION DATED JANUARY 19 1971

(Tr. 486)
Question

"As I recall the staff answered this question rather briefly that the statement was made that WASH-740 was irrelevant to the present consideration and there was some small discussion of this.

I would like to ask that the staff look again at Report WASH-740, at TID-14844, and to tell again whether these two reports are irrelevant, if they are, why; if they are not, what has changed since the time of these reports to make the situation different from what was reported."

Answer

The report WASH-740, titled "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," published in March 1957, is considered by the regulatory staff not to be relevant to the present review. Specifically, in this report three types of releases of fission products were assumed in order to estimate a range of public hazards which could result from such events and to delineate the influence of important variables on the magnitude of these hazards. The first case is one in which it is assumed that none of the radioactive material released to the interior of the containment is released to the environs. The hazard would be from direct radiation. Since the containment vessel of the Indian Point 2 reactor is constructed of concrete the shielding thus provided makes this hazard insignificant. The second and third cases assume that major fractions of the fission products in the reactor core are released from the containment to the environs. For these cases to occur would require the rupture of the primary system, and presence of significant leak paths in the containment wall. In the case of Indian Point 2, the failures mentioned are guarded against by engineered safety features, design safety margins, in-service inspection requirements, and the quality assurance aspects of construction.

The following are among the aspects in design of Indian Point Nuclear Generating Unit No. 2 which were not considered in the determination of the nature and extent of the fission product releases discussed in WASH-740.

1. The primary system is designed and fabricated (a) to high standards set by special standards groups within the industry, (b) to include the effects of radiation on the materials of construction, (c) to

include requirements for periodic in-service inspection, and (d) to include the effects of seismic loading on the system.

2. A core meltdown is prevented by the inherent stability of the reactor due to the requirement for a negative power coefficient, and the emergency core cooling system is designed to provide multiple sources of cooling water to replace primary system water lost from a postulated severe primary system break.

3. The containment structure is designed with safety margins to accommodate increases in pressure and temperature caused by release of primary coolant into the containment atmosphere. Piping and other containment penetrations are provided with double isolation capability. All of these containment features are also designed for seismic loading.

The report TID-14844, titled "Calculation of Distance Factors for Power and Test Reactor Sites," published in March 1962, has been used as guidance for the evaluation of all sites for water-cooled reactors since its publication. It is relevant. In addition to providing guidance for the interpretation of 10 CFR Part 100, TID-14844 specifies values of fission product releases and other fundamental and conservative assumptions which are used by the staff to calculate the consequences of design basis accidents. It should be noted that the source terms of TID-14844 are used as conservative values for reactor site evaluation purposes without regard to the existence or non-existence of a credible mechanism which would produce the corresponding quantity of fission products in a form available for release. Specifically, we do not consider the mechanism of an extensive meltdown of the core credible. Nevertheless, for site evaluation purposes, we do use values for releases which might correspond to such a mechanism.

(Tr. 487)
Question

"In connection with Dr. Briggs' question about WASH-740, the whole problem, a very complex problem of risk versus benefit versus cost in connection with these environmental matters has been brought up in discussion earlier (See Transcript of December 17, 1970, Roisman's opening statement at p. 307) in this hearing. It might be interesting to hear the staff in particular addressing itself to how it considers this problem."

Answer

The fundamental risk-benefit decisions with respect to industrial development of nuclear power have been made by Congress in the Atomic Energy Act of 1954, as amended. Under the Act, power reactors may be constructed and operated subject to licensing and regulatory requirements to protect the health and safety of the public and the common defense and security.

The Commission's rules, regulations, and guides include standards and requirements which represent an accommodation of risks and benefits in implementation of the responsibility conferred on the Commission by the Act. An example is the General Design Criteria for Nuclear Power Plants (10 CFR Part 50, Appendix A). See e.g. Criterion 17, Electrical Power Systems.

In addition, the policies and practices of the regulatory staff as developed over the years in consultation with other expert bodies, including the Advisory Committee on Reactor Safeguards, embody additional risk-benefit decisions, which are implemented in such publications as the General Design Criteria, Appendix A of 10 CFR Part 50.

Finally, certain decisions which involve the assessment of cost versus benefit or risk are made on a case-to-case basis. An example would be the requirement for installation of equipment which is necessary for public health and safety, viewed against plant lifetime, but need not be installed immediately because the delay incident to such installation outweighs the small incremental benefit which would be obtained by postponing plant operation until the backup equipment is installed. An example would be the regulatory staff's decision in consultation with the ACRS to require the installation at a later time of backup equipment to prevent hydrogen buildup in the containment following a loss-of-coolant accident. This backup system will be in addition to redundant flame recombiners which will be available for initial operation.

