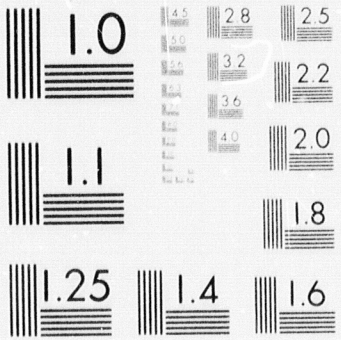

MICROCOPY RESOLUTION TEST CHART
NATIONAL BUREAU OF STANDARDS-1963-A

AUTOMATION INDUSTRIES, INC.
VITRO LABORATORIES DIVISION
14000 GEORGIA AVE
SILVER SPRING, MARYLAND 20910
(301) 871-7200

est residence (750 feet) is small, about 0.1 roentgen, due to direct radiation from fission products in the coolant system.

The hazards associated with a complete core meltdown have been computed, though it appears highly unlikely that such an incident would occur. Results are indicative of the maximum possible hazard. It was assumed that complete meltdown of a full life core occurs during full power operation, releasing 100 percent of the volatile fission products (including iodine) to the reactor building, and an internal container pressure of 1 psig exists during the entire release. One hundred percent of the volatiles constitutes 38 percent of the total fission product inventory. It is reasonable to assume that essentially all of the nonvolatiles would be retained by the coolant, since the coolant would not be vaporized. The following tabulation represents the probable hazards at the nearest controlled area boundary (750 feet) from direct building radiation, direct radiation from the cloud, and the total integrated thyroid dose, assuming release of the cloud to the atmosphere occurs only through normal leakage from the building. Inversion conditions were assumed.

TOTAL INTEGRATED DOSE RATES AT 750 FEET RESULTING FROM CORE MELTDOWN ACCIDENT

Exposure time	Direct radiation	Direct cloud radiation	Thyroid dose
1 day	22 rem	3.2 rem	36 rem.
1 week	44 rem	6.8 rem	Not applicable.

Examination of the above results indicates that this hypothetical "worst possible" situation would not result in injurious dose rates, particularly in view of the opportunity for evacuation from the nearby areas. Any credible accidents would cause substantially lower exposure possibilities.

Atomics International Division of North American Aviation began activities in the nuclear field in 1946. Among its major products in the nuclear field since that time have been research and development programs involving sodium cooled reactor systems and organic moderated reactor systems. In connection with the former, the Company constructed and operated the Sodium Reactor Experiment, and with the latter, the Organic Moderated Reactor Experiment, both for the Commission. The company has built thirteen solution type research reactors for Government and private organizations in the United States and abroad. In addition, the Company has been a contractor for the Atomic Energy Commission in the developing of reactors, including the Kinetic Experiment on Water Boilers (KEWB) and Satellite Nuclear Auxiliary Power Systems (SNAP), and in conducting various developmental studies and has participated in various research and development programs relating to reactor systems. The Atomics International Division now has approximately 2,500 employees, of whom 285 hold advanced degrees.

The Piqua organic moderated reactor was preceded by the Organic Moderated Reactor Experiment (OMRE). This was constructed to demonstrate the feasibility of the organic reactor concept and to study the behavior of organic coolants under power reactor operating

conditions. The OMRE consists of a 12 megawatt thermal reactor and has been operated at conditions generally similar to those proposed for the Piqua reactor since September 1957, accumulating a total of 1,400 megawatt days of service to date on two cores.

The design, construction and safety features under the operating procedures of the Piqua reactor are based on this experience. To date the performance of OMRE has been satisfactory.

The Advisory Committee on Reactor Safeguards has reviewed available matter concerning the proposed Piqua reactor, including the Preliminary Safeguards Report issued by North American on April 13, 1959, and the supplement thereto issued July 1, 1959. In its Report to the Chairman of the Atomic Energy Commission dated July 25, 1959, the Committee stated:

> "The Committee's comment on previous proposals have been reported. In the most recent letter upon such proposals, May 18, 1959, it was stated: '. . . the site as now proposed may be considered as not creating an undue public risk provided (a) adequate containment is constructed as now described by the applicant, (b) the maximum leakage rate for the containment is reduced to an acceptable low value, and (c) that this relatively new reactor system is adequately designed.
>
> "From the information obtained and discussed, it appears requirements (a) and (b) will be satisfied and progress toward (c) adequate design and control of this new reactor system, is satisfactory.
>
> "Dependent upon meeting the qualifications above, the Committee concludes that this reactor, as now described, may be constructed and operated at the site selected without undue risk to the health and safety of the public."

In addition to the foregoing findings and conclusions, and based upon the evidence of record, the contentions of the participants and the proposed findings and conclusions, it is further concluded as follows:

1. The City of Piqua, Ohio, is a municipal corporation organized and existing under the laws of Ohio and is qualified and authorized to engage in the generation and distribution of electric power in territory adjacent to the City of Piqua.

2. The Atomics International Division, North American Aviation, is technically qualified to design and construct the reactor.

3. Sufficient information has been presented to provide reasonable assurance that a nuclear reactor of the general type planned can be constructed and operated at the site described in the Preliminary Safeguards Report without undue risk to the health and safety of the public.

4. Construction of the reactor at the site will not be inimical to the common defense and security or to the health and safety of the public.

5. There is reasonable assurance that technical information omitted from, and required to complete, the Preliminary Safeguards Report filed by Atomics International Division, North American Aviation, Inc., will be supplied; and

6. No good cause has been shown why Atomics International Division, North American Aviation, Inc., should not proceed with construction of the reactor at the site.

WHEREFORE, IT IS ORDERED, subject to review by the Commission upon its own motion or upon appeal, after exceptions, if any, are filed:

A. The Commission authorizes Atomics International Division, North American Aviation, Inc., to proceed with the construction of the reactor in accordance with the design set forth in the Preliminary Safeguards Report and the supplement thereto filed by North American with the Commission.

B. This authorization is provisional to the extent that an authorization to operate the facility will not be issued by the Commission unless a final Hazards Summary Report is filed by the company and until the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility.

C. Exceptions, if any, and briefs in support thereof, must be filed by January 27, 1960; briefs in opposition thereto shall be filed by January 29, 1960; and if the Commission does not initiate a review on its own motion, and no extensions are filed, this decision shall, in accordance with the Commission's Rules of Practice, become final on January 28, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

BYPRODUCT MATERIAL LICENSE NO. 4-1695-2B59 AND NO. 4-5744-1

IN THE MATTER OF ADVANCE INDUSTRIAL X-RAY LABORATORIES, A DIVISION OF AIR FRAME INSPECTION, INCORPORATED, RESPONDENT AND ADVANCE INDUSTRIAL X-RAY LABORATORIES, INC., APPLICANT

Issued January 12, 1960

APPEARANCES:

Robert L. Thatcher, Esq., for Applicant.
Max Singer, Esq., for Staff (*Robert Lowenstein, Esq.*, Counsel for Division of Licensing and Regulation with him on the brief).

INTERMEDIATE DECISION

Advance Industrial X-Ray Laboratories, Inc. (Advance), Los Angeles, Calif., filed an application on April 15, 1959, for a Byproduct Material License which would authorize it to use byproduct materials for the performance of radiographic services, such as testing welds and other metal fabrications. Advance was organized in part by employees of licensee Air Frame Inspection, Inc., also of Los Angeles (Air Frame), which encountered financial difficulties in February and March 1959 and ceased doing business in May of that year. Air Frame purchased a radiographic business in 1957 and operated it within what ultimately became a division of its operations under a designation which was similar to the name of the present corporate applicant for license. During the course of operations of this division by Air Frame, an incident occurred wherein there was loss of some byproduct material and several persons were exposed to radiation which in some cases may have been in excess of permissible levels.[1]

Advance, as the applicant in these consolidated proceedings, has undertaken to continue the radiographic services once performed by Air Frame. Prior to the filing of the application, Advance informed the San Francisco office of the Commission of its intentions to acquire and continue the radiographic services undertaken by Air Frame, and thereafter, Advance filed its application. In connection with this transfer, Air Frame, by letter to the Commission, stated that it ". . . would like to cancel . . . (its license) . . . in favor of Advance . . . Advance is being maintained and operated under the same personnel and equipment as shown on attached sheet." (Parenthesis supplied).

Hearings on the application filed by Advance were held in Los Angeles on September 16 and 17, 1959. No other persons other than Air Frame, Advance and the Staff entered appearances. Briefs were

[1] The circumstances surrounding this incident were the subject of a previous hearing held in Los Angeles in July 1958.

filed on November 19 and 27, 1959. Included with the hearing on the application were two other, but related proceedings: One, a remand[2] for possible modification or disposition of the earlier proceeding involving Air Frame which arose out of the loss of byproduct material; and, second, an order to show cause why Advance should not discontinue using byproduct materials then licensed for use by Air Frame. Advance filed an answer to this show cause order alleging that it was an agent or a division of Air Frame and authorized to use the licensed byproduct materials. Prior to the hearing Advance consented to discontinue the use of the byproduct material pending these proceedings.

After completion of the hearings, Advance filed a motion seeking a temporary license to permit it to resume radiographic services, and with no objection by the Staff, an order to that effect was entered. Thereafter, the Division of Licensing and Regulation issued a temporary license to Advance.

The hearings in these consolidated proceedings were expedited by the stipulation proposed by the Staff and accepted by Air Frame and Advance, substantially to the effect[3] that Air Frame be and is no longer a party to the proceedings, but that considerations of the radiographic operations by the individuals acting either for Air Frame or Advance would be material in determining the qualifications for Advance to receive its requested license. The stipulation therefore permits a comparison of radiographic operations under Air Frame and the improvements, if any, under Advance. The Staff has concluded that, while there are some improvements, they are not such as to be beyond criticism, but nevertheless reflect adequate compliance with the regulations of the Commission and the operating procedures proposed by Advance and the former procedures of Air Frame. Therefore, the Staff recommends that a license issue to Advance for a limited period pending further review. As appears hereinafter, this conclusion and recommendation of the Staff are adequately supported by the extensive record herein.

The consideration of the application by Advance necessarily includes the prior operations undertaken by the same individuals acting under the previous Air Frame management. It appears that a substantial responsibility for the lack of compliance by Air Frame with Commission regulations, and the operating procedures devised by Air Frame, was due to a division of authority and inadequate managerial control over its radiographic services. The 1958 hearings respecting Air Frame established a lack of adequate and effective control prior to the May incident involving the loss of the capsule of byproduct material. Air Frame endeavored to adopt corrective measures after May, and at the hearing in July, requested advice as to further improvement in operating procedures. The hearings held in the instant proceeding in 1959 indicate that, before completely adequate compliance with regulations and procedures could be attained by Air Frame, it was beset with financial difficulties. As a result, those individuals who were actually engaged daily in the radiographic work sought to separate out this activity from Air Frame, and under the guidance of one of them, created a separate corporation.

At the recent hearings and in its brief, the Staff detailed significant items which occurred in the activities of Air Frame as licensee, and the services undertaken by Advance, either as an unlicensed user of Air Frame's byproduct material, or as the claimed agent of Air Frame.[4] This review included the procedures respecting possible exposures to radiographic service, use of properly calibrated survey detection instruments, the periodic testing and calibration of survey detection instruments, the proper and regular use of pocket dosimeters and film badges, and the sufficiency of instructions to personnel conducting gamma ray inspections. As reflected in the 1958 hearing, Air Frame had deficiencies in all of the foregoing described phases of radiographic work.

With this background for the 1959 hearings, the Staff developed that most deficiencies had been removed by the close supervision conducted by the now principal officer of Advance, who had been given individual authority over the radiographic operations after Air Frame had sustained the May 1958 incident. The improvements shown in the changed operations resulted in (1) five employees, including both of the field supervisors, being given formal instruction in radiation safety by Dr. Gordon Locher of Western Radiation Laboratories; (2) personnel engaged in using gamma rays for radiographic work receiving copies of statements of radiographic operating procedures and certifying that they read and understood the procedures, (3) calibrated and operable survey instruments being used regularly, (4) adequate surveys of areas for radiographic services being properly posted and roped, as well as being measured with detection instruments to determine areas for permissible limits of radiation; (5) a program of management supervision of field safety practices being conducted; (6) radiation survey instruments being inspected and calibrated at regular intervals, dosimeters being assigned to all gamma ray personnel; and (8) the readings being recorded in a comprehensible and useable manner.

Substantial as are these improvements in the radiographic work, some aspects are still subject to betterment, particularly in reference to the preparation and maintenance of records of surveys and dosimeter readings.[5] It was evident at the recent hearings that Advance sought suggestions for improved procedures in this regard, and indicated that it would adopt and incorporate into its operating pro-

[2] The Order of the Commission, dated July 15, 1959, prescribed the following matters for consideration:
(a) The effect upon the proceeding of the alleged change of the ownership and management of Respondent, Advance Industrial X-Ray Laboratories;
(b) Whether or not a license should be issued to Applicant Advance Industrial X-Ray Laboratories, Inc.;
(c) Compliance by Respondent and its personnel, including such personnel as may now be connected with Applicant, with the terms of the Temporary Order dated June 13, 1958, as modified;
(d) Such other matters as may appropriately come before the Hearing Examiner.

[3] Specifically, the stipulation proposed by the Staff was:
(a) Whatever be the decision on the application for license, certain operating procedures must be more adequately developed.
(b) Air Frame wished to terminate participation in the proceedings, and the Staff and Advance agree this renders the "old proceeding moot."
(c) Advance would cease using the byproduct material licensed to Air Frame.
(d) Advance admits its operations since January 23, 1959 to be unlicensed operations, but that Advance has not wilfully violated the Act or regulations.
(e) Any violations by Air Frame are relevant in the consideration of the application filed by Advance.

[4] At the hearing, as shown in the stipulation, Advance admitted that it was acting without authority in using Air Frame's byproduct material.

[5] Advance contended at the hearings that present Commission regulations, particularly Sections 20.401 and 20.201, do not prescribe the exact method of maintaining records.

cedures changes that would more fully comply with Commission regulations.

In summary, it is reasonable to conclude that, while the existing preparation and maintenance of the records may not be in strict compliance with Commission regulations, nevertheless, adequate records were made and reasonable compliance was had. This conclusion, when coupled with the specific commitment of Advance to comply with the regulations, as Advance now understands the requirements, provides reasonable assurance that, if Advance were to be licensed, there will be no violations in this regard.

A Staff witness testified that four principal considerations were important in making a recommendation to issue a license for the use of byproduct materials; the equipment proposed to be used, the facilities available, the procedures proposed by the applicant as the method for undertaking the proposed use of the byproduct material, and the qualifications of the applicant and its staff to safely conduct the program that he proposes to conduct. The recommendation reflects a composite consideration; and, any one of the four aspects may have a greater weight in varying cases. In addition, even after initial licensing, a change in practices or operations by a licensee may warrant a recommendation to alter the licensed activities by the imposition of new conditions, or to permit changes in the operating procedures, which are devised by a licensee, and submitted to the Commission for approval. In the instant case, two of the alleged violations, pertaining to Air Frame or Advance, were based on operating procedures that proved unworkable, or at least, difficult to adhere to, under the existing working conditions. One, for instance, was the requirement which designated the principal officer or employee to personally record dosimeter readings. At the hearings, this requirement was termed "impossible" by Advance. A violation in this regard appears to have been avoidable if a revised operating procedure has been submitted to the Commission, to specify another method of preparing and maintaining such records. An approval of such a change would have eliminated the burdensome requirement for the president of the licensee to stand by for the completion of radiographic services in the field, in order to write a record of dosimeter readings. The Commission, of course, cannot be advised whether an operating procedure is feasible unless information is presented to it.

The evidence in the instant case reflects the endeavors of the Inspection Division to assist this licensee in its compliance, but insofar as the record reveals, no previous mention was made of these two phases of operating procedures that were proving unworkable. The Staff, in its brief, apparently recognizes the character of such violations. Air Frame and Advance were providing other records of dosimeter readings which were adequate and Air Frame had had semi-annual instead of quarterly calibrations and checks of the detection instruments, which were adequate, yet, in the strict sense, constituted violations of its prescribed operating procedures.

The principal concern in these proceedings in reference to the above described four main considerations in licensing proceedings, is not whether the equipment or facilities were adequate, nor whether the prescribed operating procedures were sufficiently detailed and intelligible. Rather, it is whether the personnel would regularly and continuously comply with the regulations of the Commission, and these operating procedures. As is evident in this review of the record, substantial strides have occurred in operating practices involving these personnel, either under Air Frame's control or the initial steps undertaken by Advance.[6] It is, therefore, concluded, as the Staff recommends, that a license for a limited period should be issued to Advance with provision for special operating procedures which will permit the radiographic work to continue.

In connection with this disposition of the application filed by Advance, and in the light of the insolvency of Air Frame, it would not serve any useful purpose to do other than grant the joint motion made by the Staff and Advance to dismiss the proceedings involving Air Frame. One point that may be clarified in connection with that dismissal is to note that the prior Intermediate Decision, or Initial Decision as the Administrative Procedure Act terms it, made provision for close supervision by the Commission Staff over the particular users of the radiographic equipment utilizing the byproduct material license for use by Air Frame. This provision was made in view of the apparent endeavors of Air Frame to remedy its deficiencies in the operating procedures and to develop better and safer use of radiographic equipment. Special license provisions or conditions may vary from case to case depending upon the particular necessities of an applicant, as well as upon the developing nature of the regulatory program in reference to byproduct source materials. To this extent the consideration of special conditions proposed for licensing considerations cannot be relied upon as a precedent, due to varying circumstances affecting different applicants; and, therefore, the special conditions proposed in the prior Intermediate Decision are applicable only in that proceeding pertaining to Air Frame.

In the instant proceeding, other special conditions are believed advisable for Advance, such as provisions for the selection, training and testing of personnel using licensed materials, the establishment and statement of clear lines of authority and responsibility within the Advance organization, the techniques to be used to assure effective management control of radiation safety, and the monthly submission to the Atomic Energy Commission of the field radiography services performed by Advance.

In addition to the foregoing findings and conclusions, and upon the basis of the record of all of the evidence adduced herein, and the briefs and contentions of the parties herein; it is further found and concluded that:

1. Advance, the applicant for a license to use byproduct source materials, as provided by the Atomic Energy Act, as amended, is a California corporation, with address of 3538 Fowler Street, Los Angeles 63, Calif., whose Articles of Incorporation were filed on February 19, 1959. Applicant's Board of Directors is composed of Edwin Y. and Mary C. Martindale and Matthew Q. Bastardi. Edwin Y. Martindale is President.

[6] It cannot be said that Advance was wilfully endeavoring to defy regulatory supervision in using Air Frame's licensed byproduct material. Advance stated at the hearing that it thought that after reporting the acquisition of Air Frame's division to the San Francisco office, and sending the application for license to the Washington office of the Atomic Energy Commission, that either Advance would ". . . receive an answer that it wasn't proper, or be issued a license immediately."

2. On April 15, 1959, Advance Inc. filed with the AEC an application for a byproduct material license. Pursuant to a motion by the AEC Staff, dated June 23, 1959, the April 15, 1959, application was referred to the Presiding Officer by the Commission in its Order of July 15, 1959.

3. The Advance application covers six iridium 192 and two cobalt 60 sources of 40, 60, 10, 2, 10, and one curies of activity respectively, for use in the industrial radiography business. An Administrative Control and Operating Instruction is part of the application. Advance as an applicant is the holder of License No. 4–5744–1, issued September 25, 1959, pursuant to the Interim Order for Temporary License by the Presiding Officer of the same date.

4. From about February 22, 1959, to August 27, 1959, Advance conducted in its own name an industrial radiography business. In the course of this business employees of Advance possessed and used byproduct material which had been covered by Air Frame's license No. 4–1695–2B59.

5. License No. 4–1695–2B59, as renewed, under which Edwin Y. Martindale thought he was operating, required that Martindale prepare and maintain certain records of dosimeter readings, and for others to prepare and maintain survey records in reference to areas where radiographic services were performed.

6. As set forth above in reference to variances from the prescribed operating procedures, and in other aspects, there was no evidence concerning the operations by Advance which showed substantial non-compliance with the regulations, the Act, as amended, or the license under which Martindale thought he was operating.