(Tr. 491)
Question

"Speaking of research and development, the Board is concerned concerning the reports issued by the Advisory Committee on Reactor Safeguards over a period of time in reference to pressurized water reactors, and I wonder if a summary can be presented of what those concerns are as having been expressed by the Advisory Committee on Reactor Safeguards over, say, the last ten years because the ACRS, and I refer to them as the Advisory Committee on Reactor Safeguards, concluded many of its reports by saying if these matters are carried out then there is reasonable assurance that the reactor can be operated without undue risk to health and safety of the public."

Answer

The Advisory Committee on Reactor Safeguards (ACRS), upon completion of its review of each reactor, issues a report (letter) to the Chairman of the Atomic Energy Commission. The conclusion of a favorable report typically is that there is reasonable assurance that the reactor can be operated without undue risk to the health and safety of the public, predicated on certain matters being carried out. These matters are identified by the ACRS during the review as requiring special attention. These matters often are identified during prior reviews of similar facilities. Over the past few years (post-1967), these matters related to pressurized water reactors have been:

a. Common mode failures in control and protection instrumentation.

b. Consequences of failure to scram during anticipated transients.

c. Post-loss-of-coolant accident hydrogen control.

d. Instrumentation for in-service monitoring for excessive vibration or loose parts in the primary system.

e. Effects of local high temperature or pressure in the core during an accident and possible interaction of failure in adjacent fuel elements.

f. Use of additives in the containment spray system.

g. Effect of loss-of-coolant accident environment on long-term availability of vital components.

h. Consequences of contamination of ECCS water by structural materials and coatings in a loss-of-coolant accident.

i. Swelling of boron carbide control rods.

j. Evaluation of the fuel handling accident.

k. Protection against internally and externally generated missiles.

l. Effect of spatial power distribution on xenon oscillation.

m. High power density core.

n. Thermal shock of reactor vessel caused by emergency core cooling system action.

o. Capability of vessel cavity walls to withstand the mechanical forces of a loss-of-coolant accident.

p. Quality Assurance.

q. Loss of off-site power.

r. The design and analysis of the function of emergency core cooling systems.

s. The capacity of diesel-generator systems.

t. Instrumentation for prompt detection of failed fuel.

u. Requirements for in-service inspection.

v. Separation of control and safety function in instrumentation systems.

w. Primary system leak detection.

In certain instances, responsibility for the research and development work associated with these matters has been assumed by the applicant or the nuclear steam system supplier. In other instances, the applicant and his suppliers are committed to follow work being performed by others. Those matters in which the AEC has sponsored research and development are discussed in the response to the next question.

Each of these items is being or has been reviewed and resolved in the context of our continuing safety review of nuclear power plants being licensed. This work is being conducted on a time scale that is consistent with the developing technology in the given area, and thus the status of each varies from "completely resolved" to "continued effort" depending upon the complexity of the problem, the amount of information available on the problem, and how recently the problem was identified. Many of the problems identified in the ACRS letters are complex and will take time to resolve in a responsible, complete, and thorough manner. Some of the items, of course, involve the acquisition of data which can be best obtained through operating experience. The nature of technology is such that there will probably be continuing studies in some of these areas for many years to come and new studies in areas not as yet identified, in order to further our understanding of the phenomena involved.

(Tr. 491)
Question

"Aside from a summary statement or in addition, let me say, to a summary statement in that regard and updating of the experimental test data under those research and development projects, I wonder if we could have a witness from the staff of the Atomic Energy Commission about the research and development work. I think some boards in the past have had difficulty with summary statements maybe not being as complete as they would like to have it. If a witness is present then I think any further inquiry the Board may have can be readily considered and answered at that time.

For instance, as I recall it, there is a loss-of-fluid test. That has been going on for sometime, and maybe we can have some data about that and the other R&D programs that ACRS has outlined...

Are they carried on with the same vigor and financial support, for instance, that heretofore has been allocated to other projects and what has been discovered to date and what more is left to be done and when will that work be done and what is the data that is expected to be derived from further work in that regard?...

I think it is important that we have a witness from that work, a witness that has a responsible position.

Maybe it would be the director of the reactor development technology himself to participate in this hearing; I think it would be very helpful if he would."

Answer

A report on the water reactor safety program plan was published in February 1970. Prepared by the Water Reactor Safety Program Office, an independent staff administered by Idaho Nuclear Corporation, the plan brings together all of the current safety-oriented research for water reactors being funded by the AEC. Comments from industry and the various regulatory bodies were incorporated in the plan.

A copy of this report (WASH-1146) is provided herewith.

(Tr. 495)
Question

"On page 113 of the detailed statement on environmental considerations by the staff...we find HEW's statement something to this effect: The estimate of liquid radioactivity discharges and so forth, in our judgment, is not adequately documented.

What do they want in order to make the reviews? Did the staff get this to them? Is there anything further from HEW other than that which is reflected in the staff detailed environmental statement reflected on page 113?

In fact, is there any supplementary communication to any of the agencies to which the Applicant's statement is submitted?"

Answer

The Department of Health, Education, and Welfare was furnished a copy of the Consolidated Edison Company, Indian Point 2 FFDSAR with subsequent amendments. In addition the Department was asked to comment on the content of the applicant's environmental report under the requirements of NEPA. The Department's comments were received by the Commission and sent to the applicant for comment. The HEW comments and the applicant's reply to these comments are included in the Detailed Statement on Environmental Consideration as Appendices F and K, respectively. The Department made no request for additional information, it has not at this time indicated its intent to conduct further review, and the AEC at this time has not had further communication with the Department on this subject.

In its comments, HEW questioned the documentation of the estimates of liquid radioactive effluents presented by the applicant in the FFDSAR. The AEC recognizes that the estimates are based on design criteria. As discussed in the response to the following board question on page 9 (including the tables delineating releases of radioactivity from PWR plants), actual operating experience is required to determine the variance from the estimates.

(Tr. 495)
Question

"Then there is this further statement shown on page 113 of the staff detailed environmental statement which says something like this: Current PWR, I take that as 'pressurized water reactors,' operating experience indicated that both the liquid radioactive discharge and gaseous discharges will be considerable higher and the Applicant has not... (presented) new design...(information) to support the lower...(estimated) discharges. Can the staff give us what figures reflect the current PWR operating experience and indicate that both the liquid and gaseous discharges will be higher, higher than what, the Applicant considered, or what has been designed in other reactors and what kind of design information does HEW believe will be necessary for it to support or give a conclusion respecting the estimated lower discharges?"

Answer

PWR operating experience in 1969 with respect to liquid and gaseous discharges are indicated in the enclosed Tables I, II, and III. The information tabulated was obtained from the periodic reports of licensees and supplemented by information gathered during inspections by the AEC Division of Compliance. Similar tables for 1967 and 1968 have been published in "Selected Materials on Environmental Effects of Producing Electric Power," Joint Committee on Atomic Energy, August 1969 (pages 116-118). In regard to the estimates of liquid effluents presented in the Indian Point 2 FSAR, it is recognized that the numbers quoted are design values based on design criteria. Determination of the actual numbers will of course require operating experience.

TABLE I

RELEASES OF RADIOACTIVITY FROM POWER REACTORS IN LIQUID EFFLUENTS, 1969

MIXED FISSION & CORROSION PRODUCTS

Facility	Released (Ci)	Concentration Limit 1/ (10^{-7} uCi/ml)	Percent of Limit 2/	Released (Ci)	Percent of MPC 3/
DRESDEN 1 *	9.5	1	22	~ 6	< 0.001
SAN ONOFRE	8	1	14	3500	0.2
HUMBOLDT BAY *	1.5	1	8.7	< 5	< 0.001
NINE MILE POINT *	0.9	1	8.2	< 1	< 0.001
BIG ROCK *	12	22	5.6	28	0.01
OYSTER CREEK *	0.48	1	4.1	5	0.001
SAXTON	0.01	1	2.5	< 1	0.008
INDIAN POINT 1	28	37	1.5	1100	0.07
CONN. YANKEE	12	12	1.4	5200	0.24
GINNA	0.02	1	0.4	< 1	< 0.001
LA CROSSE *	8.5	300	0.11	~ 25	0.003
YANKEE	0.019	1	0.07	1200	0.14
PEACH BOTTOM **	0.001	1	0.002	40	0.031

* Boiling Water Reactors
** Gas-cooled Reactor

TABLE II

RELEASES OF RADIOACTIVITY FROM POWER REACTORS IN GASEOUS EFFLUENTS, 1969

Facility	NOBLE AND ACTIVATION GASES Curies Released	Permissible[4/]	Percentage of Permissible	HALOGENS AND PARTICULATES Curies Released	Permissible[5/]	Percentage of Permissible
DRESDEN 1	800,000	22,000,000	3.6	0.26	85	0.3
SAN ONOFRE	260	567,000	0.045	< 0.0001	0.8	< 0.001
HUMBOLDT BAY	490,000	1,560,000	31	0.65	5.6	12
NINE MILE POINT	55	25,800,000	< 0.001	< 0.001	63	< 0.001
BIG ROCK	200,000	31,000,000	0.65	0.2	38	0.53
OYSTER CREEK	7,000	9,450,000	0.075	0.003	126	0.002
SAXTON	1	3,750	0.035	< 0.0001	10	< 0.001
INDIAN POINT 1	600	5,360,000	0.01	0.025	7.6	0.33
CONN. YANKEE	190	18,900	1	0.001	0.27	0.37
GINNA	1	360,000	< 0.001	< 0.0001	1.7	< 0.001
LA CROSSE	480	480,000	0.1	< 0.063	1.6	< 4
YANKEE	4	6,600	0.062	< 0.0001	0.03	0.01
PEACH BOTTOM	72	189,000	0.038	< 0.0006	0.12	< 0.5