7. Edwin Y. Martindale, the principal officer of Advance, has the technical background necessary to supervise the radiation safety responsibilities involved in the proposed license; and Richard Lent and Donald Harbison, the other users specified in the application filed by Advance, have the technical background necessary to act as field supervisors, under the general supervision of Mr. Martindale, for activities covered by the proposed license. The storage site and facilities available to the Applicant are adequate. An adequate arrangement for consulting and other services has been made with Dr. Gordon Locher of Los Angeles, Calif., who is qualified to perform such services.

8. The joint motion made by the Staff and Advance should be granted to dismiss the proceedings against Air Frame, pursuant to the remand of the proceedings by the Commission to the Presiding Officer.

Wherefore, It is Ordered, in accordance with the Commission's Rules of Practice, subject to review by the Commission on appeal or on its own motion, that:

1. The application dated April 15, 1959, filed by Advance Industrial X-Ray Laboratories, Inc., 3538 Fowler Street, Los Angeles 63, Calif., as applicant for a license to use byproduct materials is granted in accordance with the provisions of the Atomic Energy Act, as amended, and rules and regulations issued thereunder, subject to the following terms and conditions that the Division of Licensing and Regulation issue to Applicant a 1-year license to process and use byproduct material listed in its application of April 15, 1959:

a. The license shall not be issued until the Applicant submits by amendment of its application, satisfactory information as to:

(i) the qualifications and training which will be required for various jobs in connection with licensed activities;

(ii) the tests, certifications or other procedures which will be used by Advance as licensee to assure that each employee will have the required qualification and training;

(iii) the authority with respect to licensed operations and radiological safety which each position within the licensee's organization will be given;

(iv) the procedures and forms to be used by the licensee to assure maintenance of survey records;

(v) the procedures and forms which will be used by the corporate management of the licensee to assure that the requirements of the license are fulfilled by employees; and

(vi) such other matters as may be developed on a record through stipulation or by a hearing to be required by the regulations.

b. The license shall require that the licensee submit to the Commission on a monthly basis a list of the field radiography jobs that have been performed by it with byproduct materials; and that such list shall include for each job the name and address of the customer and the location and date of the job.

2. If the parties do not agree upon the terms of the foregoing provisions of the proposed license or the information to be furnished by the Applicant to complete the application, either party may by motion request the Presiding Officer to reopen the proceeding and issue such further order as may be appropriate.

3. The temporary license issued to the Applicant on September 25, 1959, pursuant to the Interim Order for Temporary License of September 25, 1959, shall be continued until a license is issued pursuant to this Order, provided that, unless extended by order of the Presiding Officer, on motion of any party, it shall expire 90 days after this Order becomes final.

4. The joint motion made by the Staff and Advance to dismiss all pending proceedings against Air Frame Inspection, Incorporated, including its Division known as Advance Industrial X-Ray Laboratories, is granted and all such pending proceedings are dismissed, and Atomic Energy Commission License No. 4–1695–2B59 as heretofore renewed is revoked and terminated.

It is Further Ordered that the date of issuance and service of this Intermediate Decision is January 12, 1960; that exceptions, if any, and briefs in support of such exceptions hereto may be filed on or before February 1, 1960; briefs, if any, in opposition to such exceptions may be filed on or before February 3, 1960. In the absence of any exceptions being filed, or in the absence of any review by the Commission on its own motion, this Intermediate Decision shall become final on February 2, 1960.

Samuel W. Jensch,
Presiding Officer.

DOCKET NO. 50-146

IN THE MATTER OF SAXTON NUCLEAR EXPERIMENTAL CORPORATION

Issued January 21, 1960

APPEARANCES:

George F. Trowbridge, Esq., of Marks & Trowbridge, for Applicant.
Maurice Axelrad, Esq., for Staff of Atomic Energy Commission.

INTERMEDIATE DECISION

Saxton Nuclear Experimental Corporation (Saxton) organized as a non-profit corporation under the laws of the Commonwealth of Pennsylvania, filed an application on July 24, 1959 pursuant to the provisions of Section 104(b) of the Atomic Energy Act of 1954, as amended, for a construction permit to authorize the building of a nuclear power reactor. The application requests authority for both a construction permit and an operating license, but this proceeding is limited to the initial undertaking involving the construction permit. The issuance of a construction permit depends only upon a finding by the Commission that a facility of the general type proposed can be constructed and operated at the specified location without undue risk to the health and safety of the public. The construction permit does not constitute final approval of any technical specification of the facility. This limitation of the proceeding to the construction permit contemplates that a subsequent proceeding will be convened to consider the issuance of an operating license.

The nuclear power reactor sought to be constructed by Saxton is a closed cycle pressurized water reactor with a nominal rating of 20 MW (thermal) which is proposed to be connected to an existing turbine generator at the Saxton, Pa., steam generating station of the Pennsylvania Electric Company. It is proposed that the reactor will primarily be utilized (1) for experimental purposes, specifically, for the demonstration of the capability and life of various cores that will be used in the reactor; (2) for obtaining data on individual test fuel assemblies, as well as entire reactor cores, that will improve the overall economics of large-scale commercial size reactor cores; (3) for determining optimum primary coolant operating conditions, including the possibility of some bulk boiling in the reactor core; and (4) for obtaining information during the latter stages of the research and development program that will be helpful in designing a nuclear superheat reactor, or a reactor containing a superheat section.

Pursuant to the notice published by the Commission, a hearing was held in Germantown, Md., on December 15, 1959. Appearances were entered on behalf of Saxton and the Staff of the Commission.

No other persons expressed a desire to intervene or to otherwise participate in the proceedings. The issues specified by the Commission for consideration were:

(1) Whether there is information sufficient to provide reasonable assurance that a utilization facility of the type proposed in the application can be constructed and operated at the location proposed therein, without endangering the health and safety of the public;

(2) Whether there is reasonable assurance that the technical information omitted from, and required to complete the application will be supplied.

(3) Whether Saxton is technically qualified to design and construct the proposed facility;

(4) Whether, pursuant to Section 50.40(b) of the AEC's Regulation, Saxton is financially qualified to design and construct the facility; and

(5) Whether the issuance of the construction permit will be inimical to the common defense and security or to the health and safety of the public.

Saxton is a subsidiary of Pennsylvania Electric Corporation, Metropolitan Edison Company, New Jersey Central Power & Light Company, and Jersey Power & Light Company, which four utilities are operating subsidiaries of General Public Utilities Corporation, a registered public utility holding company. Saxton has entered into a contract with Westinghouse Electric Corporation (Westinghouse) for construction[1] of the proposed reactor for the total fixed sum of $6,250,000, which figure is stated to be not subject to escalation.[2]

The four utilities owning Saxton have agreed, subject to approvals of applicable regulatory commissions, to supply the necessary funds for this construction, as well as provide an additional sum of approximately $2,250,000[3] for estimated costs of operation over a 5-year period. Westinghouse has contracted to (1) design, construct, and furnish to Saxton the complete nuclear steam generating plant, together with certain nuclear and nonnuclear supporting facilities; (2) furnish all fuel requirements for the first 5 years of operation after initial criticality; and (3) conduct a substantial preoperational and postconstruction research and development program over a period of 5 years, including the furnishing of certain equipment and personnel. Westinghouse has also agreed to assume financial responsibility for all special nuclear material needed during the period of operation as required by Commission regulations, and to pay directly to the Commission the charges in connection with such materials.

The research and development project to be undertaken by Saxton will involve operation of the reactor under utility operating conditions, with primary emphasis on lowering nuclear power costs through exploring the possibilities of increasing nuclear fuel use

[1] The estimated completion date is not earlier than December 1, 1961, nor later than December 1, 1962.

[2] If modifications to the presently proposed design are agreed upon between Saxton and Westinghouse, presumably, the four utility owners of Saxton will provide any needed additional funds.

[3] If this sum is not sufficient, the utilities owning Saxton have indicated that they will provide what additional is needed in order to carry out the objectives of the project. See *In the Matter of Power Reactor Development Co.*, Opinion and Final Decision, May 26, 1959, pp. 56-63.

efficiencies and economies and lowering future nuclear power plant capital costs. The program will include experimental operation of the reactor with different cores, with different flow conditions, with different arrangements and procedures, and with experiments involving transient conditions of operation within limited ranges. Final safety analysis of the experimental program must await detailed design of the reactor, development of detailed experimental programs, and information about the reactor to be gained in actual operation of the plant. Saxton and the staff agree, however, that there is no reason why a suitable experimental program cannot be developed and safely carried out with the Saxton reactor along the general lines presently proposed by Saxton. The detailed plans for experimental operation of the reactor will be reviewed and evaluated by the Commission in connection with the issuance of an operating license.

The proposed reactor is to be built on a 150-acre site owned by Pennsylvania Electric Company and located about three-fourths of a mile from the borough of Saxton in Liberty Township, Bedford County, Pa. The Borough of Saxton is approximately 95 air miles east of Pittsburgh and 72 air miles west of Harrisburg. The area surrounding the site is sparsely settled, with a population of approximately 18,000 persons within a 10-mile radius and a population of less than 1,800 within a 1-mile radius of the site.

The Saxton Steam Generating Station is located on the Raystown Branch of the Juniata River. This branch flows northeast and joins the Frankstown Branch of the Juniata River to form the main branch of the Juniata River, emptying into the Susequehanna River above Harrisburg, Pa. Public water supplies in this area within a 10-mile radius of the site are obtained primarily from springs and wells, none is derived from ground water in the valley in which the reactor is to be located. There is no drinking, commercial, or industrial use of water from the Raystown Branch of the Juniata River downstream from the site (approximately 5 miles) ; however, several miles below the site the river becomes a recreational area. The only known use of drinking water from the main branch of the Juniata, downstream from Saxton, is an emergency intake at the Borough of Newport, 60 miles east of Saxton. This water is filtered before entering the Borough mains.

Information concerning the hydrology, topography, and geography of the area, together with preliminary information on meteorology obtained at the site, indicate that the site is favorable from a safety standpoint. A continuing program of weather observations and meteorological investigations will be conducted by Saxton. In addition, Saxton in the near future will initiate a program of radiation monitoring in the vicinity of the site, including sampling of air, water, soil, vegetation, and animal life.

As stated, the Saxton reactor is of the pressurized water, more particularly of the thermal neutron, heterogeneous type employing slightly enriched uranium dioxide as the nuclear fuel, moderated and cooled by light water. The steam generated in the steam generator is directed to an existing turbine generator plant at the site. The reactor is designed to develop twenty thermal megawatts of power. To achieve flexibility in the conduct of tests which may be

run with the reactor facility, the steam generator is designed to extract a maximum of 28 megawatts of heat from the coolant. The Saxton reactor is basically similar to the Shippingport Pressurized Water Reactor (PWR) and the Army Packaged Power Reactor (APPR), from which an extensive amount of data and successful operating experience have been accumulated. Experiments have been performed with the designated Borax and Spert reactors, which have characteristics essentially the same as the Saxton reactor, to investigate the characteristics and demonstrate the stability of pressurized water reactors. The results of these experiments indicate that this type of reactor, because of its nuclear thermal and hydrodynamic characteristics, is inherently stable. The Saxton reactor is also substantially similar, although smaller in size and power output, to the Yankee Atomic Electric Company reactor, at Rowe, Mass., and the Belgian thermal reactor, being constructed at Mol, Belgium.

Conceptual design and design criteria for the major components of the facility have been developed in considerable detail in Saxton's application and in testimony at the hearings. Although detailed engineering design of the Saxton reactor is not complete, the information provided, as partly summarized hereafter, supports the conclusion that there is reasonable assurance that the Saxton reactor can be satisfactorily designed and safely operated. After all engineering details of the plant have been settled, Saxton will submit additional information to the Commission as an amendment to its application for construction permit in the form of a final hazards summary report, which information will be considered at the hearing on the operating license. Saxton expects to submit this information approximately June 1, 1961.

It is contemplated that the initial core will consist of 32 fuel assemblies arranged in a grid pattern forming an approximately cylindrical active core 40.2 inches in height and 33.6 inches in diameter. The nuclear fuel is uranium dioxide, enriched in the uranium-235 isotope to approximately 3–4 percent, compacted and sintered in the form of solid cylindrical pellets. The core is similar in design to the cores of a number of other reactors, particularly the Yankee reactor.

The reactor will be contained within a vertical cylindrical carbon steel pressure vessel approximately 17 feet in overall height, with an internal diameter of 58 inches and a shell thickness of 4.5 inches. The vessel is designed and will be constructed in accordance with the ASME Boiler and Pressure Vessel Code. The design pressure is 2,500 psia. The operating pressure of the primary coolant system is not expected to exceed 2,000 psia, and relief valves are provided to limit system pressure transients to less than the design pressure of 2,500 psia.

The reactor control system, as designed, is comprised of control rods and a nuclear poison dissolved in the primary coolant. The control rods, six in number, are of a cruciform shape containing a cadmium-indium-silver poison section clad with stainless steel. The control rod system design is similar to that used on a number of reactors. In particular, the rod material and drive mechanisms are similar to those of the Yankee reactor. The control rods possess

reactivity values sufficient to shut down the reactor in the hot clean condition by 2 to 3 percent in excess reactivity. A chemical neutron poison material, boric acid, will be added to the primary coolant to maintain the reactor in a subcritical condition when it is cold. The use of liquid poison for reactor control is being extensively studied by Westinghouse with respect to this reactor and the Yankee and Belgian reactors. The results of these studies to date indicate that this method of control will be feasible. Since the Yankee and Belgian reactors will be completed before the Saxton reactor, operating information on this method of control is expected to be available for further review before the Saxton reactor is completed.

Standard instrumentation is used to measure process parameters throughout the plant and neutron levels of the reactor. The usual protective systems and interlocks for this type of reactor are also provided. The design of the instrumentation system has not been completed. However, there are no unusual features proposed, and no unusual problems are anticipated.

A safety injection system is provided to supply boronated water to the reactor vessel for cooling the core in the event normal core coolant is lost. This water can be pumped directly into the main coolant loop from the refueling water storage tanks. These tanks have sufficient capacity to provide cooling water to the core for one hour. After this time, the water which has leaked from the primary system and collected in the lower part of the containment vessel can be recirculated by sump pumps through the storage well heat exchanger and back to the core for continued cooling.

A steel containment shell, partially lined with concrete, will enclose the reactor and high pressure primary system. The containment vessel is designed as a pressure vessel in accordance with the ASME Code to contain—in the event of a pipe rupture—the flashing of all the primary coolant, including any water or steam that might be associated with experimental operation of the reactor, to an equilibrium mixture of water and steam. The design pressure is conservatively chosen, and the integrity of the vessel would be expected to be preserved in the event of a rupture of the primary system. The design leakage rate of the containment vessel is 0.2 percent of the contained volume in 24 hours at the design pressure of the vessel. Experience with other vessels of this type indicates that this leakage rate can be attained.

Waste disposal facilities will be provided to receive and process all liquid and solid wastes from the plant. Gaseous wastes will be collected and processed prior to being dispersed through the stack under monitored, controlled conditions. There are no unusual features proposed or anticipated in the waste disposal system, and there is reasonable assurance that a satisfactory design of the waste disposal system can be developed to monitor, control, and process plant wastes and permit discharges to the environment within permissible limits.

The major accidents that can occur to the Saxton reactor have been analyzed, including accidents resulting from Acts of God, fire, human errors, and equipment failures. For most of these accidents, little or no damage would result. However, for certain types of

conceivable, but highly improbable, accidents, quantities of fission products in the reactor could be released into the containment vessel. Applicant's accident analysis indicates that the most severe accident of this type would result from the loss of coolant due to a pipe rupture accompanied by simultaneous failure of the safety injection system. Results of such an accident have been analyzed by the applicant based on assumptions representing a realistic upper limit of the quantities of fission products which might be released to the containment vessel and based on conservative calculations as to the radiation dosages to which persons in the vicinity of the site might be subjected.

Assuming that the containment vessel leaks at its design rate of 0.2 percent of the contained volume in 24 hours, that atmospheric dispersion conditions are the worst that might prevail, i.e., a severe temperature inversion exists, and that exposed persons are unprotected and stand on the site boundary—approximately 1,000 feet—at the cloud centerline for 8 hours following the assumed accident, the radiological doses in roentgen or rem to these individuals are calculated to be:

Direct radiation to whole body:	
From containment	6 roentgen
From leakage	0.8 roentgen
Lifetime dose due to inhalation of nuclides affecting the following organs:	
Bone (Sr^{89}–Sr^{90})	8.5 rem
Thyroid (iodine)	1,500 rem

Because of the assumptions used by the applicant in its accident and hazards analysis, it is likely that the actual consequences of such an accident would be less than those postulated. In the unlikely event that an accident of the magnitude described by applicant should occur, it might become necessary to evacuate a small number of persons in the immediate vicinity of the site. Evacuation times are sufficiently long to permit an orderly evacuation of the threatened area, and plans are being developed by Saxton to prepare for this possibility.

Considering all of the factors involved in the analysis of this hypothetical accident, including the conservatism employed and the unlikelihood of such an accident occurring, the possible dosages described above represent an acceptable degree of potential hazard in public areas.

On September 14, 1959, the Advisory Committee on Reactor Safeguards reported to the Commission on the Saxton project, in part, as follows:

> "Because of the favorable meteorology, topography, geology, hydrology and low population density, it is the Committee's opinion there is reasonable assurance that a reactor of this general type can be constructed and operated at this site without undue hazard to the health and safety of the public. A preliminary presentation of the planned research program was made. Evaluation of this program must be deferred pending more detailed studies of all phases of design and operation."

Westinghouse has had an extended history in the nuclear energy and electric power field, including the design, testing, and operation of nuclear reactors at Arco and Shippingport, and of submarine

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reactors. Westinghouse has also been responsible for the design, and associated research and development of the Yankee reactor, and the Belgian thermal reactor, which are essentially the same as the Saxton reactor.

In addition to the foregoing constituting findings and conclusions, and based upon the entire record in this proceeding including proposed findings and conclusions, contentions and statements of the participants, the Presiding Officer further finds as follows:

1. Saxton Nuclear Experimental Corporation, a duly organized nonprofit corporation, existing under and pursuant to the laws of the Commonwealth of Pennsylvania, is a qualified applicant to receive a construction permit provided by the Atomic Energy Act of 1954, as amended.

2. The nuclear facility proposed to be constructed by Saxton is a utilization facility within the meaning of the Atomic Energy Act involved in the conduct of research and development activities leading to a demonstration of the practical value of such facilities for industrial or commercial purposes.

3. Westinghouse Electric Corporation is technically qualified to design and construct the reactor, and accordingly, Saxton has provided for adequate technical qualifications to design and construct the proposed facility.

4. Sufficient information has been presented to provide reasonable assurance that a nuclear reactor of the general type described in the application and Preliminary Hazards Summary Report can be constructed and operated at the proposed location without endangering the health and safety of the public.

5. Saxton is financially qualified to design and construct the proposed facility.

6. There is reasonable assurance that technical information omitted from, and required to complete the application will be supplied.

7. The issuance of a construction permit to Saxton will not be inimical to the common defense and security or to the health and safety of the public.

WHEREFORE, IT IS ORDERED, subject to review by the Commission upon its own motion, or upon appeal, after exceptions, if any, are filed:

A. Saxton Nuclear Experimental Corporation is authorized to construct the nuclear facility described in its application and in accordance with the design set forth in the Preliminary Hazards Safeguards report filed with the Commission, and the Director of the Division of Licensing and Regulation of the Atomic Energy Commission is directed to issue to Saxton a provisional construction permit in the form attached to the Notice of Hearing issued by the Commission.

B. This authorization is provisional to the extent that an authorization to operate the facility will not be issued by the Commission unless a final hazards summary report is filed by Saxton and until the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility.

C. Exceptions, if any, and briefs in support thereof, must be filed by February 10, 1960; briefs in opposition thereto shall be filed February 12, 1960; and if the Commission makes no review on its own motion, and no exceptions are filed, this decision shall, in accordance with the Commission's Rules of Practice, become final on February 11, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

DOCKET NO. 50-29

IN THE MATTER OF YANKEE ATOMIC ELECTRIC COMPANY

Issued February 26, 1960

ORDER DENYING PETITION TO INTERVENE AND GRANTING LIMITED PARTICIPATION

On the 4th day of February 1960, a petition to intervene in the above entitled proceeding was filed by Elliott Earl, Manchester, Conn., for himself as a customer of the Hartford Electric Light Company, and as President and Managing Director of the Institute for Nuclear Serology of Manchester, Conn. The petition also alleged that if intervention by Elliott Earl were not permitted in the proceeding, then request was made for limited appearance in the proceeding to permit him to file a statement of position pursuant to Section 2.731 of the Commission's Rules of Practice.