TABLE III

COMPARISON OF EFFLUENTS 1967 - 1969

	LIQUIDS			GASES		
	Curies Released			Curies Released		
Facility	1969	1968	1967	1969	1968	1967
DRESDEN 1	9.5	6	4.3	800,000	240,000	260,000
SAN ONOFRE	8	1.5	0.32	260	4.8	4
HUMBOLDT BAY	1.5	3.2	3.1	490,000	897,000	900,000
BIG ROCK	12	7.9	10	200,000	232,000	264,000
SAXTON	0.01	0.009	0.02	1	18.6	22
INDIAN POINT 1	28	34.6	28	600	55.2	23
CONN. YANKEE	12	3.8	0.39	190	3.7	0.02
LA CROSSE	8.5	0.074	< 0.005	480	< 1	< 5
YANKEE (ROWE)	0.019	0.009	0.056	4	0.66	2.3
PEACH BOTTOM	< 0.001	< 0.001	0.002	72	108	7.5

FOOTNOTES FOR TABLES

1/ Facility licenses require that the release of radioactive liquids in plant effluents be in accordance with 10 CFR Part 20, "Standards for Protection Against Radiation." For mixtures of radionuclides in the effluent, Part 20 provides two alternatives for determining permissible concentration limits. If the identity and concentration of each nuclide is known, Appendix B, Note 1, prescribes a formula for calculating the limiting value. Note 3 prescribes a method for selecting one of a series of values if it can be shown that certain radionuclides are not present in the mixture. The values calculated or selected by licensees may vary from year to year.

2/ One of the limits specifically mentioned in Note 3.c. of Part 20 is 1×10^{-7} uCi/ml, which is sufficiently restrictive that it can be used for any mixture of fission and corrosion products in water from any nuclear power reactor without any identification of the radioisotopic composition of the mixture. Typical isotopic compositions of radioactivity in water from power reactors are such that limits higher by two orders of magnitude or more are expected to be available to the licensee if he wishes to support them with adequate radioisotopic analyses. The percent of limit given in this column generally represents upper bounds to the value that would be applicable on the basis of a complete analysis of the composition.

3/ The maximum permissible concentration of tritium in water is 3×10^{-3} uCi/ml.

4/ Where the technical specifications express a release limit in terms of a constant factor times the 10 CFR Part 20 concentration limits, the MPC used is 3×10^{-8} uCi/cc. This MPC is based on typical noble mixture releases with less than two hours holdup. (For a holdup longer than two hours the MPC is larger.)

5/ Where the technical specifications do not state an annual limit for the iodines and particulates, values of 1×10^{-10} uCi/cc and 3×10^{-11} uCi/cc, respectively, were used. These MPC's are based on the most restrictive isotopes normally found--I-131 and Sr-90. The annual limit was reduced by a factor of 700 to account for reconcentration.

(Tr. 497)
Question

"Apparently the position taken by HEW is said to be taken because gaseous releases during normal operations at Indian Point No. 1 have been much higher than at other similar operating PWRs which could be interpreted to indicate that the gaseous waste holdup was not used to the fullest extent, and so forth.

Could the staff get those figures or could the Applicant? What were the releases from Indian Point No. 1 which were higher than other similar operating PWRs? What are other similar PWRs and what were the figures for releases from them?"

Answer

A tabulation of gaseous effluents from operating nuclear power plants is included in Tables I, II and III.

(Tr. 500)
Question

"In the report there is indicated that certain changes or conditions will be required such as purging the containment or removal of the hydrogen, adding filters to the ventilation system.

I would like to have an indication as to why these changes or additions are not required before the plant goes into operation, why it is possible to let some changes or additions come along a year or two or three years after the plant begins to operate.

What considerations led to the conclusion that these could be delayed?"

Answer

The ACRS, in their letter of September 23, 1970, recommended that equipment for controlled purging of the containment atmosphere should be designed and provided during the first two years of operation at power. For Indian Point 2, the purging system is considered to be a backup to the redundant flame recombiners which will be available at initial operation. In our judgment, the two year period is considered to be reasonable. Plant operation for this period without the backup purging system is acceptable.

In its letter, the ACRS also recognized that the installation of a charcoal filter system in the refueling building would be completed by the end of the first year of full power operation. The purpose of this filter system is to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. Since fuel handling of irradiated fuel will take place 18 to 24 months following initial operation, a period of one year is a reasonable time for installation of the system.

Enclosures
1. WASH-1146
2. Tables published in "Selected Materials on Environmental Effects Producing Electric Power"