In the petition to intervene, Elliott Earl alleged that the pressurized water reactor of Yankee Atomic Electric Company (Yankee), which is nearing completion at Rowe, Mass., will be utilized, in part, for the sale of power to the Hartford Electric Light Company, of which latter company Elliott Earl is a customer. The petition to intervene also alleged that Elliott Earl was President and Managing Director of the Institute for Nuclear Serology of Manchester, Conn., and that he has been following with precise care and study certain aspects of nuclear developmental problems and their management in the public interest, common defense, and security, and as specifically related to nuclear waste management.

On February 19, 1960, Yankee filed a notice of opposition to the request by Elliott Earl for permission to intervene on the general grounds that the interest of such petitioner would not be affected by the disposition of issues specified in the Notice of Hearing issued by the Atomic Energy Commission on January 27, 1960. Yankee moved in the alternative, however, to grant the request by Elliott Earl for limited appearance in the proceeding. The Staff of the Commission filed an answer to the petition substantially similar to that filed by Yankee. In addition the Staff made a motion to strike certain material attached to the request or petition to intervene on the ground that it is irrelevant, immaterial and constitutes advertising matter. The motion is denied for the reason that the attached material is related to the presentation of qualifications submitted by Elliott Earl, and as hereinafter determined, is insufficient foundation for the petition.

In reference to the allegations in the petition to intervene that Elliott Earl is a customer of the Hartford Electric Light Company

it has not been established that a customer of a customer[1] is automatically entitled to participate in proceedings involving the initial customer, and there has been no showing that the interests of Elliott Earl, in reference to the service from or the rates of, Hartford Electric Light Company are not already adequately protected by the regulatory authorities having jurisdiction over the customer companies involved. Elliott Earl has not alleged that his service from or the rates of the Hartford Electric Light Company would be adversely affected by any action proposed to be undertaken by Yankee Atomic Electric Company.

Elliott Earl, however, has shown himself to be concerned with nuclear power considerations, and thus is one among many persons who are similarly situated. That concern, however, does not appear to be sufficiently substantial, nor supported by any special technical competence, and thus the decision to be rendered by the Commission in this proceeding will not affect in any material way the interest of Elliott Earl in this proceeding. It may be, however, that in view of the general interest had by Elliott Earl in nuclear power matters, a statement of position filed by him would add matters entitled to consideration in this proceeding, and accordingly provision is here made for him to submit his statement either at the outset of the hearing, or at the conclusion thereof.

The Presiding Officer *finds*:

1. Good cause has not been established by Elliott Earl to permit his intervention in this proceeding either for himself, as a customer of the Hartford Electric Light Company, or as President and Managing Director of the Institute for Nuclear Serology.

2. It may be that a statement of position filed by Elliott Earl in this proceeding may contribute matters entitled to consideratoin respecting the issues involved in this proceeding.

The Presiding Officer *orders*:

A. The request or petition to intervene in this proceeding filed by Elliott Earl on the 4th day of February 1960, is denied.

B. The request filed by Elliott Earl for a limited appearance in this proceeding in accordance with Rule 2.731 of the Commission's Rules of Practice is granted.

SAMUEL W. JENSCH,
Presiding Officer.

[1] The Yankee Atomic Power Company project contemplates the sale of power to its sponsoring companies, of which Hartford Electric Light Company is one, in proportion to the latter's stock interest, which is 9.5 percent.

BYPRODUCT MATERIAL LICENSE NO. 4-616-?

IN THE MATTER OF X-RAY ENGINEERING COMPANY

Issued February 29, 1960

ORDER GRANTING REQUEST FOR AMENDMENT OF LICENSE BY SUBSTITUTION OF EQUIPMENT, AND DENYING REQUEST FOR AMENDMENT OF ORDER WHICH SUSPENDED CERTAIN LICENSED OPERATIONS

On February 8, 1960, X-Ray Engineering Company, with offices at Burlingame, Calif., filed a joint request, first for an amendment of its Byproduct Material License No. 4-616-3, to permit the substitution of the "Gamma Ray Projector" manufactured by Technical Operations, Inc., Model No. 489, Style 470, wherever the existing license, as amended, refers to Isotope Specialties Company, Inc., cameras Model No. LSS-2D, or LSS-3A. In support of this portion of the motion, X-Ray represented that the cameras issued or manufactured by Isotope Specialties Company, Inc., have not proven satisfactory when used in the field, and that the projector manufactured by Technical Operations, Inc., is satisfactory. X-Ray further represented that it does not believe that it is safe to continue to use any remote control camera as that manufactured by Isotope Specialties, Inc., in view of the great possibility of danger in its use.

On February 11, 1960, the Staff filed an answer to this part of the motion for amendment of license and indicated that it would not oppose this substitution of equipment, since the Staff stated the motion related merely to equipment to perform work already authorized, and conferred no new privilege upon X-Ray Engineering Company. The Staff further stated that it understood that it had been informed that Style 470 of the gamma ray projector manufactured by Technical Operations, Inc. was no longer available, and had been replaced by that manufactured as Model A-424-1. In view of this change, the Staff suggested that X-Ray be permitted to use either model of the 489 series of cameras.

By the motion filed on February 6, 1960, X-Ray also requested an amendment of the order issued by the Commission on October 16, 1954, as further amended by Commission Order dated October 28, 1959 to now permit X-Ray to render radiographic service in the State of Hawaii in order to accommodate a single transaction under a contract between X-Ray and one of its largest customers. In the motion, X-Ray prescribed specific conditions which, in its opinion, would exact compliance by X-Ray with all of the rules and regulations of the Commission in reference to the conduct of industrial inspection (radiographic services) using atomic byproduct materials. The Staff of the Commission, in its answer, opposed the granting of X-Ray's motion, and stated that for the reasons given at the hearing which had been convened to consider alleged violations by X-Ray

of its license, the Staff urged that the public interest requires termination of the X-Ray Engineering Company license.

On February 19, 1960, the Staff filed its Proposed Findings and Conclusions and brief in support thereof recommending cancellation of X-Ray's license for alleged violations which are described as wilful.

It appears that the operations by X-Ray under its license have been considered by the Commission in reference to the violations alleged in Commission orders. At the present time, the operations by X-Ray are limited to the State of California where the major portion of X-Ray's activities are conducted. In support of the motion to enlarge operations to include the State of Hawaii, X-Ray has represented that it desires to accommodate one of its largest customers, and that a substantial volume of X-Ray's business depends upon this customer.

The Presiding Officer finds:

1. The Byproduct Material License No. 4-616-3 issued by the Commission to X-Ray Engineering Company should be amended to permit the use by X-Ray of either model of 489 series of Gamma Ray Projector manufactured by Technical Operations, Inc. in substitution for models No. LSS-2D or LSS-3A cameras issued by Isotopes Specialties Company, Inc.

2. Adequate basis has not been established by X-Ray for modification of the Commission's Order of October 16, 1959, as amended on October 28, 1959, which limited X-Ray's operations to the State of California.

3. In view of the nature and extent of the alleged violations set forth by the Commission, the pendency of the consideration thereof, and the answers respecting thereto, the existing order of the Commission dated October 16, 1959, as amended on October 28, 1959, respecting the scope of X-Ray's operations should not be modified or changed.

The Presiding Officer *orders*:

A. The Byproduct Material License No. 4-616-3 issued to X-Ray Engineering Company is amended to permit the use by X-Ray of either model of 489 series, Gamma Ray Projector manufactured by Technical Operations, Inc. in substitution for models No. LSS-2D or LSS-3A cameras issued by Isotopes Specialties Company, Inc.

B. The motion by X-Ray Engineering Company for modification of the Commission Orders issued October 16, 1959, as amended, by Order of October 28, 1959, is denied.

SAMUEL W. JENSCH,
Presiding Officer.

DOCKET NO. 50-155

IN THE MATTER OF CONSUMERS POWER COMPANY

Issued March 28, 1960

ORDER FOR INTERVENTION AND LIMITED PARTICIPATION

On March 25, 1960, the State of Michigan, by Paul Adams, its Attorney General and Jerome Maslowski, Assistant Attorney General, filed a petition to intervene in the above entitled matter and alleged that the State of Michigan, under its police power, is charged with the duty of protecting the health and welfare of the people of Michigan, and that its Attorney General should be permitted to intervene herein in order to assist in performing these services. There is no allegation in the petition that either the construction or operation of the proposed nuclear reactor will be harmful to the health or detrimental to the interests of the people of the State. The petition requests participation in the proceeding to the extent the State believes it necessary to be fully informed of all aspects of the proposed construction and thus perform its obligation to protect the health and interests of the people of the State.

On March 25, 1960, the Water Resources Commission of the State of Michigan filed a petition seeking a limited participation in the proceeding for the purpose of presenting its statement concerning certain specified waste disposal problems. The petition further alleged that information was needed from Consumers Power Company for consideration before a recommendation could be made that a satisfactory solution had been developed for these waste disposal problems.

The Presiding Officer finds:

1. The filing of the foregoing petitions on March 25, 1960 does not permit an interval of reasonable time to exist before the date of the hearing of March 29, prescribed by the Commission, within which the Applicant or the Staff may express views concerning those petitions.

2. The State of Michigan, by its Attorneys General and the Water Resources Commission of the State of Michigan are entitled prima facie, as representatives of public organizations created by constitution and statute, to participate in this proceeding.

The Presiding Officer *orders*:

A. The State of Michigan, by its Attorney General, Paul Adams, and Jerome Maslowski, Assistant Attorney General, is permitted to intervene and become a formal intervener in this proceeding, and the Water Resources Commission of the State of Michigan is permitted to enter a limited appearance in this proceeding, both permissions in accordance with the petitions heretofore filed.

B. Consumers Power Company and the Staff may express their views or objections on the date this proceeding convenes, to the intervention by the State of Michigan, as heretofore provided, and to the limited participation by the Water Resources Commission, and if objections are presented, reconsideration will be given to this Order for the aforesaid intervention and limited participation.

SAMUEL W. JENSCH,
Presiding Officer.

BYPRODUCT MATERIAL LICENSE NO. 4-616-3

IN THE MATTER OF X-RAY ENGINEERING COMPANY

Issued April 1, 1960

ORDER GRANTING RECEIPT INTO EVIDENCE OF ADDITIONAL EXHIBITS AND GRANTING EXTENSION OF TIME FOR FILING PROPOSED FINDINGS

On March 21, 1960, X-Ray Engineering Company, Burlingame, Calif., filed a Motion verified on March 18, 1960, to amplify the record to include additional exhibits and for an extension of time, and until April 4, 1960 within which to submit proposed findings and conclusions as provided at the hearings held in this matter on December 2, 3, and 4, 1959. On March 30, 1960, the Staff filed its answer to this motion stating no objection to the receipt of the additional exhibits, but opposing the extension of time for submittal of the proposed findings and conclusions.

In its Motion, X-Ray stated that both it and the Staff had, since the conclusion of the hearings, reviewed a considerable volume of documents and correspondence which had occurred in the years 1957 and 1958, and as a result, X-Ray desired to have added to the record certain letters and documents which reflected the preparation of specific safety manuals which may be effective for only limited designated periods of time, or, as to one proposed manual, X-Ray contends that it never became effective. While the Staff does not concede the interpretations X-Ray places upon the correspondence, the Staff does not object to the admission of the specified portions thereof, and the documents designated. The Staff further states that all of the correspondence during this period might be admitted into the record, but the Staff adds that in its opinion it would not be prejudicial to its position to exclude the balance of the correspondence not designated by X-Ray.

The Staff objects to the 17-day extension of time for submittal of proposed findings and conclusions on the ground that X-Ray has already had ample time for this preparation. X-Ray alleged that most of its preparation time has been utilized in the search for additional documents and correspondence needed in order to adequately present its position in this proceeding.

The Presiding Officer *finds*:

1. Good cause has been shown for the admission into evidence of the additional correspondence and documents specified in the motion filed by X-Ray on March 21, 1960, and further, good cause has been shown for an extension of time until April 4, 1960 for the filing by X-Ray of its proposed findings and conclusions based upon the record herein.

The Presiding Officer *orders*:

A. The correspondence and documents designated as Exhibits A through M, inclusive, as described in the Motion filed by X-Ray Engineering Company on March 21, 1960, are received in evidence in this proceeding.

B. X-Ray Engineering Company may file its proposed findings and conclusions on or before April 4, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

DOCKET NO. 50-144

IN THE MATTER OF CAROLINAS VIRGINIA NUCLEAR POWER ASSOCIATES, INC.

Issued April 12, 1960

APPEARANCES:

George D. Gibson, Esq., and *George C. Freeman, Jr., Esq.*, Hunton, Williams, Gay, Powell & Gibson, for Applicant—Carolinas Virginia Nuclear Power Associates, Inc.

Maurice Axelrad, Esq., for the Staff of Atomic Energy Commission.

INTERMEDIATE DECISION

Carolinas Virginia Nuclear Power Associates, Inc. (CVNPA) a non-profit corporation organized under the laws of North Carolina, with principal offices at Charlotte, filed an application on July 9, 1959, with the Commission seeking a construction permit and license to construct and operate a utilization facility pursuant to the provisions of Section 104(b) and 185 of the Atomic Energy Act, as amended. The application by CVNPA was filed in accordance with its contract executed in January 1959 pursuant to the Power Demonstration Reactor Program [1] which is undertaken by the Commission in aid of the development of various types of nuclear reactor facilities which would be feasible for the generation and distribution of electricity. In addition to the construction permit and operating license, CVNPA requests a special nuclear material license and by-product material license for the contemplated operation. The Commission, on January 14, 1960, issued its Notice providing for a hearing which was held on February 23, 1960. This notice was published in the FEDERAL REGISTER (Vol. 25, F.R. Page 522) as a part of the public notice given of the proceeding. The issues specified for hearing were:

1. Whether there is sufficient information to provide reasonable assurance that a utilization facility of the general type proposed in the application can be constructed and operated at the location specified therein without endangering the health and safety of the public;

2. Whether there is reasonable assurance that the technical information omitted from and required to complete the application will be supplied;

[1] CVNPA is sponsored by four established electric utility companies: Carolina Power & Light Company, Duke Power Company, South Carolina Electric & Gas Company, and Virginia Electric and Power Co. None of the foregoing utilities nor CVNPA is subject to foreign control. The Power Demonstration Reactor program provides in this instance for the reimbursement to CVNPA for its research and development expenses up to approximately $15,000,000. Any excess research and development expenses, and all other construction costs and all other contractual indebtedness of CVNPA are to be paid by it and are severally guaranteed by the sponsoring companies.

3. Whether the applicant is technically qualified to design and construct the proposed facility;

4. Whether the applicant is financially qualified to engage in the proposed activities in accordance with Commission regulations;

5. Whether the issuance of a construction permit will be inimical to the common defense and security or to the health and safety of the public.

The application by CVNPA, as amended,[2] describes its project as essentially a thermal neutron, heterogeneous reactor, utilizing vertical pressure tubes, and fueled with slightly enriched uranium dioxide clad in zircaloy. The reactor would employ heavy water as coolant and moderator. This nuclear facility is designed to produce approximately 17 net megawatts of electric power in the conventional steam electric generating plant located adjacent to the proposed facility. The thermal power capacity of the facility is approximately 60 megawatts. While the proposed reactor is similar in many ways to a pressurized water reactor of proven design, its use of heavy water as a coolant and moderator and of pressure tubes make it a novel type of reactor.

The nuclear plant will be built on property leased from South Carolina Electric & Gas Company, near Parr, S.C., and adjacent to that company's existing hydro and steam electric generating facilities located there. The steam produced by the nuclear plant will be purchased by South Carolina Electric & Gas Company for use in the steam electric generating plant to generate electricity.

The Community of Parr is in a remote, sparsely populated section of South Carolina. Population density within the 5-mile radius is 15 persons per square mile. The nearest city of any size is Columbia, which is 25 miles to the southeast. CVNPA will prohibit all residence within an area with a minimum radius of one-half mile from the nuclear plant and this area constitutes what has been described as the exclusion area for the reactor project. CVNPA will promulgate safety regulations applicable to all persons within this area, all as permitted by its lease agreement with South Carolina Electric & Gas Company. There are no through roads within the exclusion area. The railroad traffic on the single track passing through it and automobile traffic on the nearby secondary highway could be controlled in an emergency. There is no boat traffic and no recreational activity on the stretch of the Broad River adjacent to the site.

The nuclear plant will be located near the Broad River. It will be built on a grade 63 feet in elevation above the normal level of the adjacent hydro pond, and thus free from the danger of floods. It will be separated from the river by the hill crest and any surface run-off from it could be initially away from the river. The permeability of the subsoil at the site is extremely low and would offer more than average resistance to subterranean infiltration of surface liquids, consequently there is almost no chance that any radioactive liquids which might escape during a severe accident would seep into

[2] Six amendments were filed by CVNPA on July 17, 1959, August 6 and 24, 1959, October 23, 1959, December 16, 1959, and January 5, 1960. The record in this proceeding included the application and all amendments with the exception of two groups of several pages, each from Amendment No. 4 which were classified as Restricted Data. The first group of pages was superseded by Amendment No. 5, and the second group of pages was still a part of the application, but CVNPA introduced an unclassified version at the hearing.

the river. The activity level of any light water liquid wastes emptied into the river will be monitored and will be below the levels specified in Title 10, Code of Federal Regulations, Part 20, including those listed in the amendments proposed thereto by the Commission on May 27, 1959 (24 F.R. 3537). The possibility of accidental contamination of the river is very remote, and, even if it should occur, it can be diluted by discharges from Parr Dam of sufficient magnitude to protect the public. The City of Columbia, the first user of river water downstream, is 25 miles away.

Information concerning the geology, hydrology, seismology and topography of the area together with preliminary information on meteorology indicates that the site is excellent from a safety standpoint. CVNPA has set up many instruments for various meteorological observations at the site and is engaged in a continuing program of observation and analysis. In this, CVNPA has the assistance of expert meteorologists of Lockheed Aircraft Corporation. In addition, CVNPA has retained Nuclear Science and Engineering Corporation to assist it in the study CVNPA is conducting of the natural radioactivity of the area with the cooperation of the South Carolina Water Pollution Control Authority.

As stated, the reactor may be considered as a heavy water cooled and moderated, pressure tube type reactor. This type is a variation of, but possesses some similarities to the pressurized light water type of reactor such as the Shippingport, Pa., and the submarine reactors. The general engineering technology of such pressurized light water reactors is well developed, and their operating characteristics are well known. A large part of this engineering technology and experience is directly applicable to the reactor proposed to be constructed by CVNPA. This technology is particularly helpful in relation to the pumps, valves, fittings, and other parts of the pressurized systems of the proposed reactor.

Although there are no heavy water moderated, pressure tube type reactors in operation at present, several are under construction in the United States and Canada. In particular, the CVNPA reactor is similar in power level, and in a number of design concepts to the Plutonium Recycle Test Reactor which is now nearing completion at Hanford, Washington, and which should be in operation several years before the scheduled completion of the CVNPA nuclear plant. The experience of these projects relating to the use of heavy water and of zircaloy pressure tubes should be of assistance in the construction and operation of this nuclear plant.

The reactor fuel will consist of slightly enriched pellets of uranium dioxide. These pellets will be inserted into zircaloy fuel rods, each 8 feet long and one-half inch in outside diameter. Nineteen of these fuel rods will be closely arranged to form a fuel element. Each fuel element will be positioned vertically within a leg of zircaloy U-tube. The reactor will contain 42 U-tubes in all. The U-tubes will contain heavy water as a coolant, which will be under a normal pressure of 1,500 psi.

At rated capacity operation the heavy water will enter the U-tube at 505° F. After leaving the U-tube[3] it will pass to a boiler where

[3] At 555° F.

its heat energy will be transferred to light water, converting the light water to steam. The steam will pass through an oil-fired superheater to the adjacent steam electric generating plant, where it will be used to generate electricity. The coolant, meanwhile, will return to the U-tubes, beginning its cycle anew. A series of thermal baffles will reduce the temperature of the U-tube walls to less than 300° F. during normal operation.

The reactor also will be moderated with heavy water, which will be contained in a single vertical tank 10 feet 3 inches in diameter and 11 feet in height. The lower two-thirds of the U-tubes will be immersed in the moderator. The moderator will be at atmospheric pressure. During normal operation the average temperature of the moderator will be 150° F. but some boiling of the moderator immediately adjacent to the tube walls may occur. The effects of this boiling on reactor reactivity and kinetics have not been quantitatively determined. If investigations planned for the research and development program indicate that this boiling would be detrimental to a safe level of reactivity and to the kinetics, suitable adjustments can be made to eliminate the boiling.

The reactor will be controlled and shut down by the insertion of 32 control rods. Each rod will be individually driven at exact speeds to be determined during the research and development program. They will be arranged for gravity "scram." The reactor will be designed to shut down automatically if certain abnormal conditions occur, including a high neutron flux in the reactor, rapid rise in reactor power, low main pressure coolant system pressure, low main primary coolant system flow, and electric power failure. A special shut-down cooling system will help remove decay heat from the pressure tubes and cool the primary coolant system, after the reactor has been shut down.

The reactor possesses a fast acting negative doppler coefficient. This coefficient together with the other characteristics of the reactor shows that it will possess adequate inherent stability against accidental insertions of reactivity of any magnitude likely to occur. It is expected that the pressure control system will maintain the required primary coolant system pressure during steady-state operation, limit pressure fluctuations caused by thermal expansion and contraction of the primary coolant during power plant load transients, and prevent the pressure in the primary coolant system from exceeding the maximum allowable pressure.

Two independent systems will eliminate or reduce potential hazards from a loss of coolant accident. First, an emergency water injection system would supply a continuous flow of light water to the pressure tubes, which should prevent fuel melting. Second, the vapor container water spray system would condense any steam that might form and reduce the pressure, significantly reducing the escape of fission products to the atmosphere.

The applicant has studied a number of accident situations which might be encountered: for example, various means of introducing excess reactivity, malfunctions of various items of equipment, and rupture of the primary coolant system. The accidents discussed by CVNPA represented those which appear to have a credible possibility of occurring in this facility. In general, the likelihood of

these accidents is sufficiently low and the consequences sufficiently small that no undue hazards to the public would result from the operation of the reactor. The upper limit of potential hazard to the general public from accidents occurring in this facility is characterized by the "maximum credible accident" discussed in the application and testimony.

In the "maximum credible accident," CVNPA assumed that a primary coolant pipe was severed completely and abruptly, releasing the coolant rapidly to the containment building. It was also assumed that neither the emergency water injection system nor the water spray system functioned. Under these conditions it was assumed that the fuel would melt releasing fission products to the building, and, because of the concurrent steam pressure in the building, to some extent to the outside atmosphere.

Radiological doses were calculated for this accident on the basis of the further conservative assumptions that the containment building does not rupture but leaks at its design rate; that the leaked material is released at ground level; that a severe temperature inversion exists; and that exposed persons are unprotected and remain on the radioactive cloud centerline for the entire period of exposure. Doses in roentgen or rem for persons standing at a point on the site boundary nearest the reactor (½ mile) are:

	Exposure Time (hours)	
	1	*3*
Direct radiation to whole body:		
From containment sphere	0.1	0.2
From cloud	0.03	0.08
Lifetime dose due to inhalation of nuclides affecting the following organs:		
Bone	1.0	3.2
Thyroid	26	77

In considering these numerical results it should be recalled that the likelihood of an accident of this magnitude occurring is extremely low and that there are a number of conservative factors in the assumptions used in these calculations. The magnitude of the exposure doses calculated does serve, however, to indicate the upper limit of public hazard which could be expected to result from operation of this facility. The magnitude of doses calculated in this most severe case indicates that in some remote circumstances people who are near the site may have to be evacuated. The dose rates are sufficiently low to permit ample time for any required evacuation to be accomplished before excessive exposure doses would accumulate.

All the major water systems and all pressurized parts of the main coolant system will be located within a vapor container. The container will be a cylindrical reinforced concrete structure, lined with steel sheets. It will have a steel hemispherical dome covered on the outside and lined on the inside with concrete. The specified design pressure is 14 psig and the specified maximum leakage rate will be 0.1 percent of the contained volume per day. This design will assure that the container would not be breached in case of the "maximum credible accident." Personnel will not be admitted inside the container during reactor operation.

Waste disposal facilities will be provided to receive and process all liquid and solid wastes from the nuclear plant. Any gaseous

wastes will be stored for a suitable decay period in equipment that can retain up to 60 days' production. After being diluted with air, the decayed gases will be discharged to the atmosphere from a stack at activity levels in accordance with Commission safety regulations, being constantly monitored in the process. The discharge will occur only during suitable weather conditions. No unusual features are proposed or anticipated in any of the waste disposal systems or procedures. The methods proposed should afford proper protection for personnel within the exclusion area and the general public.

The zircaloy pressure tubes in the reactor require further study and investigation in order to complete the evaluation of the safety aspects of their use. Operating experience of PRTR and other reactors now being built and CVNPA's outlined research and development programs should supply the needed information on the effects of corrosion on zircaloy and the effects of irradiation on the structural integrity of the material. The reactor will be constructed so that the pressure tubes can be removed easily for periodic inspection and, if necessary, for replacement. Data now available indicate that zircaloy tubes can be used safely in the reactor with an expected lifetime of at least several years. In the research and development program that will continue during the first 5 years of operation of the nuclear plant, the U-tubes will be inspected periodically, to verify and to extend these data.

The final design of the reactor has not been completed; but criteria for its major components have been developed in considerable detail in the evidence produced in this proceeding. Present analyses and calculations from these preliminary design parameters indicate that no particularly unfavorable safety characteristics will arise from the nuclear, thermal, or hydraulic features of the reactor. The research and development program now being conducted by CVNPA, supplemented by the experience of similar reactors operating earlier than the reactor, affords reasonable assurance that the presently unresolved safety questions concerning its unproved features can be answered and thus assure that the reactor can be satisfactorily designed and safely operated. The final design criteria will evolve from the full-scale critical experiments which will begin in the spring of 1960. After all of the engineering details of the nuclear plant have been settled, CVNPA will submit additional information to the Commission as an amendment to its License Application in the form of a final hazards summary report.

As will be provided hereinafter, in view of the untested character of certain components, and new design problems involved in this nuclear reactor, conditions will be attached to this construction permit requiring proof by tests of the adequacy and safety of the pressure U-tubes, the specially designed use of heavy water and the zircaloy clad fuel elements before this construction permit may be converted to an operating license. This evidence will be additional to that presented as the basis for opinions that the completed project has been constructed in accordance with the specifications of this construction permit and will provide reasonable assurance that the health and safety of the public will not be endangered by its operation. The Commission's Chief of the Hazards Evaluation Branch of Division of Licensing and Regulation testified in part, as follows:

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"The difference . . . between this reactor and, let us call it, conventional pressurized water reactor, lies not so much in the use of light water in one case, or heavy water in the other, as on the particular design proposed here of containing the coolant in a high pressure, small volume system separate from the water moderator, in a separate lower pressure system."

The first issue specified for consideration in this proceeding is whether there is reasonable assurance that a utilization facility of the general type proposed can be constructed and operated at the proposed site without endangering the health and safety of the public. There appears to be some inconsistency between the sufficiency of evidence for requirements of a general type of pressurized water reactor facility and the recognition that CVNPA proposes to construct a pressurized water reactor with several novel features. A Staff witness stated that if the proposed pressure U-tubes, the special designed uses of heavy water, and the zircaloy cladding were removed from this project, of course, the reactor would not be what CVNPA proposed.[4]

Therefore, it appears required that since some novel and "first of its class" features are involved in the CVNPA project, the findings on the issue of health and safety are limited to characteristics of a standard pressurized water reactor, similar to Shippingport, etc. Since experience cannot be a guide as to these newer elements and design of the CVNPA project, the construction permit must be provisional, until tests can be performed to produce evidence concerning the novel design and new components of this reactor project. The record does establish that, by virtue of the research and development program to be undertaken, the technical information now absent from the application will be supplied and thus complete the application.

The estimated total cost of the project is approximately $43,000,000. The Commission, under its contract with CVNPA, has agreed to reimburse CVNPA for research and development costs up to a maximum of $13,725,000 and to waive use charges on heavy water and special nuclear material up to $1,350,000. CVNPA will pay all remaining costs, which are presently estimated at approximately $28,000,000. At present, CVNPA's total receipts are from contributions from the sponsoring companies. Although CVNPA will receive some operating revenue from the sale of steam to South Carolina Electric & Gas Company, the greater part of its receipts

[4] The witness further testified, respecting a conventional pressurized water reactor of a general type, and the CVNPA project, in part, as follows:

"I must admit it is difficult to put this (CVNPA) reactor into a class, into any class, of reactors which have thus far been operated. There have been no operating precedents for this type of reactor in this power range or level. There have been a number of loops and other experimental facilities on a small scale, but nothing in the way of a reactor of this type. As indicated here, it (CVNPA) can be characterized as a heavy water pressure tube variation of a general pressurized water type. But that is distinctly a variation, and it is not strictly a pressurized water type, in the sense that we have had those reactors in the past, let us say, as a Shippingport is or some others . . . I would call it a pressurized heavy water . . . if one eliminates those features . . . mentioned, then we don't have the Carolinas-Virginia type of reactor anymore. It is completely different—so, I am afraid actually what we are dealing with here is a new type of reactor. This is the first of its class, first of its type. There will be four or five others, at least five others now in construction which will be of this class . . . one does not necessarily have to define a type in terms of a previous type in order to make the judgment that it is safe enough to operate. That is, one can define a new type of reactor, if he is sufficiently confident in his judgment and the research, some of which has been indicated here, proceeds with satisfactory results, and a final evaluation can be made that this project can proceed safely. . ." (Parentheses added.)

will always be contributions from the sponsoring companies. The sponsoring companies have contracted with each other and CVNPA, for the benefit of each, the Commission and all contractual creditors, to guarantee severally in the following proportions the payment of all contractual indebtedness of CVNPA.

Carolina Power & Light Company	20%
Duke Power Company	34%
South Carolina Electric & Gas Company	12.5%
Virginia Electric and Power Company	33.5%

All requisite corporate and regulatory authority has been granted the sponsoring companies to enter into this contract. The resources and credit of the sponsoring companies are adequate to enable them to meet their obligations to CVNPA. CVNPA appears to have adequate insurance against all non-nuclear accidents, and at the appropriate time will take out all insurance against nuclear risks that the law and the Commission require.

CVNPA has contracted with Westinghouse Electric Corporation (Westinghouse) for the performance of the necessary research and development program. Utilizing the results of this program, Stone & Webster Engineering Corporation (Stone & Webster) and Westinghouse will collaborate on the detailed engineering design of the nuclear plant and will supervise its subsequent construction. Both Westinghouse and Stone & Webster have had extensive experience in these fields of the nuclear industry.

On December 14, 1959, the Advisory Committee on Reactor Safeguards reported to the Commission on the reactor, in part, as follows:

"The Committee notes that complete evaluation of the effect on safety of certain design details of the CVTR is not possible at this stage, but depends on the outcome of various development and testing programs described by the applicant. The question of Zircaloy embrittlement under irradiation, and its corrosion characteristics, for example, will be finally evaluated at a later stage, when studies now under way have been completed.

"The Committee concludes that a reactor of this general type with containment as proposed by the Associates for the Parr location may be constructed with reasonable assurance that it can be operated at the site selected without undue risk to the health and safety of the public."

The Committee further stated in its letter dated February 1, 1960:

"It has been brought to the attention of the Advisory Committee on Reactor Safeguards that a portion (the southernmost 46 acres) of Hampton Island is to be removed from the exclusion area of the Carolinas Virginia Tube Reactor in order to provide right of way for a bridge to be built by the State of South Carolina.

"Since the location of the road over which the bridge is to be built was known to the Committee at the time of issuance of its previous advice on this site (reference letter dated December 14, 1959), and moreover the exclusion radius of the CVTR is not affected, the Committee agrees that this change will not materially affect the health and safety of the public."

The letter from the ACRS of December 14, 1959, antedated Applicant's Amendments Nos. 5 and 6. However, these amendments were forwarded to the Committee prior to their letter of February 1, 1960. Witnesses for the Applicant and the Staff testified that these amendments did not make any changes that would decrease the safety of the facility. In view of that testimony, and the fact that these two amendments had actually been submitted to ACRS before their last communication, it appears that their statutory review has been rendered. At the instant hearing on February 26, 1960, provision was

made for further communications from the ACRS respecting these two amendments. None has been received and, in view of the Staff interpretation of the effect of these amendments, it is concluded that ACRS is of the same opinion and its prior opinion respecting this proposed reactor is unchanged.

In addition to the foregoing findings and conclusions, upon the basis of the record of evidence including exhibits, and the proposed findings and conclusions submitted by the participants herein, it is further found and concluded that:

1. Carolinas Virginia Nuclear Power Associates, Inc., a non-profit corporation with principal offices at Charlotte, N.C., is a qualified applicant to receive a construction permit provided by the Atomic Energy Act, as amended.

2. The nuclear reactor project proposed to be constructed by CVNPA is a utilization facility within the meaning of Section 104(b) of the Atomic Energy Act, as amended.

3. There is sufficient information to provide reasonable assurance that the utilization facility of the general type proposed in the application, as amended, by CVNPA can be constructed at the location specified therein without endangering and will provide adequate protection to the health and safety of the public.

4. There is reasonable assurance that the technical information omitted from and required to complete the application, including particularly that related to the special designed use of heavy water, the U-pressure tubes, and their use of zircaloy, will be supplied.

5. CVNPA, Westinghouse, and Stone & Webster, collectively, are technically qualified to design and construct the proposed utilization facility.

6. CVNPA is financially qualified to engage in the proposed activities in accordance with the Atomic Energy Act, as amended, and the rules and regulations of the Commission.

7. The issuance of the proposed construction permit to CVNPA, substantially in the form attached to the Notice of Hearing issued herein, as modified by the order hereinafter made, will not be inimical to the common defense and security or to the health and safety of the public.

Wherefore, It is Ordered, subject to review by the Commission upon its motion or upon appeal after exceptions, if any are filed:

A. The Director of the Division of Licensing and Regulation of the Atomic Energy Commission is directed to issue a provisional construction permit substantially in the form attached to the Notice of Hearing issued herein by the Commission, as modified by this Order.

B. This authorization is provisional subject to the condition that an authorization to operate the facility will not be issued by the Commission until evidence has been adduced by CVNPA concerning the tests performed of all designs and components of this proposed utilization facility, including, among others, specially designed use of heavy water, the U-pressure tubes, and the use of zircaloy cladding of the fuel elements, and a basis has thereby been established as to all components when operated as a unit, that there is adequate protection to the health and safety of the public as required by the Atomic Energy Act, as amended, and by the rules and regulations of the Commission.

C. Exceptions, if any, and briefs in support thereof must be filed by May 3, 1960; briefs in opposition thereto shall be filed on May 5, 1960, and if the Commission does not initiate a review on its own motion, and no exceptions are filed, this decision shall, in accordance with the Commission's Rules of Practice become final on May 4, 1960.

Samuel W. Jensch,
Presiding Officer.

DOCKET NO. 50–130

IN THE MATTER OF NORTHERN STATES POWER COMPANY

Issued April 21, 1960

APPEARANCES:

Donald E. Nelson, Esq., for Northern States Power Company.
Gerald Charnoff, Esq., for the Staff of Atomic Energy Commission.

INTERMEDIATE DECISION

Northern States Power Company (NSP), a corporation organized under the laws of the State of Minnesota, with principal offices at Minneapolis, filed an application on April 3, 1959, for a construction permit and operating license, in conformity with its contract with the Atomic Energy Commission, to construct a 203 megawatt (thermal) utilization facility to be used for research and development in accordance with Sections 104(b) and 185 of the Atomic Energy Act, as amended. The NSP contract was executed as a part of the Power Demonstration Reactor Program,[1] initiated by the Atomic Energy Commission in aid of the development of various types of nuclear reactor facilities which would be feasible for the generation and distribution of electricity. Associated with NSP in the construction of this proposed facility, which has been designated the Pathfinder plant, is a non-profit corporation, known as Central Utilities Atomic Power Associates, which is presently composed of nine midwestern electric utilities[2] in addition to NSP which will contribute a portion of the research and development costs to an extent of $3,650,000. NSP has executed a contract with Allis-Chalmers Manufacturing Company, Milwaukee, Wis., for it to design and construct the proposed facility which will have an electrical capacity of 62,000 MW.

The Pathfinder facility will be connected to the NSP generating plant located near Sioux Falls, S. Dak. The Pathfinder plant site is on the Big Sioux River, downstream from, and about 3.5 miles northeast of the City of Sioux Falls, S. Dak., a city of approximately 65,000 population. The plant site has an area of 1,170 acres and is wholly owned by NSP. The closest dwelling to the reactor will be 0.47 mile away. The present population within a 2-mile radius is

[1] The Power Demonstration Reactor Program provides, as of the date of the application, among other matters, that NSP will be reimbursed in connection with research and development costs in an amount not to exceed $8,310,000, and in addition, the Commission will assume costs related to this project, but concerning activities undertaken at certain Commission facilities. The use charges for source and special nuclear material will also be waived by the Commission for this project, which charges may not exceed $1,800,000.

[2] The original organization was composed of 10 utilities in addition to NSP; one, the total of assets of Mississippi Valley Public Service Company, has recently been acquired by NSP. The nine remaining companies in addition to NSP are: Central Electric and Gas Company, Interstate Power Company, Iowa Power & Light Company, Iowa Southern Utilities Company, Madison Gas and Electric Company, Northwestern Public Service Company, Otter Tail Power Company, St. Joseph Light & Power Company, and Wisconsin Public Service Corporation.

112 persons. Two railroads cross the site within a ½-mile radius of the plant and two public roads touch the site with an interstate highway projected north of the site at a distance of 0.6 mile from the plant.

The type of facility proposed to be constructed by NSP is a boiling water reactor, of a design similar to the Borax series of reactors, such as EBWR and VBWR. The feasible operation and safety of this general type of reactor have been demonstrated through an extensive period of successful operation. In power level, the Pathfinder facility is larger than the Elk River, Minn., reactor, but is smaller than that at Dresden, Ill. Three new features have been incorporated in the Pathfinder facility which introduce new design and operating concepts in boiling water reactors not heretofore undertaken; (1) the proposal that novel and relatively unproven fuel elements be utilized in the boiling region of the core; (2) the proposal that a superheater be incorporated as an integral part of the basic reactor core, and that novel fuel elements be utilized in this region, and (3) the proposal that variable rate of primary coolant recirculation be utilized for controlling the power level of the core.

On January 7, 1960, the Commission issued its Notice providing for a hearing which was held on February 15 and 16, 1960 (F.R. Vol. 25, Page 254). Appearances were entered on behalf of NSP and the Staff. No persons sought to intervene or to otherwise participate in the proceeding. After the conclusion of the hearing proposed findings and conclusions were filed by NSP on March 21, by the Staff on April 1, and a reply by NSP on April 4, 1960.

The Pathfinder plant is designed for a gross electrical power output of 66,000 KW. It will normally operate at full power. The net electrical capacity will be 62,000 KW. A direct-cycle, heterogeneous, boiling water reactor with an integral nucelar superheater will be used, operating at a thermal power level of 203 MW. Light water will be used as coolant and moderated. Slightly enriched uranium dioxide clad with aluminum alloy is planned as the boiler fuel. A zirconium alloy clad material is considered as an alternate to the aluminum cladding and can be used if the research and development is not successful with respect to the aluminum cladding. Highly enriched uranium dioxide dispersed in stainless steel cermet, clad in stainless steel will be used as fuel in the superheater. Reactor water is pumped through the boiler core, for control purposes and to permit a high power output. The steam conditions at the throttle are expected to be 525 psig and 825° F. with a flow of 619,000 lbs/hr at a power level of 66 MW.

The reactor core will be contained within a cylindrical pressure vessel 11 feet 6 inches in diameter and with an overall height of 31 feet 6 inches. The lower head will include a 4-foot diameter cylindrical water inlet section. The pressure vessel will be fabricated of 3-inch thick composite plate consisting of SA–212 Grade B carbon steel clad on the inside with type 304J, stainless steel one-quarter inch thick to provide adequate corrosion protection. The cladding will be integrally bonded to the carbon steel backing material by a nickel interface layer. The reactor core will approximate a right cylinder having a mean diameter of 74 inches and a height of 72 inches. It will consist of the central superheater core surrounded by the boiler core.

The boiler core will consist of 96 boiler fuel element assemblies. Slightly enriched uranium dioxide in the form of sintered cylindrical pellets will be used as fuel. In the presently planned design, the pellets will be hermetically sealed in X–8001 aluminum alloy tubes 18 inches long. The fuel tubes will be arranged in bundles. Each boiler fuel element assembly will consist of four bundles welded end-to-end and contained in a 5.218-inch square aluminum box.

The superheater core will consist of 428 tubular fuel elements. Each fuel element will be composed of two concentric fuel tubes, a central burnable poison pin, spacers, and a handling knob. The fuel tubes will be 7 feet long with an active fuel section of 6 feet. The fuel will be highly enriched uranium dioxide—stainless steel cermet tubes sandwiched in type 304L or 316 stainless steel cladding. The fuel tubes and poison pin will be welded together at the top with proper spacing of tubes and proper orificing to the annuli. The fuel element will be supported in a double-walled stainless steel support tube, which is welded to and supported on the bottom of the core by the superheater support plate. These support tubes also serve as thermal and physical barriers between the steam flowing downward through the annuli in the fuel elements and the water moderator circulating upward outside the tubes. The space between the two walls of the support tube is filled with stagnant steam during normal operation, and heat transfer from the superheated steam to the water moderator is thereby minimized.

Twenty control rods will be located in the core, sixteen in the boiler core and four in the superheater core. The rods will be fabricated from ¼-inch thick stainless steel containing 2 percent boron by weight. The cross section of each control rod will be in the form of a 10-inch cruciform. The rods will have extensions which will be latched to the control rod drives. The rods will be moved within control rod guide tubes, which are a part of the core structure. These guide tubes will have a hollow cruciform cross section and will be fabricated of Zircaloy-2 in the boiler region and stainless steel in the superheater region.

A central control station will be provided from which all control actuations required for normal power operation will be made. Local instruments will be provided to permit on-the-spot observation of the performance of the various components. During a normal startup, the turbine will be brought up to speed locally. When the turbine is up to speed, its control will be assumed by the central control room.

To insure safe operation of the plant, the control and instrumentation system will monitor various reactor and plant conditions, and will warn the operator, or scram the reactor when necessary, if certain limits of activity or power are exceeded. An automatic scram will be initiated if the conditions of operation, such as short reactor period or a power level about 115 percent of normal, threaten the safety of the plant or personnel. An alarm will be sounded whenever a condition that is only potentially dangerous exists, warning the operator to correct the condition. The temperature and void coefficients of reactivity of the Pathfinder reactor will be negative in value, which is consistent with those of previous boiling water reactors.

As stated, the proposed facility is a boiling water reactor having several new features and unique characteristics with which there has been no previous experience in any other reactor of a size comparable to that proposed by NSP. These are the use of aluminum alloy as a cladding for fuel in the boiler core, the use of cermet fuel of the type proposed to be used and the superheater core, and the variable rate of primary coolant recirculation proposed to be used for controlling the power level of the core.

The phenomena which determine the characteristic performance of a boiling water reactor with integral nuclear superheat are well recognized, but no reactor has thus far been operated with an integral built-in superheater. NSP has analyzed the nuclear, hydraulic, and thermal aspects of such a system in steady-state operation. It is currently engaged in studying the phenomena of this novel system which are of significance from the standpoint of safety. In principle, there appears to be no reason why a nuclear superheater could not be safely incorporated as a part of a nuclear core; however, its incorporation as an integral part of the core introduces problems which must be dealt with by further analysis and by an appropriate research and development program.

Current information indicates that cermet fuel of the type proposed to be used in the Pathfinder superheater probably can be used safely. However, the useful life of the fuel under Pathfinder conditions and the effects of radiation and high temperatures on the cermet fuel are not fully known. There also remains some doubt as to the consequences which might accompany or follow failures, including meltdowns of the superheater fuel elements.

The performance and capabilities of the superheater have been calculated. Verification or determination of important nuclear characteristics of both the superheater and boiler cores will be made in experiments in the Allis-Chalmers critical facility. These experiments, however, will not provide any data with regard to the effects of radiation and temperature on the superheater fuel. Should use of the nuclear superheater not be feasible for the first core loading, the alternative of operating without it remains. With zero superheat, approximately 63 percent rated generator output would be realized. At this point the moisture content is estimated at 16 to 18 percent which is undesirable for extended operation because it decreases the life expectancy of the exhaust end blades. The plant could be operated for a few years without a superheater loading by operating the boiler core at reduced power. The Pathfinder reactor can be redesigned, constructed, and operated safely as an ordinary water reactor without a nuclear superheater, should further data fail to support the present conclusion that operation of the reactor with the nuclear superheater can be undertaken without undue hazard to the public and the on-site employees.

In most boiling water reactors today the fuel elements have been clad with zircaloy-2 or stainless steel. For the Pathfinder plant, NSP proposes to use an element composed of uranium dioxide clad with an aluminum alloy. Such elements have been used only to a very limited extent, and more experience is required before such elements can be confidently accepted for routine operation in power reactors. NSP has proposed a research and development program to be car-

ried out on these elements, including testing of prototype elements in a reactor environment similar to that expected in the power reactor, before initial operation of the Pathfinder plant. Such a research and development program is necessary, and the proposed program appears to be reasonably calculated to establish the safe use of such cladding. Allis-Chalmers has been investigating aluminum alloys since 1957. An aluminum alloy was used as cladding for Borax fuel elements, but has not been used in a large power reactor. Aluminum has certain inherent limitations relative to corrosion resistance and strength, but a design that recognizes and accommodates these limitations appears obtainable. Should the development program fail to justify the choice of aluminum alloy by the time it is necessary to specify the first core for Pathfinder, it is technically possible to substitute zircaloy-2 as the fuel cladding material. Such a fuel element is being designed as an alternate and will be used if necessary. Zircaloy-2 clad fuel elements are being used in both EBWR and the Dresden Nuclear Power Station. Only minor modification of the Pathfinder plant would be required to use Zircaloy clad fuel. There has been substantial experience with this cladding material, and such material would constitute an acceptable alternate choice, although if zircaloy is to be used, a specific proposal including the particular design will need to be submitted to the Commission for consideration and for approval.

NSP has proposed a program whereby reactivity in the reactor may be varied by changing the flow rate of recirculated water through the boiler portion of the core. The details of this plan need to be developed more fully, but NSP has a testing program under way reasonably calculated to lead to information upon which to base the design for such a control plan. A feasible alternative, the use of control rods, is possible should recirculation control prove unacceptable. The present design of the Pathfinder project provides for control of the reactor up to 75 percent power by manual adjustment of the control rods. Above 75 percent power, the reactor is proposed to be controlled by control rods or adjustment of the recirculation flow rate.

A change in the flow rate effects a temporary readjustment of the volume of steam bubbles in the boiler core. This changes the moderator density and, therefore, the reactivity. If an increase in power is desired, the coolant flow rate is increased in a controlled manner. The reverse process is effected by decreasing the flow. A research and development program associated with this feature of the reactor is in progress. Indications are that control of recirculation in the proposed power range is feasible. As an alternative, if those indications are not substantiated, automatic control in the power range in the conventional manner can be accomplished with only minor modifications in the control circuitry, or manual control over the entire power range can be used.

Gaseous wastes generated in the reactor water, and ventilation air that might carry radioactive materials, will be passed through filters before being exhausted through the stack. The activity of the stack effluent will be held below specified limits by dilution, filtration, or hold up, as necessary, to comply with AEC regulations, 10 C.F.R.,

Part 20. All water leaving the site will be demineralized and will be monitored to insure that it does not exceed the nonoccupational maximum permissible concentration levels, and corrosive water will be neutralized. Areas of the site in which water is at radiation levels above those specified by AEC regulations, for nonrestricted areas will be restricted, and solid wastes will be disposed of by shipment to offsite approved disposal areas.

Decontamination nozzles will be provided on various components where internal contamination can be expected to be required. A decontamination room will be provided where equipment which can be moved will be decontaminated. Personnel will be suitably protected by protective clothing, monitoring, and decontamination facilities. The shield and fuel storage pools will be provided with filters and demineralizers to clean the water. Although it would be difficult, the design of the proposed facility will permit the installation of appropriate in-core instrumentation in cores subsequent to the first one (for which general in-core instrumentation plans have been described).

The maximum design accident considered by the NSP assumes that a recirculating water line of the reactor system at the base of the core is severed completely and abruptly so as to release the coolant rapidly to the containment building. The chain of events which might follow a rapid loss of coolant are: (1) melting of fuel, (2) release of fission products to the containment structure, (3) leakage of fission products to the atmosphere, (4) transport and diffusion of fission products in the atmosphere, and (5) exposure of persons to the cloud of fission products.

The Pathfinder control and instrumentation system will include safety circuits and interlocks that will shut down the reactor whenever safe limits of operation are exceeded, or whenever conditions such as equipment or instrument failures exist that could make contained operations potentially dangerous. The consequences of the postulated maximum design accident are more severe than those believed to be possible from any credible accident. The reactor building is designed to contain radioactivity resulting from such a maximum design accident. In event of such an accident there would be sudden loss of coolant. Water and flashing steam would be released into the reactor building. The core would eventually melt, and fission products would be released into the reactor building. The reactor building design specifications are such that it will leak no more than 1 percent of the contained volume per 24 hours at a differential pressure of 78 psig, the design pressure for the building. It is expected that actual leakage rate will be less than 0.1 percent per day upon completion of construction. A spray system to further reduce exposure can be installed at any time. No metal-water reaction is expected.

Without taking credit for a number of conservative factors such as meandering wind direction, the likelihood of a more restrictive leakage rate than is specified for the design of the containment vessel, and the advantages of the incorporation of a water spray system, the postulated maximum design accident for the proposed facility

would result in the following maximum doses to persons exposed to the radioactive cloud for one hour:

TABLE OF LIFETIME DOSES

(r or rem)

	Site boundary (about ½ mile)	*Location of recipient nearest edge of Sioux Falls (5.5 miles)*
Thyroid	2750	140
Bone Dose (Sr^{90})	43	2

Because of the magnitude of the calculated doses, consideration should be given by NSP to feasible design modifications which might reduce the potential hazards. Such design modifications might include reduction of the leakage specification for the containment vessel and the incorporation of a water spray system which would lower the consequences of any accident by quick reduction of building pressures and by washing down a portion of the fission products. NSP has specified a leakage rate of 1 percent per 24 hours at a differential pressure of 78 psi for the containment vessel. However, the applicant has testified that it expects to obtain a leakage rate of less than 0.1 percent per day and that upon completion of construction, if the leakage rate is 1 percent, the leakage rate could be reduced.

The suitability of the Pathfinder facility site must be considered both in its relation to the City of Sioux Falls, the center of which is 5.5 miles southwesterly from the site, and its location on the south bank of the Sioux River. The water supply of Sioux Falls is obtained from a field of shallow wells about 4 miles west of the Pathfinder site. The well field, however, is upstream from the proposed atomic plant, and the bottom of the field aquifer is at a higher elevation than the maximum artesian head in the vicinity of the atomic plant. Contamination of the artesian basin at the plant site should be prevented by the plant design and the internal pressure of the artesian aquifer. There are no public water supplies taken from the Big Sioux River[3] below the proposed atomic plant, and generally it is believed that the Big Sioux River will not be generally used as a recreational stream, but even if it should be any radioactivity contained in materials released to the stream will be at a safe level.

From a meteorological standpoint, the area of the plant site experiences extremes of heat and cold. Winds from every direction occur in every month. Inversions at the site are common, and the valley tends to channel air drift. Wind roses have been prepared and are part of a meteorologist's report prepared for NSP and have been submitted with the safeguards report. The design of the Pathfinder plant will be predicated upon the assumption that it can withstand the effects of the worst flood known to have occurred in the last 100 years. Earthquakes in this area are slight, but will be considered in the design. NSP plans to make arrangements to control the traffic on the railroads and the highways which cross or are located near the plant site in the event of an unforeseen

[3] NSP represents that the Big Sioux is used for irrigation, stock watering, recreational fishing and occasional swimming. It is said that the magnitude of these uses varies from year to year, and sometimes depends upon the stream flow which in some years is so low that most of these uses are not feasible.

accident. Such arrangements would probably involve the local railroad and highway authorities and the highway police.

A program of monitoring background radiation at six stations, four of them roughly at a 50-mile radius around the site, one at the site, and one at Sioux Falls, was started in the spring of 1959. This is a comprehensive program and includes analysis of fall-out, streams and water supplies, wild life, vegetation and milk. The program will be continued after the plant is in operation.

The public and state officials of South Dakota and neighboring states have been fully informed of the plans for this plant. The State of South Dakota will require a permit before waste is disposed into a stream. It is the intent of the applicant to secure such a permit.

The site is considered to be an acceptable one for a facility, of the general type and power level of the proposed Pathfinder facility. The general containment concept for this facility, as may be modified by data derived from the research and development program, appears to be adequate.

In a letter dated December 14, 1959, the AEC's Advisory Committee on Reactor Safeguards stated, in part, as follows:

> "The applicant has research and design work in progress on these problems which must be resolved before operation as proposed. Part of this development includes partial power range operation and testing in the reactor before full-scale operation.
>
> "The applicant proposes stepwise approach to full power and to design superheat. Operation with reduced superheat or without superheater fuel, with zircaloy fuel cladding instead of aluminum or without using the special control features has been suggested as feasible by the applicant, and proposed in case of need.
>
> "The Committee concludes that the approach suggested as above by the applicant will enable construction of the reactor proposed at its site with reasonable assurance that it can be operated without undue hazards to the health and safety of the public."

As heretofore noted, the Pathfinder facility is generally described as a boiling water reactor, but it is proposed to incorporate several new features and unique characteristics which have not been proven out[4] in any other reactor facility. For the purposes of an operating license, both NSP and the Staff are agreed that further analysis and research and development explorations must be conducted before a judgment can be made that the superheater, the aluminum cladding, and the recirculation rate of the primary coolant can be safely incorporated into this particular reactor. In addition, the Commission's HEB Chief testified that by omitting all the novel and unproven features, the Pathfinder facility could be constructed and

[4] By way of further detail in addition to the problems respecting the utilization of the superheater, the Chief of the Commission's Hazards Evaluation Branch (HEB) of the Division of Licensing and Regulation, testified, in part, as follows: "NSP . . . proposes to use an element (for fuel) composed of uranium dioxide clad with an aluminum alloy. Such elements have been used only to a very limited extent and there needs to be developed substantially more experience before such elements can be confidentially accepted for routine operation in power reactors . . . such a research and development program is necessary and . . . the proposed program is a reasonable one." (Parentheses added.) Further, in connection with the superheater the witness added: "The nuclear superheater is in essence a portion of the reactor which is cooled by the steam generated in the boiling section of the reactor . . . there exists little knowledge of the proposed fuel element capabilities." (for these superheater elements). . . . in our opinion, however, prototype elements similar to those to be used in the superheater should also be tested under environmental conditions of temperature, pressure and radiation comparable to those likely to be encountered in the reactor (Parentheses added)."

operated as a straight boiling water reactor,[5] but certain conventional features for even this type of plant warrant further investigation.[6] The witness further testified that there were also some problems of extrapolating from a test reactor to a reactor of the size of the Pathfinder facility, even though it was somewhat smaller than the Dresden facility. The witness added that: "There will be the same questions for the Pathfinder plant, but the extrapolation is much less" and the Dresden data will be helpful in this consideration. An important qualification to the consideration given to the Pathfinder facility with its novel features is amplified by the HEB Chief as follows:

> "The initial operation of the reactor will have to be under specified and limited conditions until we prove out the beliefs that we have formed as to the reliability of components (even) by the tests which were earlier performed." (Parentheses added.)

And again:

> "Yes, I think that there should be a research and development program carried forward, and that that should be a condition of proceeding with the project."

The determination of the existence of reasonable assurance that the utilization facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public is thus limited by the present record to the conventional type of boiling water reactor.[7] The Atomic Energy Act requires the finding, in accordance with Section 182, that there has been provided ". . . adequate protection to the health and safety of the public," and the Commission, in consideration of the entire Act and its regulations, has announced the principle:

> "There can be no doubt that public safety is the first, last, and permanent consideration in any decision on the issuance of a construction permit or a license to operate a nuclear facility." In the Matter of Power Reactor Development Corporation, Docket No. F–16.

The several novel features to be incorporated in the Pathfinder facility appear not only to require performance of tests in a research and development program, but also may well require testing in the reactor itself, after its construction.[8] The requirements of the Act and the regulations of the Commission warrant the provisional character of this construction permit so that data will be developed upon a record of all aspects of the novel features. It can be, and is,

[5] The testimony, at one part, was: ". . . if we aren't able to reach the adjustments of safety in these novel features, then presumably they would be left out, and we would be left then with a plant which we could accept, as to containment and site."

[6] The Commission's witness testified: "The following are particular features on which further effort must be expended before the operating license stage review: (1) Attention should be given to provision for appropriate in-core instrumentation in cores subsequent to the first one (for which general in-core instrumentation plans have been described) and (2) certain concepts of waste disposal as described in the hazards report may not have been satisfactory in the proposed retention pond and decay storage pit and posed problems of accidental release of waste material which would require further study . . . the approach now suggested . . . seems to be reasonable. . ."

[7] Again, the HEB Chief stated: "In any event, . . . (NSP) . . . has shown, and we concur that the Pathfinder reactor can be designed, constructed and operated as a straight boiling reactor without a nuclear superheater should data fail to support a conclusion that operation of the reactor with the nuclear superheater can be undertaken without undue hazard to the public and the on-site employees." (Parentheses added.)

[8] The Commission's HEB witness also testified: "It is recognized that despite such a test program there will remain the necessity for some of the performance testing of the elements to be accomplished in the reactor itself during initial operations. Failure could occur there despite successful preoperational testing."

found, however, that all technical data of the kind related to a conventional boiling water reactor, without the three foregoing aspects, will be forthcoming in order to complete the application. To this extent, the requirements of Section 50.35 of the Commission's regulations are satisfied. Respecting the three major novel features, if the research and development program does not confirm the opinions as to their use, these features will be omitted and satisfactory substitutes are suggested. As to these latter, further hearings will permit a consideration of such alternatives, and based upon other experience of the use of the suggested substitutes, although not in the particular design apparently needed for the Pathfinder project, there is reasonable assurance that technical data needed to complete the application, upon the assumption that substitutes can be used, will be forthcoming.

While NSP has not previously constructed nor operated an atomic reactor facility, it has constructed and operates a utility system in the four-state area of Minnesota, Wisconsin, North and South Dakota, having a gross generating capability of 1,695,000 KW. NSP has contracted with Allis-Chalmers Manufacturing Company to construct the nuclear utilization facility plant and to place it in operation, utilizing NSP personnel in the operating phases of this responsibility. Most of the plant operating personnel will be men presently employed by NSP and experienced in steam plant operation. A schedule of training for these men has been developed. Progress under the training schedule is being closely supervised, and to this time, is satisfactory.

Allis-Chalmers has had experience in the construction of nuclear facilities. Its atomic energy division designs and builds nuclear reactors and related equpment. This division has undertaken the design and construction of six research and materials testing reactors and three power reactors and has completed preliminary design and studies of four additional research reactors. Of these, two research and testing reactors are in operation and three are in the latter stages of construction. NSP and its contractor, Allis-Chalmers Manufacturing Company, collectively are technically qualified to design, construct, and operate the proposed facility.

The estimated capital cost of the Pathfinder plant is $22,660,000, of which $870,000 is for overhead costs of NSP, but the actual cost of the project may be 10 percent to 15 percent higher than the current estimate. Total operating revenues in 1959 for NSP amounted to $170,353,969. Operating revenue deductions, including income taxes, totaled $138,421,995, leaving a net operating income of $31,931,974. Interest and other income deductions totaled $7,026,529. The addition of other income items produced a net income of $25,035,832. The Company has a ratio of bonds to net plant being 44.9 percent and has a base for stockholders' equity of 47.9 percent of total assets. The estimated cost to NSP of $19,082,928 required to construct the Pathfinder plant will not place an undue burden on that Company. Even if the contribution anticipated from CUAPA were disregarded, and if NSP were required to expend $22,660,000 to finance the construction of the proposed project, NSP is financially qualified to construct the proposed reactor. Excess operating costs of the plant, estimated at $1,200,000 during the first 5 years of

operation, will be absorbed by NSP in operating expense as they occur, as will all fuel costs. In addition, Allis-Chalmers Manufacturing Company is financially qualified to absorb the additional costs of the Pathfinder plant in the event that the estimated costs of construction should be low by 2½ to 3 million dollars. At December 31, 1958, Allis-Chalmers Manufacturing Company had total assets amounting to $469 million. Its long-term debt was $92 million. Its current assets totaled $333 million, and its current liabilities amounted to $69 million. During 1958, its earnings amounted to approximately $20 million.

In addition to the foregoing findings and conclusions, upon the basis of the record of evidence including exhibits, and the proposed findings and conclusions submitted by the participants herein, it is further found and concluded that:

1. Northern States Power Company, a Minnesota corporation with principal offices at Minneapolis, is a qualified applicant to receive a construction permit provided by the Atomic Energy Act, as amended.

2. The nuclear reactor project proposed to be constructed by NSP is a utilization facility within the meaning of Section 104(b) of the Atomic Energy Act, as amended.

3. There is sufficient information available to provide reasonable assurance that the utilization facility of the general type proposed in the application, as amended, by NSP can be constructed and operated at the location specified therein without endangering and will provide adequate protection to, the health and safety of the public.

4. There is reasonable assurance that, in accordance with (1) research and development programs outlined in the evidence, (2) the tests proposed to be undertaken of the components, and (3) the availability of proven features to be incorporated in the reactor should necessity therefore require, the technical information omitted from the application will be supplied, including particularly that related to fuel elements to be utilized in the boiler region of the core, the nuclear superheater and its fuel elements proposed to be incorporated as an integral part of the basic reactor core, and the controls of the variable rate of primary coolant recirculation.

5. NSP and Allis-Chalmers Manufacturing Company, collectively are technically qualified to design, construct, and operate the proposed utilization facility.

6. NSP is financially qualified, and as supplemented by contributions from associated electric utilities is able to engage in the activities proposed in its application in accordance with the Atomic Energy Act, as amended, and the Rules and Regulations of the Commission.

7. The issuance of the proposed construction permit to NSP, substantially in the form attached to the Notice of Hearing issued herein, as modified by the Order hereinafter made, will not be inimical to the common defense and security or to the health and safety of the public.

WHEREFORE, IT IS ORDERED, subject to review by the Commission upon its motion, or upon appeal after exceptions, if any, are filed:

A. The Director of the Division of Licensing and Regulation of the Atomic Energy Commission is directed to issue a provisional construction permit substantially in the form attached to the Notice of Hearing issued herein by the Commission, as modified by this Order.

B. This authorization is provisional subject to the condition that an authorization to operate the facility will not be issued by the Commission until its proposed research and development program has been completed and evidence has been adduced by NSP and it has been found by the Commission, that the NSP proposed nuclear utilization facility, including among other components which must be proven acceptable by tests herein described, the novel features respecting the fuel elements in the boiler region of the core, the incorporation of a superheater with unproven types of fuel elements as an integral part of the basic reactor core, and the proposal that a variable rate of primary coolant recirculation be utilized for controlling the power level of the core, has been constructed and can be operated with adequate protection to and without endangering the health and safety of the public in accordance with the requirements of the Atomic Energy Act, as amended, and by the Rules and Regulations of the Commission.

C. Exceptions, if any, and briefs in support thereof must be filed by May 11, 1960; briefs in opposition thereto shall be filed on May 13, 1960, and if the Commission does not initiate a review on its own motion, and no exceptions are filed, this decision shall, in accordance with the Commission's Rules of Practice, become final on May 12, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

628162—62——22

DOCKET NO. 50–29

IN THE MATTER OF YANKEE ATOMIC ELECTRIC COMPANY

Issued April 22, 1960

APPEARANCES:

Donald G. Allen, Esq., John W. Cochran, Esq., and *Allen O. Eaton, Esq.,* for Yankee Atomic Electric Company.

Gerald Charnoff, Esq., for the Staff of Atomic Energy Commission.

INTERMEDIATE DECISION

Yankee Atomic Electric Company, Boston, Mass., (Yankee), has filed Amendments No. 14, 15, and 16 to its application for a construction permit and operating license for a pressurized light water reactor constituting a utilization facility contemplated by Section 104(b) of the Atomic Energy Act, as amended. Amendment No. 15 requested that the Construction Permit No. CPPR–5 previously issued to Yankee on November 4, 1957 be amended to thereby provide for approval at this time of all technical specifications of this nuclear facility and in addition, to amend the permit by deletion of the following paragraph:

> "This permit is provisional to the extent that a license authorizing operation of the reactor will not be issued by the Commission unless Yankee has submitted to the Commission (by proposed amendment to the application) the complete, Final Hazards Summary Report (portions of which may be submitted and evaluated from time to time) and the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in accordance with the specified procedures."

Amendment No. 15 transmitted 42 pages of revised technical information which was represented to constitute, with the previous filings, Yankee's Final Hazards Summary Report. Yankee stated that this report reflected the final design of the reactor, and Yankee has designated those portions thereof which it submits as technical specifications to be included in the aforesaid construction permit and in any superseding operating license.

The Commission, on January 27, 1960, issued its Notice providing for hearings which were held on March 3 and 4, 1960 (Vol. 24 F.R. Page 820). Appearances[1] were entered on behalf of Yankee and the Staff of the Commission.

[1] By Order entered on February 26, 1960, Elliott Earl, of Windsor, Conn., who alleged himself to be a customer of a company which planned to purchase electricity from Yankee, and who also stated that he was President of the Institute for Nuclear Serology, which is interested in nuclear and atomic subjects generally, was denied permission to intervene as a party in the proceeding but was permitted to enter a limited appearance in the proceeding as provided by the Rules of the Commission. Elliott Earl did not appear at the hearings nor file any statement of his views or position in the proceeding within the time provided for proposed findings and conclusions. No other persons sought to intervene or otherwise to participate in the proceeding.

Yankee submitted proposed findings and conclusions on March 22, 1960; the Staff filed comments and similar proposals on April 1, 1960.

In the Notice of Hearing issued by the Commission, the following particular issues were specified for consideration:

1. Whether Yankee has submitted all technical information required by the Act and the Commission's regulations to establish the final design of the reactor;

2. Whether the technical specifications proposed by Yankee pursuant to Section 50.36(b) of the Commission's regulations shall be deemed to be the technical specifications of Construction Permit CPPR–5 and the superseding operating license, when and if such operating license is granted;

3. Whether the processes to be performed, the operating procedures, the final design of the facility and equipment, the proposed use of the facility, and the technical specifications collectively provide reasonable assurance that the health and safety of the public will not be endangered by the proposed operation of the facility; and

4. Whether Construction Permit CPPR–5 shall be amended in whole or in part and under what conditions, if any, by granting AEC approval with respect to (a) the applicant's final design of the facility and equipment, (b) the applicant's proposed technical specifications, or (c) the proposed operating procedures.

In addition, the Commission set forth in the Notice that if the request for amendment made by Yankee were granted in the entirety, the sole remaining issues to be considered at the subsequent hearing with respect to conversion of the Construction Permit CPPR–5 to an operating license would be:

1. Whether construction of the facility has been completed in compliance with the terms and conditions under Construction Permit CPPR–5;

2. Whether Yankee is technically and financially qualified to engage in the proposed activities in accordance with Commission regulations;

3. Whether Yankee has furnished proof of financial protection in accordance with Commission regulations; and

4. Whether the issuance of an operating license to Yankee will be inimical to the common defense and security.

The Notice of Hearing referred specifically to Amendments 14, 15, and 16 to Yankee's application. The evidence at the hearing was extended to include a consideration of Amendments 17 and 18. A further Amendment No. 19 was filed too late for either review by the Advisory Committee on Reactor Safeguards or by the Staff of the Commission.

The issue of financial ability of Yankee was determined and found adequate at an earlier hearing held April 9, 1959.[2] While there appears to be some difference of opinion respecting the finality of any determinations of the question of safety in this proceeding,

[2] Intermediate Decision dated May 19, 1959.

Yankee recognized at the hearing that safety issues[3] are present, and at issue, in any licensing proceeding whether involving a construction permit, or at any later stage of the operating licensing proceedings.

The facility being constructed[4] by Yankee is a pressurized light water moderated and cooled reactor designed to produce 134,000 kilowatts of electrical energy and is located at Rowe, Mass. Amendment No. 14 to its license application dated September 15, 1959, included material identified as the Final Hazards Summary Report, which superseded a preliminary report filed April 12, 1959. Amendment No. 15, dated October 2, 1959, also involved in the instant proceeding, supplied certain operating and maintenance instruction, and also supplied the technical specifications which Yankee desires to have approved and incorporated in the construction permit, and ultimately incorporated in any superseding operating license. Amendment No. 16, dated December 4, 1959, submitted some revisions in the technical specifications, but this submittal has again been revised.[5]

Yankee was incorporated as a Massachusetts utility company in 1954. Its stockholders are eleven New England Electric utility companies[6] who will purchase the entire electrical output of the plant upon its completion. Total capital costs of the project are estimated at $57,000,000 of which approximately $50,000,000 is applicable to plant construction costs and the balance to the cost of the initial core, working capital and other corporate expenses. The foregoing estimate is exclusive of an additional $5,000,000 of costs associated with the preliminary research and development program,

[3] The Notice of Hearing stated that if Yankee's request in the instant proceeding is granted in its entirety, then the remaining issues would be as therein listed, excluding any reference to safety considerations. The President of Yankee testified regarding the finality of major safety questions in the instant proceeding as follows: "We have asked in the present proceedings to have our provisional construction permit converted into a definitive construction permit by approval of the final design of the plant, including all safety considerations and the technical specifications . . .

* * * * * * *

"Our hope is that all *major safety matters* will be reviewed and disposed of at this time, in order to simplify the issues in the final hearing on issuance of an operating license." (Emphasis added.)

The Staff stated that . . . "the health and safety issue will again arise . . . in connection with the conversion to the operating license" and to that extent the Staff apparently contends that Yankee's request, as set forth in the Notice of Hearing can not be granted in its entirety. The Staff added that it believed that the "larger issues" of health and safety could be determined at this hearing. Amendment Nos. 17, 18, and 19, filed by Yankee are not within the list of Amendments specified in the Commission's Notice of Hearing. While witnesses characterized Amendment No. 19 as involving only minor design changes (Yankee submitted 40 pages of revised technical information) it can not be determined on this record whether there are major issues of safety involved in that amendment which affect the other reactor components, or the technical specifications described herein.

[4] Yankee estimated that its facility was 80 percent to 90 percent completed at the time of the hearing.

[5] Amendment No. 17 is described by Yankee as constituting editorial changes in the Final Hazards Summary Report and Amendments 18 and 19 involve design changes.

[6] See the following table:

Sponsoring company	*Percent of stock ownership*
New England Power Company (Subsidiary of New England Electric System)	30.0
The Connecticut Light & Power Company	15.0
Boston Edison Company	9.5
Central Maine Power Company	9.5
The Hartford Electric Light Company	9.5
Western Massachusetts Electric Company	7.0
Public Service Company of New Hampshire	7.0
Montaup Electric Company (Subsidiary of Eastern Utilities Associates)	4.5
New Bedford Gas & Edison Light Company	2.5
Cambridge Electric Light Company	2.0
(Subsidiaries of New England Gas and Electric Association)	
Central Vermont Public Service Corporation	3.5

which has been carried out with Commission financial assistance under the Power Demonstration Reactor Program pursuant to AEC Contract No. AT(30–3)–333 dated June 4, 1956. Yankee's financing program, which was considered in a prior proceeding in this docket, contemplates a stock investment of $20,000,000 by the sponsoring utilities and senior financing by the sale of $20,000,000 of first mortgage bonds and $17,000,000 of bank borrowings. The basic stock investment has now been completed and current construction costs are being financed by periodic drawdowns of funds pursuant to commitments with senior investors. The applicant reports that its original estimate of costs appears to be adequate and may be underrun. Construction is well advanced and is scheduled to be complete so as to permit loading of the initial core by August 1, 1960, with a possibility that this date may be anticipated by as much as a month.

The Yankee plant is located on the Deerfield River in northwestern Massachusetts just south of the Vermont border and approximately 10 airline miles east of North Adams, the nearest major population center. The site was selected, among other reasons, because of its proximity to existing transmission facilities, which interconnect with the main New England transmission grid, and because of the availability of adequate cooling water from Sherman Pond, which serves an existing hydro-electric generating station. The Deerfield River forms a valley in the surrounding Berkshire Hills which rise to a height of approximately 1,000 feet on either side of the plant site. The surrounding area is predominantly rural and sparsely populated. In the immediate vicinity of the site is an exclusion area of approximately 2,000 acres owned by Yankee and its affiliate, New England Power Company. A secondary highway passes the site on the opposite side of the river approximately 1,000 feet from the plant at its nearest point. The topography, meteorology, hydrography, geology, and seismology of the site are described in detail in the license application and appear to be generally favorable for the proposed operation.

As stated, the plant will consist of a pressurized water reactor for the production of steam and a turbine generator of conventional design for the production of electricity. The reactor itself and all associated components of the primary main coolant system will be enclosed in a steel containment vessel which will be sealed at all times when the reactor is critical. The control room is located in the turbine building adjacent to the containment vessel. There is sufficient shielding between the containment vessel and the control room to permit prolonged operation of the control room in the event of the worst credible release of radioactivity to the containment vessel.

The primary coolant system is a closed circuit that is kept under pressure to prevent boiling in the reactor. Water from the primary system is then led through steam generators where steam is created in the secondary system to supply the turbine generator. Water from the primary system is treated in the waste disposal system for further use or ultimate disposal. Residual wastes will be disposed of in accordance with Commission regulations.

The core of the reactor consists of 23,142 stainless steel clad fuel rods, approximately 7½ feet in length and 0.34 inches in outside

diameter. The nuclear fuel consists of sintered uranium dioxide enriched in the fissionable isotope U-235 to 3.4 wt. percent. Individual fuel rods are assembled into groups of 304 or 305 to form 76 fuel assemblies. The fuel assemblies are designed to permit insertion of 24 moveable control rods and 8 fixed shim rods. Individual drives for each control rod are mounted on the reactor vessel head and operate on a magnetic jack principle. If a scram signal is initiated, or if the electrical supply to the control rod drives is cut off for any other reason, the magnetic coils of the drives will be de-energized and the control rods will drop into the core by gravity to shut down the reactor. The control rods are fabricated from an alloy of silver cadmium and indium and are nickel-plated to reduce corrosion. The control rods are designed to have a neutron absorption worth adequate to shut down the reactor at all times during operation and will be supplemented by a boric acid system designed to provide additional control to keep the cold, clean reactor subcritical during extended shutdowns.

The main coolant system consists of four identical loops of 20 and 24 inch-stainless steel piping leading from the canned motor-type circulating pumps to the reactor vessel and thence to the steam generators. The primary system is expected to operate at pressures ranging from 1,800 psi to 2,300 psi and at temperatures in the range of 495° F. to 535° F. Corresponding steam conditions in the secondary system will be 475° F. at 525 psi. Pressure in the primary system will be controlled by an electrically heated pressurizer designed to control transients caused by an increase or decrease in power demands. The primary plant has 17 auxiliary systems, including the pressure control and relief system, charging and volume control system, chemical shutdown system, purification system, component cooling system, corrosion control system, sampling system, radioactive disposal system, shutdown cooling system, vent and drain system, safety injection system, reactor control system, nuclear instrumentation and reactor projection system, radiation monitoring system, vapor container atmosphere control system, decontamination system, and fuel handling system. For the most part, these systems are conventional in design and pose no safety or engineering problems in connection with a nuclear power plant.

The radioactive waste disposal system was the subject of a preliminary report from the Advisory Committee on Reactor Safeguards dated October 21, 1958, and appears adequate to contain and treat all radioactive wastes from the plant so that they can ultimately be disposed of safely and in accordance with the Commission's regulations. The instrumentation included in the reactor control system appears to provide all necessary information to the reactor operator for the safe control of the reactor. In addition to the control instrumentation, the applicant proposes to include in-core instrumentation in the form of flux wires and thermocouples which will be used to obtain additional information as to flux and heat distribution within the core. The radiation monitoring system includes instruments to provide continuous information as to the level of radioactivity at the exit of the main plant stack and at other critical points in the plant, with visual and audible alarms in the event of activity in excess of predetermined levels.

The technology of pressurized water reactors is well established through successful operation of various Commission-owned experimental facilities. These include the naval propulsion reactors used to power the Nautilus and other atomic submarines, the Army Package Power Reactor, and, of particular relevance to the Yankee reactor, the pressurized water reactor now in operation at the Shippingport power plant. The principal innovations in the Yankee reactor relate to core design. As compared with the Shippingport core, these innovations consist of the use of uranium dioxide instead of metal uranium, the use of a uniformly enriched core instead of a seed-and-blanket arrangement, and the use of boric acid as a supplementary control feature. These matters have been the subject of extensive investigation through the Commission-financed research and development program, and the design of the initial core has been based on the results of this program.

From the point of view of safety, the important characteristics of a pressurized water reactor are the normally negative temperature and power coefficients, which tend to limit neutron multiplication as temperatures in the fuel and moderator increase. Calculated results and critical experiments conducted as a part of the Yankee research and development program confirm the strongly negative character of the coefficients of reactivity in the reactor. Calculations indicating that these negative reactivity coefficients will not be affected appreciably by the buildup of plutonium will be verified in the Yankee power reactor at startup, after 2,000 hours of operation, and at succeeding intervals during core life. The effects of plutonium buildup on these coefficients have been stated to be small enough to permit these measurements[7] to be made with safety during the course of operation of the Yankee reactor. It is expected that the measurement at successive 2,000-hour intervals will indicate trends which are taking place long before any of these would significantly affect safe operation of the reactor.

The initial Yankee core has been designed to insure a capability of 392 MW thermal, or 110 MW electrical, in the expectation that the actual capability of the core will probably be commensurate with the design capacity of the plant of 485 MW thermal or 136 MW electrical. A similar increase in actual core capability has been a common experience in other reactors, and if available here will contribute to a lower cost per kilowatt hour for the Yankee plant. The availability of this added capability will be determined when the results of tests are available from operation at 392 MW thermal.

The controlling factors of this added capability are the hot channel factors, which state the relationship between average temperature conditions and maximum temperature conditions in the core.

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[7] In its October 21, 1958 letter, ACRS stated in reviewing Amendment No. 7 filed by Yankee: "The Committee recommends that Yankee be asked to provide a description of the specific experiments which will be made to determine the effects of plutonium buildup on prompt and over-all temperature coefficients. It suggests that controlled transient experiments, with known sinusoidal or steps changed in reactivity, may be a convenient means of measuring these coefficients." In the February 1, 1960 letter, ACRS referred to no other data in this regard than Amendment No. 7. It is not known if Yankee has outlined the experiments requested by ACRS. Whether or not submitted to ACRS, Yankee testified at this hearing, apparently without presently accepting the measurement suggestion of ACRS, that ". . . the exact method of making the tests will be determined to some extent after making these tests during the startup period." Since ACRS requested a review of the specific experiments which appear now to await formulation until after initial startup, it may be that the statutory review by ACRS must include these plans when they are known.

The hot channel factors used in designing the core are believed to embody a high degree of conservatism, which can be determined by means of the in-core instrumentation provided for in the reactor design. Safe operation of the reactor at power levels in excess of 392 MW thermal, but not exceeding 485 MW thermal, can be achieved if operating results at 392 MW thermal demonstrate that thermal conditions in the hottest channel and other operating conditions at the increased power level will be within the operating limitations incorporated in the technical specifications set forth in the proposed technical specifications. Accordingly, the Applicant has proposed, and the Advisory Committee on Reactor Safeguards and the Hazards Evaluation Branch have endorsed, a program which will require a minimum of 500 hours of operation at 392 MWT in order that the necessary experimental information can be obtained and evaluated by the Commission before operation at higher levels of power may be authorized.

The final Hazards Summary Report includes detailed operating instructions which set out the procedures to be followed and the various tests to be made during initial startup of the reactor and during normal operation. Additional operating instructions cover in detail the procedures to be followed under emergency conditions which might arise from malfunctioning of equipment or as a result of a prior operator error. In addition, operating instructions are provided for maintenance and refueling of the reactor. The Advisory Committee on Reactor Safeguards and the Hazards Evaluation Branch have indicated their approval of the scope and content of the proposed operating procedures.

The hazards evaluation section of the final Hazards Summary Report analyzes a series of possible accidents that might occur in cases predicated on an assumed failure or malfunctioning of equipment, in cases involving operator error, and in cases compounded by both causes. The maximum credible accident is identified as arising from a major loss of coolant accident which might result from a complete rupture of one of the 20-inch main coolant lines. The anticipated sequence of events following such a break would be loss of the main coolant, by flashing into the vapor container, followed by a shut-down of the reactor upon loss of moderator. Operation of the safety injection system, either automatically or manually, would re-cover the core with water, terminate the accident, and handle removal of decay heat without damage to the fuel elements and without any significant release of fission products to the vapor container.

If it is further assumed that the safety injection system failed to operate, either automatically or manually, a melt-down of the fuel elements could ensue, with a release of fission products to the vapor container, but with no return to criticality of the melted-down core. Calculations which assume extreme values for the amount of resulting radioactivity in the vapor container and for possible leakage of air-borne particles from the vapor container, indicate that the possible exposure to the public at the nearest boundary of the exclusion area would be within certain tolerances, not believed to be harmful by Yankee, and could be avoided or minimized by the emergency procedures provided for in the operating instructions. Although the sequential compounding of failures assumed in the foregoing sequence of events is believed by Yankee and its contractors to go beyond the bounds of credibility, the analysis of the foregoing "hypothetical" accident indicates that no serious risk to the public health and safety would ensue. The conclusions of the Hazards Evaluation Branch of the Commission are in accord with the analysis presented by Yankee.

The technical specifications of the reactor submitted by Yankee are subdivided into five main sections dealing with the site, design specifications, performance specifications, initial startup, and operating procedures and restrictions. The purpose of the technical specifications is to state explictly the characteristics of the reactor and the operating conditions assumed by the Commission in evaluating the safety of the facility and to determine that they will provide adequate protection for the health and safety of the public. Yankee has suggested that by incorporating the technical specifications in a definitive construction permit, or in a subsequent operating license, any departure from the design characteristics or operating procedures approved by the Commission will require prior authorization by means of a license amendment. As a result, Yankee contends that any such change will be subject, as perhaps serial amendments are later filed, one by one, to prior review by the Commission, and Yankee concludes that approval of the change must be predicated on a reconsideration of the fundamental safety issues involved.[8]

Although a construction permit does not authorize operation, a full review to the extent possible, of the safety of a reactor necessarily assumes a specific set of operating procedures and safety restrictions, consistent with the design of the facility and intended use, as known at the time of considering the technical specifications. As a result, the technical specifications to be incorporated in a construction permit should generally include provisions designed to indicate the operating procedures and restrictions assumed in approving the then known design features of the reactor. A corollary of this procedure appears to be that, if the design of the facility is substantially changed after the review of technical specifications, at the construction permit stage, further consideration of safety must be given to the final specifications after the design is made final. In the instant proceeding involving Yankee, a further proceeding will be required prior to issuance of an operating license, at which time the question of safety will again be reviewed and will be based upon the final design of the facility reflected by the completed construction, and upon the final inspection thereof. (See Commission decision dated May 9, 1959, *Westinghouse Electric Corporation*, Docket No. 50–22.)[9] It follows that the technical specifications presently authorized for incorporation in the construction permit

[8] This approach was rejected by the Commission as unsatisfactory in the Vallecitos BWR proceeding wherein some 40 separate amendments were filed. (See *In the Matter of General Electric Company*, Docket No. 50–18.) For the Vallecitos reactor, the Commission ultimately provided a procedure to obviate many detailed submittals of changes, and while such revised procedure rested upon a different basis, the objective was to avoid "piece meal" separate considerations when it appears probable that many amendments are likely to be filed as revisions occur.

[9] The Commission has been explicit in its statement that the finding must be made in an operating license proceeding: ". . . that the facility has been completed in accordance with the application and construction permit." The following is to be noted: ". . . the Commission expects an applicant for a final operating license . . . to present on the record full and complete substantial and material evidence . . . that the facility . . . has been constructed . . .". (*Westinghouse Electric Corporation*, Docket No. 50–22, June 19, 1959.)

will be subject to revision in the light of the record in that subsequent proceeding, and the order entered herein will expressly so provide.

An important aspect of the safety consideration appears to relate to this design change, reflected by Amendments 18 and 19 to Yankee's application proposed to be made at this stage of the construction program.

As a result of the review of the final Hazards Summary Report by the AEC Hazards Evaluation Branch, it became apparent that the design of the plant would not permit continued operation with minor leakage from the primary system to the secondary system at a time when fission products are present in the primary system as a result of failures in fuel element cladding. A design change to permit safe operation under such conditions was indicated in Amendment No. 18[10] to the license application, dated January 13, 1960, and was covered in detail by Amendment No. 19,[11] dated February 24, 1960. However, the detailed amendment covering this change has not been reviewed by the Advisory Committee on Reactor Safeguards or by the Hazards Evaluation Branch of the Commission.

Accordingly, Amendments No. 18 and 19 cannot be considered in the record in this proceeding as a part of the final design of the reactor for the purposes of this proceeding. The filing by Yankee of these two amendments necessarily precludes the design presented in this proceeding from being considered as the final design of the facility.

In addition, an approval of the presently considered specifications for the present design, through Amendment No. 15, which is not a final design of the facility, may not remove the advisability of the submission of data by Yankee concerning developmental operation of the reactor after criticality. There are many novel changes in Yankee's pressurized water reactor. The evidence is that calculations have been made that all novel components will operate satisfactorily as a unit. Until these novel features substantially affecting the safety of the reactor experience from which complete extrapolations can be made, it may be advisable to provide that a step wise approach to power be undertaken and that data regarding each step be submitted to the Commission for its information and review.[12] Yankee has proposed an approach to 392 MW (thermal)

[10] Amendment No. 18 consists of one page and states that: "Design changes are being developed which will permit continued operation of the plant even with leakage from the primary to the secondary system. . . . These changes will modify present design as follows: (1) Radiation monitoring will be provided in the discharge from the air ejector system. (2) sufficient air dilution of the air ejector discharge will be provided to bring gaseous effluent activity concentration within acceptable limits. (3) the direct connection from the secondary blowdown system . . . will be eliminated and in its place a connection will be made . . . to the condenser circulating water discharge system. . . ."

[11] Amendment No. 19 included 40 pages of revisions to technical information for Yankees' Final Hazards Summary Report to reflect the three categories of changes generally described in Amendment No. 18. The changes reflected by Amendment No. 19 are stated to also include: ". . . a drawing showing the latest control rod design and a revised startup procedure covering the initial testing and operation of the control rod drives."

[12] This step approval of operations may also be desirable (and can be considered at the operating license proceeding), among other features, for the problem of plutonium buildup and its effect on kinetic characteristics of the reactor. ACRS first suggested the construction of a part core for critical experiments in order to measure the effects. Yankee thereafter concluded as it stated: ". . . that such experiments could better be made in the actual reactor because of the extrapolations required both in size and temperature and in pressure in translating the results of critical experiments to the actual reactor." Both the Staff and ACRS agreed to this approach, and Yankee will devise the necessary experiments after initial criticality. Yankee testified, however, regarding the plutonium buildup: ". . . it is only a minor factor, if any, as a function of power level as distinct from a function of coefficient of reactivity." So it is not directly a power sensitive factor and there is no reason to believe that there will be any problem in this case.

power and the submission of all data concerning operations prior thereto so that the Commission may specifically consider the advisability of authorizing an increase in power. These suggestions are particularly appropriate in licensing proceedings for operating authority.

For example, the problem of nucleate boiling at the surfaces of the fuel elements was described and the results of the research and development were indicated. The Chief of the Hazards Evaluation Branch of the Commission testified:

> "There are no suitable instruments by which the beginning or existence of nucleate boiling in a reactor core can be directly determined. Yankee therefore intends to use in-core devices to measure neutron flux distributions at preselected locations in the core and to measure coolant temperatures at the outlet of coolant channels. From experimental observations obtained with these devices during power operation, one should be able to reliably predict which regions of the core may sustain nucleate boiling at higher power levels."

Particularly, as to safety aspects, the Hazards Evaluation Chief added:

> "The safety of operation at power levels higher than those previous(ly) reached will depend on experimental information obtained from these in-core instruments and from general observations of the reactor behavior and performance at lower levels of power. We believe the conservatively calculated margins of safety with respect to the extent of nucleate boiling which might occur are sufficient to give assurance of safe operation under hydraulic and thermal conditions which will exist at power levels up to 392 MWT . . . power levels substantially above MWT can be predicted with a great deal more confidence when experimental data from operation up to that power level have been obtained."

In considering the effect of plutonium buildup, the Chief of the Hazards Evaluation Branch indicated that theoretical calculations were not sufficient but actual measurements were needed, in this language:

> "As the Yankee reactor is initially operated with fresh low enrichment fuel, the U-235 content is consumed and plutonium is generated from the U-238 present. Toward the end of fuel life, there is a substantial amount of plutonium present . . . this shift in composition conceivably could lead to significantly different characteristics of the reactor. Yankee presented calculations which indicated that the effect of plutonium buildup on nuclear parameters would not seriously affect reactor stability or other safety features, even at the end of core life.
>
> "These calculations appear to be quite dependable, at least in so far as their prediction of general trends to be expected. However, it is recognized that *theoretical calculations can not predict the eventual reactor characteristics with certainty; actual measurements of some sort are necessary.* Consequently, Yankee has proposed that the power coefficient of reactivity and the moderator temperature coefficient, the two parameters of chief concern, will be measured at each 2,000 full-power hour interval over the life of the first core. We believe this will be a safe means and the most dependable, of determining the significant effects of plutonium buildup. . . . The *details of procedure to be used in the determination of the coefficients at the intervals indicated have not been worked out* during preliminary measurements in the reactor experimental program." (Emphasis added.)

The same Staff witness also cautioned that the components of the control rods and drive mechanisms warranted periodic inspection to ascertain if resistance to corrosion was being exhibited, since, as he testified: ". . . extensive experience with such components has not been accumulated . . ."

This foregoing evidence certainly indicates that data from experience will constitute a basis for the required final evaluation of safety for the Yankee reactor. A present inference from the data so far presented is that complete extrapolation cannot be made from previous experiments, and to that extent, actual reactor operations will be needed to secure data which is concluded to be necessary for confident predictions of reactor behavior. It further appears that any data from any source different from that contemplated for any one of these novel features, i.e., of the use of boric acid control, nuclear boiling, plutonium buildup, or possibly the corrosive conditions or control rods or drive mechanisms, might require revision of many of the technical specifications presented in this reactor. The latest data possible affecting any of these features should be that utilized for considering the final technical specifications for actual operations. It is, therefore, concluded that while the technical specifications concerning this reactor design of the facility and known data of expected operations of the Yankee reactor are adequate, the numerous questions that exist and which concern the novel features of this reactor require that no more be done than that the modified technical specifications in this proceeding be conditionally approved. After completion of construction, by that time, any data from any experience for these features as related to the then final design that may be available, should be presented, not necessarily to modify the technical specifications, but to confirm the tentative approval and also to determine the form and step wise approach to power that may be necessary to provide adequate protection to the health and safety of the public. In a proceeding for operating license authority, an earlier approval of technical specifications should not be construed to preclude or obviate the presentation of any data from whatever source that may be of value in considering the adequacy of the health and safety aspects of a proposed reactor operation.

Yankee argues for approval of technical specifications, at this time, in this proceeding, regardless of the effect of its Amendments Nos. 18 and 19 so that operation and initial loading of the reactor will not be delayed by a proceeding for licensing operating authority. Yankee points out that it will sustain a monetary loss from an inactive investment if delay occurs. *In the Commonwealth Edison Company* proceeding, Docket No. 50-10, provision was made to avoid claimed delay and loss by an interim order to permit initial loading and operations to criticality at a low level of power. Commonwealth accepted that order and when the construction of its reactor was completed, use was immediately made of its facility by such initial loadings, during the pendency of the licensing proceeding. It may be that Yankee would find that a similar procedure would be useful and this phase may be considered in its proceeding seeking licensing operating authority.

These conclusions are not inconsistant with the AEC regulations and the provisions of the Yankee construction permit authorizing and encouraging Yankee to submit from time to time portions of the "final design" for evaluation. Yankee has shown a commendable effort in seeking to make final, to the extent practicable, operating specifications and features of the final design. However, on this

record, no finding can be made either that (1) the design is final, because of the substantial questions presented by Amendments 17, 18, and 19, or (2) that "the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in accordance with the specified procedures." For these reasons, the request by Yankee for deletion of the paragraph quoted on page one of this decision cannot be granted. However, the procedures of the Commission, contained in its regulations and developed by order and decision (particularly in the Commonwealth Edison proceeding, supra) afford a means for assuring that the planned loading, startup, testing, and operating schedule of Yankee can be carried out provided the required information is timely supplied.

In addition to the foregoing findings and conclusions, upon the basis of the record of evidence including exhibits and the proposed findings and conclusions submitted by the participants herein, it is further found and concluded that:

1. Yankee Atomic Electric Company, Boston, Mass., is the holder of Construction Permit No. CPPR-5 pertaining to a utilization facility provided by Section 104(b) of the Atomic Energy Act, as amended, and which permit was issued on November 4, 1957, and has been amended on February 27, 1958, October 28, 1958, July 7, 1959, and September 11, 1959.

2. Yankee has requested approval of all technical specifications for the present design of its utilization facility, and has further requested incorporation of such specifications in the construction permit and in any superseding operating license.

3. Yankee has not submitted all of the technical information available for consideration in this record as required by the Atomic Energy Act, as amended, and the Commission's regulations to establish the final design of the reactor.

4. The modified technical specifications considered in this proceeding pursuant to Section 50.36(b) of the Commission's regulations can be given only conditional approval due to the incompleteness of the design of the facility as shown in this record, as the technical specifications of Construction Permit CPPR-5 and for any superseding operating license.

5. Upon the basis of the presently incomplete design of the facility, it appears, however, that the processes to be performed, the operating procedures, the present design of the facility and equipment, the proposed use of the facility and the present form of the technical specifications, collectively provide, on the limited basis presented, reasonable assurance to that extent that the health and safety of the public will not be endangered by the proposed operation of the facility.

6. Final approval of Yankee's technical specifications, and proposed operating procedures must await the presentation of evidence of the final design of the utilization facility as reflected by its completed construction, as well as the presentation of any other data available concerning the novel features of this reactor.

7. Consideration should be given in the proceeding for an operating license for an interim order to permit initial loading and

attainment of criticality at a low power level pending full consideration of all data adduced for the scope of the operating license.

8. Construction Permit No. CPPR-5 should be amended to provide that conditional approval limited by the incomplete nature of the present design is given to the modified technical specifications and presently proposed operating procedures for Yankee's reactor utilization facility.

WHEREFORE, IT IS ORDERED, subject to review by the Commission upon its own motion, or upon appeal after exceptions, if any, are filed:

A. The technical specifications as submitted by Yankee Atomic Electric Company, and as modified by proposals submitted by the Staff are conditionally approved as adequate for the protection of the health and safety of the public based upon the design of the utilization facility as reflected by the application and amendments inclusive of No. 16 thereof, but not inclusive of Amendments 17, 18, nor 19 filed by Yankee for its application.

B. Final approval is not extended to the proposed technical specifications and proposed operating procedures reflected in this record.

C. At the contemplated hearing to be convened concerning the issuance of an operating license, consideration shall be given to the issuance of an interim order prior to the final order in that proceeding to thereby specify conditions for the initial loading and attainment of criticality to a low level of power.

D. Exceptions, if any, and briefs in support thereof, shall be filed by May 16, 1960, and briefs in opposition thereto shall be filed by May 18, 1960, and if the Commission makes no review on its own motion, and if no exceptions are filed, this decision shall, in accordance with the Commission's Rules of Practice, become final on May 17, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

DOCKET NO. 27-13

IN THE MATTER OF COASTWISE MARINE DISPOSAL COMPANY

Issued April 29, 1960

ORDER PERMITTING INTERVENTION BY CITY OF LONG BEACH AND DESIGNATING TIME AND PLACE FOR HEARING

On April 21, 1960, the City of Long Beach, Calif., filed its petition to intervene in the above captioned proceeding and alleged that it had not been informed of the original request made to this Commission by Robert Boswell for a license to undertake certain activities thereafter authorized by the Commission to be undertaken in Long Beach, and further, that the City of Long Beach had a vital interest, separate from that of the Commission and Boswell, in the operations undertaken to date and for the future by said Boswell. The City further alleged that Boswell, on September 4, 1959, sought a business license to engage in processing of radioactive waste, but the City alleged that the officers of the City did not, at that time, become fully aware of the nature of the business proposed to be conducted. The City alleges that the location of the business operated by Boswell is unsafe from the standpoint of the health of persons in the community and is in violation of the City's zoning provisions.

By a separate petition, City requested that these proceedings be convened within the City of Long Beach for the convenience of many persons who might be heard, and furthermore, the City alleged that there were other locations in Los Angeles County where operations by Boswell could be conducted without danger to health of inhabitants of that area.

The Staff of the Commission, on April 29, 1960, filed a combined answer to the foregoing petitions and entered no objection to the petition by the City to intervene, provided the issues of the proceeding would not be thereby enlarged by such intervention. The Staff did object to a consideration of the scope of municipal regulation as set forth in the City's petition on the ground that it was not directly related to the present issues in the proceeding, and also objected to any transfer of the location of the hearing from the Federal courthouse in Los Angeles to other facilities in the City of Long Beach.

The Presiding Officer *finds*:

1. Good cause has been shown to permit intervention by the City of Long Beach in this proceeding without any addition to or enlargement requested by the City of the issues specified for consideration by the Commission.

2. Sufficient grounds have not been established to warrant any transfer of the place of the hearing in this cause from the presently designated location.

The Presiding Officer *orders*:

A. The City of Long Beach, Calif., is permitted to intervene and become a party to this proceeding in accordance with the Rules of Practice of the Commission, without any addition to or enlargement requested by the City of the issues specified for consideration by the Commission.

B. The application by the City of Long Beach to transfer the location of the hearing is denied.

C. This hearing shall convene at 10:00 A. M. on the 9th day of May 1960 in Courtroom No. 9 of the United States District Courthouse, 312 North Spring Street, Los Angeles, Calif.

SAMUEL W. JENSCH,
Presiding Officer.

BYPRODUCT MATERIAL LICENSES No. 4-1695-2B59 AND No. 4-5741-1

IN THE MATTER OF ADVANCE INDUSTRIAL X-RAY LABORATORIES, A DIVISION OF AIR FRAME INSPECTION, INCORPORATED, RESPONDENT, AND ADVANCE INDUSTRIAL X-RAY LABORATORIES, INC., APPLICANT

Issued May 2, 1960

ORDER EXTENDING TEMPORARY LICENSE

On April 28, 1960, Advance Industrial X-Ray Laboratories, Inc. filed a request for an extension of the temporary license heretofore issued by the Commission, following hearings held on September 16 and 17, 1959, on the application filed by Advance. In the request for extension, Advance stated: "An additional two weeks will be necessary to fully prepare and finalize the said application . . ." On May 2, 1960, Advance transmitted a telegram to the Presiding Officer requesting an extension of time as previously submitted by the foregoing letter dated April 28. On May 2, the Staff of the Commission stated that the April 28 letter did not technically comply with Condition 3 of the Commission's Order, but the Staff had no objection to the letter being considered as a motion for the extension of the existing license, nor to the granting of the motion.

The Presiding Officer *orders*:

A. An extension of time is granted and the term of the license issued to Advance is extended from May 2, 1960, to May 16, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

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DOCKET NO. 50-155

IN THE MATTER OF CONSUMERS POWER COMPANY

Issued May 6, 1960

APPEARANCES:

George F. Trowbridge, Esq., of Marks & Trowbridge, and *Harold P. Graves, Esq.*, for Consumers Power Company.

Jerome Maslowski, Esq., Assistant Attorney General of Michigan, for State of Michigan, and with *Ralph W. Purdy* for Water Resources Commission of Michigan.

Troy B. Conner, Jr., Esq., for Staff of the Commission.

Donald Edward Van Farowe, for Michigan Department of Health.

INTERMEDIATE DECISION

Consumers Power Company (Consumers) a public utility operating company, organized under the laws of the State of Maine, but engaged in business solely in the State of Michigan and with principal offices at Jackson, Mich., filed an application on January 18, 1960, for a construction permit and an operating license for a nuclear utilization facility provided by Sections 104(b) and 185 of the Atomic Energy Act, as amended. The application was filed in response to the Power Demonstration Program [1] announced by the Commission in aid of endeavors to develop and construct nuclear reactors capable of producing significant quantities of electrical energy. The utilization facility sought to be constructed by Consumers is a direct, single cycle forced circulation boiling water reactor with an initial nominal capacity of 50,000 kw electrical (with corresponding thermal power of 156 MW) and an ultimate designed capacity of 75,000 kw electrical (with corresponding thermal power of 240 MW).

On February 19, 1960, the Commission issued its Notice providing for a public hearing which was held on March 19, 1960 to consider the issuance of a construction permit for the utilization facility sought to be constructed by Consumers. This Notice was published in the FEDERAL REGISTER (25 FR 1699).

By order entered at the hearing, intervention was granted of the petition by the State of Michigan. Limited appearances, only, were requested by and granted to the Water Resources Commission of the State of Michigan and the Michigan State Department of Health.

[1] As of the date of the hearing, contracts had not been executed by the Atomic Energy Commission and Consumers, and its contractor for design of the nuclear facility; but negotiations were under way contemplating that General Electric Company would be reimbursed for costs to a specified limit in carrying out a research and development program and Consumers would be reimbursed in a lesser amount for its research and development work. In addition, Consumers seeks a waiver from the Commission of charges for the use of nuclear fuel. Consumers does not request any funds from the Commission in connection with the construction of the Big Rock Plant.

In addition to appearances on behalf of Consumers and the Staff of the Commission appearances were entered by the State of Michigan by its Assistant Attorney General and by the two Michigan Commissions by representatives thereof. No other persons sought to intervene or to otherwise participate in the proceeding. On April 1, 1960, Consumers submitted proposed findings and conclusions, and the Staff submitted similar proposals on April 8, which were accepted by Consumers on April 13, 1960. On this latter date, the State of Michigan filed a statement of its position, substantially to the effect that it and its agencies entered no objection to the issuance of the construction permit, but reserving the right to make statements or comments respecting any operating license, which is expected to be considered at a later time. The State of Michigan added: ". . . we expect the Applicant to comply with all applicable state laws and it has indicated to us that it desires to do so."

On April 22, 1960, Consumers filed a further statement relative to the position asserted by the State of Michigan, repeating its assurances: ". . . that the Company intends to cooperate fully with all interested State agencies in the course of construction and operation of the Big Rock Plant."

The Notice of Hearing issued by the Commission limited the consideration solely to the construction permit requested by Consumers, and specified the issues to be considered as follows:

1. Whether the Applicant has submitted sufficient information to provide reasonable assurance that a utilization facility of the general type proposed in the application can be constructed and operated at the proposed site without undue risk to the health and safety of the public;

2. Whether there is reasonable assurance that the technical information omitted from and required to complete the application will be supplied;

3. Whether the Applicant is technically qualified to design and construct the proposed facility;

4. Whether, pursuant to Section 50.40(b) of the AEC's regulations, the Applicant is financially qualified to engage in the proposed activities; and

5. Whether construction of the reactor will be inimical to the common defense and security or to the health and safety of the public.

The nuclear power reactor proposed by Consumers would be located at Big Rock Point, in Charlevoix County, Mich., between the towns of Charlevoix and Petosky on the northern shore of the lower peninsula of Michigan. The site is about 4 miles northeast of Charlevoix, which has a population of 2,700, and 11 miles west of Petosky, which has a population of 6,500. The site consists of approximately 600 acres and provides a minimum exclusion radius of about a half a mile. Access to the site is available by U. S. Michigan Route 31 and the Chesapeake & Ohio Railroad which pass near the site region at a distance of ½ to 1 mile.

The design of the nuclear system will be based on known technology which has been acquired in the development of plants such as the Dresden Nuclear Power Station, EBWR at Argonne National Laboratories, and VBWR at the Vallecitos Atomic Laboratory. In

this direct cycle forced circulation boiling water reactor, the energy is to be removed from the reactor core by the light water coolant and the resulting steam-water mixture flows to the steam drum and from there the steam flows directly to the turbine control valves. Water from the steam drum is returned to the reactor vessel by reactor coolant recirculating pumps. The steam flows through a conventional turbine cycle with a condenser, feed pumps, and extraction feed water heaters. Feed water is returned to the steam drum. A light water moderated and cooled reactor core is to be contained in a single reactor vessel. The fuel will consist of slightly enriched uranium oxide pellets encased in stainless steel tubing. The individual fuel rods are to be grouped into fuel assemblies. The entrance to each assembly is provided with an orifice which gives a means of adjusting the pattern of flow and steam generation within the reactor core. The assemblies are positioned vertically in the core. Control rods enter the bottom of the reactor core and are operated by a hydraulic system located beneath the reactor vessel. The control rods are used to control start up and to shut down the reactor, to control power, and to shape the neutron flux pattern.

The reactor building consists of a spherical steel containment vessel and a reinforced concrete structure therein. This building will house the nuclear steam supply system and auxiliaries, and facilities for storage and handling of new and spent nuclear fuel. The spherical reactor containment enclosure will be designed to withstand internal pressure of 27 psig and will be tested pneumatically at 1.25 of design pressure. The design pressure exceeds the 24 psig peak pressure that would result from the postulated "maximum credible accident" as most recently calculated by Consumers. Missile protection to the containment enclosure is provided by concrete shielding which shields the reactor tank. The reactor vessel containing the core will be approximately 32 feet long and 9 feet in diameter. Base material of the vessel will be ASTM A302 Grade B firebox and be clad with stainless steel type 304. The vessel will be designed, built and tested in accordance with the ASME Boiler and Pressure Vessel Code. The operating pressure of the vessel will be 1,500 pounds per square inch absolute. Access to the vessel internals and core will be through a removable head closure which will also contain ports. In addition to the core, the vessel will house the control rods and control rod drives as well as the required instrumentation.

The reactor vessel will also contain the nozzles for the risers to the drum, inlet nozzles from the recirculation pumps and nozzles for the core spray and poison systems. A thermal shield is provided within the vessel which prevents exposure of the vessel walls to excess fast neutron flux levels.

The single steam drum is provided to separate the steam generated in the reactor from the steam and water mixture. It also provides adequate storage capacity to accommodate surges between the reactor vessel and the drum during transient conditions and assures a positive supply to the recirculating pumps. It is located high enough to maintain natural circulation in case the recirculating pumps are inoperative. The drum will be a carbon steel or alloy steel pressure vessel with stainless steel cladding on all internal surfaces, and will be provided with moisture separators and driers and will be designed

and constructed in accordance with the ASME Boiler and Pressure Vessel Code.

The reactor core occupies approximately the volume of a right circular cylinder 75¾ inches high and 61.8 inches in diameter. It contains 120 square fuel assemblies and 32 control elements in a square array. The fuel assemblies are made up of 64 fuel rods in a square array positioned by spacers in the center and tie plates at the ends and having an active fuel length of 72 inches. Provision is made in design for variations in rod length rising from manufacturing variations or relative thermal expansion.

Each fuel assembly is surrounded by a fuel channel which adds to the stiffness of the assembly, guides the control elements and separates the coolant in the fuel chambers from that flowing around the control elements. This permits individual orificing of each fuel assembly. The fuel rods consist of stainless steel tubing containing fuel pellets of sintered uranium oxide 0.318 inch in diameter. The tubing has an outside diameter of 0.372 inch and a wall thickness of 0.024 inch. The ends of the tubing are plugged with stainless steel fittings which lock the rod in the spacers or tie plates. These fuel assemblies and fuel elements are designed for the higher power densities expected to be obtained in the Big Rock Point reactor. The higher power densities are gained by the use of smaller diameter rods which results in larger heat transfer surfaces and makes possible higher heat flux in the reactor without encroaching on normal fuel element burnout limitations.

The turbine building which houses the power extraction equipment is in many respects a conventional turbine facility. The position of the turbine with respect to the containment building is such that, in the event of a turbine failure, no missles generated would move in the direction of the enclosure. The turbine foundation is to be constructed sufficiently large and strong to support a shield wall, if shielding should prove advisable in the future. All auxiliary equipment which has significant radiation levels will be provided with shielding and in many cases with remotely controlled equipment. Special precautions will be taken in the ventilation of the building and all ventilation air is to be discharged through the monitored and controlled offgas discharge stack.

The 240-foot waste discharge stack will be used to carry all potentially radioactive gases from the plant to the atmosphere. Continuous monitoring will be provided on the stack and the design of the plant will include sufficient short-term decay facilities so that the significant quantities of short-lived activation product gases will be reduced to insignificant quantities (less than 100 microcuries per second) prior to release to the stack.

Equipment for treatment of liquid wastes by demineralization, evaporation, or storage is to be provided in a liquid waste disposal facility. Any release of liquid wastes will be performed by batch control following analysis and treatment to the extent required by each batch. Tank sizes are to be adequate to insure storage capacity for the proposed batch-type operation. Batches to be discharged will be diluted in the discharge canal at a controlled rate to insure that concentrations entering Lake Michigan are well within permissible limits.

No solid wastes will be discharged to the environs outside the site boundaries. Concrete vaults to be installed on the site will be used to store radioactive wastes until they can be removed by a licensed disposal contractor.

In the reactor proposed to be constructed by Consumers, as in other boiling water reactors, the temperature, void and doppler coefficients of reactivity are all negative in value and hence contribute significantly to the safety of the system as protection against power excursions.

Thirty-two control rods are to be provided which will enter the bottom of the reactor core and will be operated by a hydraulic system located beneath the reactor vessel. The control rods will enter the bottom of the core to achieve efficient fuel utilization and power distribution in the boiling water reactor. Inserting the rod upward into the reactor core decreases reactivity; withdrawal or lowering the rod increases reactivity. The control rod worth is to be sufficient to maintain the reactor subcritical in the cold clean condition and the failure of operation of any signal rod in scram operation will not impair the ability to shut down the reactor.

The control rod drive mechanisms are to be designed using a fail-safe philosophy such that loss of electric power or loss of air pressure acts to initiate reactor shut-down. Normal operation of the control rod drives will be by reactor feed water. Scram water is to be supplied from either the reactor vessel or an accumulator to insure that an emergency source is available for scram purposes in the event of low reactor pressure. An individual drive mechanism is used to position each of the control rods. All rods are scrammed for emergency shut down.

It is expected that the control rod drive mechanism proposed will operate satisfactorily in the reactor. It may be noted, however, that before final approval of the design need be made, additional operating information from the Dresden Nuclear Power Station, which has a similar system, will be available as well as data from tests now being conducted by General Electric Company. In view of the advisability of procuring additional data in this regard, the Staff of the Commission did not express its opinion regarding the control rod drive system pending the results of the foregoing operating experience and tests.

Movement of the control rods for power level control purposes will be manual, using controls in the main control room. Operation of the control rods for safety reasons is to be initiated by a reactor safety system consisting of two parallel safety channels. Each channel has its own separate power supply and sensing elements. Both are of fail-safe design throughout; that is, de-energizing will cause a trip and both must be tripped to cause a scram. In most cases, two or more sensing element contacts for a given condition are provided in series in each channel for greater safety. Sensors in this system are provided to monitor the following plant conditions: high neutron flux, high reactor pressure, low water level in the steam drum, simultaneous closure of turbine stop valve and primary bypass valve, closure of turbine stop valve and primary bypass valve, closure of steam back-up isolation valve, high enclosure pressure, short reactor period, low condenser vacuum, low water level in the reactor vessel and simultaneous closure of recirculation water line valves.

In addition to scramming the reactor, scram signals from this system will also perform other important safeguard actions. All scram signals will close the sphere ventilation isolation valves. Scram signals from either high enclosure pressure or low water level in the reactor vessel will close necessary isolation valves in lines penetrating the enclosure. Scram signals from closure of either the steam back-up sphere isolation valve or high reactor pressure will initiate emergency cooling by means of the emergency condenser.

As a back-up to the control rod system, there is a manually operated emergency shut-down control system by which sodium pentaborate can be injected to shut the reactor down in case of an emergency.

An emergency cooling system, shut-down cooling system and a core spray system, are available in case of failure of the primary coolant system. These are capable of removing the heat from the reactor for the duration of any emergency that may occur.

Neutron flux measurements are to be supplied to the safety system from an out-of-core neutron flux monitoring system. In-core flux monitoring is to be provided for use in conjunction with the research and development program. The Applicant has not reached a decision as to whether it is necessary or not to incorporate the in-core flux monitoring system in the reactor safety system. It remains to be demonstrated that the maximum power densities proposed for this reactor can be safely achieved without the use of in-core monitoring acting in connection with the reactor safety system. It is contemplated that experience with the Dresden reactor and its monitoring system should help to provide data helpful in this consideration. In any event, incorporation of the in-core flux monitoring system into the reactor safety system could be accomplished prior to operation without requiring major alterations in other features of the plant. It is the view of the Staff of the Commission that sufficient in-core instrumentation should be provided to detect the occurrence of hazardous conditions in the core. Since the plans presented by Consumers for in-core monitoring are not yet complete, the Staff judgment regarding in-core instrumentation has not been formulated.

Much detail engineering work on the plant remains to be done. Consumers proposes to supply additional technical information based on final design of the facility in the form of a Final Hazards Summary Report approximately September 1961. The main purpose of the program, to cover 7½ years, is to develop a boiling water reactor core capable of high power density combined with long fuel life and low fuel fabrication cost. Three years of the program will be carried out before the plant startup, and this portion of the program is actually already underway.

The second portion of the program will be carried out in the first 4½ years of plant operation, during which the experimental operation of the plant itself is planned.

The component parts of the research and development program can be broadly grouped into fuel development, nuclear development and the planning of procedures for plant performance.

The fuel development for high power density operation includes design and fabrication process development, irradiation testing in

the Vallecitos Boiling Water Reactor to 15,000 megawatt days per ton and postirradiation examination and full scale irradiation at the Big Rock Point plant.

In order to obtain information useful for large power reactors, the demonstration of average power densities up to 60 kilowatts per liter in limited core regions of the Big Rock Point plant will be one of the major objectives of the research and development program.

The nuclear development phase includes stability and transient performance studies, including improvement and application of analytical methods, control rod isolation tests in the plant and stability and rod change performance studies in the plant itself up to 1,500 pounds per square inch pressures. The nuclear development phase also includes out-of-pile burn out and hydraulic stability tests and in-pile performance studies on specially instrumented fuel elements.

Comprehensive hazards evaluation of the in-plant program will precede actual operation of the facility. To accomplish the objectives of the research and development program, the reactor will be started up with existing known boiling water technology, and the target operating conditions will be approached in a planned stepwise manner, with each step contingent on the result of the previous step.

Average power densities of 45 kilowatts per liter have not yet been demonstrated in other boiling water reactors, and the Staff of the Commission reserved judgment as to whether the expected power densities could be safely achieved. Witnesses for both Consumers and the Staff, however, expressed confidence that this objective of the program could be safely attained, through the proposed step wise approach contemplated in the developmental program outlined in this proceeding. The Staff indicated that, although the initial core is being designed for a power density of 45 kilowatts per liter, the pressure vessel is being sized to accommodate a larger core that can operate at power densities less than the proposed 45 kw/liter, comparable to those for which the Dresden and Humboldt Bay reactors have been designed, but still capable of developing a power level of 156 MW (thermal). According to Consumers, the advance in technology represented in this planned operation of a core at higher power densities is not being done without the ability to revert back to operating limits currently established, nor without in-pile test to prove that this high limit can be safely achieved. It is contemplated that all data needed for the determination that must be made respecting power density will be available when the operating status of this reactor is proposed in a subsequent hearing.

As a part of the Consumers' safeguard evaluation program, a study has been undertaken of a variety of normal and off-standard conditions from a safeguard viewpoint. The situations studied included abnormal conditions arising when the reactor is operating and control is held by an operator, conditions arising as a result of equipment malfunction and conditions arising as a result of operator error. The analysis included accidents in these categories or a reasonable combination of them. The analysis also included such incidents as earthquakes, fire, and floods.

The maximum credible accident which has been postulated for the Big Rock Point Plant is a coolant loss accident resulting from a rupture in the primary coolant system when the reactor system contains maximum internal energy in the coolant. An analysis of the consequences of this accident has been used as part of the basis for the design of the plant. The assumptions included failure of the core spray system which then results in fuel meltdown.

The radiological effects of the maximum credible accident have been evaluated and the major effects estimated are that no one in the environment of the plant site would, it is expected, receive as much as a roentgen of external exposure from the direct radiation from the radioactive vapors in the enclosure or from the "cloud" or fallout contamination from the leakage of radioactive vapors from the enclosure.

In addition, it is concluded, no one in the environment of the plant would receive more than about 10 rems internal exposure from radioactive material deposited in a critical organ. The highest calculated dose would be to the thyroid. The thyroid dose to persons with 4 hours of unprotected exposure in very limited areas under extreme combinations of unfavorable circumstances could amount to as much as 240 rems. The area affected by contamination would be largely a function of weather conditions after the accident. It is conceivable that the general population in a small area near the plant might have to be evacuated for a short period (up to several months) as a result of ground contamination. Only a few persons would be involved according to current population distribution patterns. Monitoring and possible confiscation of crops and milk might have to be resorted to over an area of up to about 2 square miles.

Consumers has submitted a preliminary study of the meteorological aspects of the proposed reactor at the site specified and is continuing its studies in this regard. Tentatively, it is considered that neither meteorology nor climatology appears adverse. Consumers plans to submit additional data on this aspect of its project from time to time, which will be further considered at the time of the request for an operating license. Prevailing winds are from the west and there are no significant population centers downwind for a distance of over 10 miles. Only scattered rural residences and a few small commercial facilities are found within several miles of the site. Other than the cities of Petosky and Charlevoix, there are no nearby significant industrial operations, except for a cement plant about 9 miles east of the site. Typical of most of the northern portion of the southern peninsula of Michigan, the general region of the site is an important summer vacation land; however, this summer occupancy is not an important factor within 2 miles of the site. Consumers' consultant on meteorology from the University of Michigan has concluded that the meteorology of the site region will produce no significant limitation on plant design and operation. The meteorological program to be undertaken by Consumers provides for the installation of instrumentation at the site and a detailed study of specific meteorological phenomena pertinent to reactor projects.

The geology and hydrology of the site have been investigated and it is indicated that plant site surface water drainage is directly to Lake Michigan. This feature gives rise to a substantial safety concern, and the plant will be designed to compensate for this problem.

Water for domestic use and sanitary purposes in the area is drawn either from Lake Michigan or underground wells. There are 30 wells

within 2 miles of the site and several communities have municipal water systems using Lake Michigan as a source of water. The lake in this area is also popular for recreational purposes. Water from Lake Michigan will be used as a source of feed water and condenser cooling water and the used cooling water will be returned to the Lake. Preliminary design plans of the plant includes plans for preventing contamination of the Lake or the underground water table. Liquid wastes from the reactor plant are to be held in retention tanks to permit decay of radioactivity and will be released to the Lake at controlled rates with appropriate dilution. Monitoring of such waste disposal will be made as well as periodic sampling of the Lake water at various locations. Surface run-off of accidental leakage of liquid wastes from any above-ground storage facility is to be controlled by diking. Flow of subsurface water will be retarded and any radioactivity spilled on to the ground would tend to be restrained in mobility by the low permeability and ion exchange properties of the surface soils. At no time will the level activity of intentionally discharged liquids exceed permissible limits. An environmental survey program is planned that includes periodic sampling of well water.

The ground water region encompassing the plant site is bounded by a small creek on the east and a relatively high region of land to the south between Lake Michigan and Lake Charlevoix. The site ground water drainage to Lake Michigan is through relatively impermeable overburden which is expected to have fair to good ion exchange properties.

The lake currents at the shore line are expected to be from east to west most of the time with significant diffusion of any liquid effluents into the main portion of Lake Michigan.

With respect to the diffusion of plant effluents into the Lake water, a program is proposed during the period prior to plant operation to determine more precisely the effect which lake currents would have on the diffusion rates and the resulting diffusion coefficients which could be expected under varying conditions.

There have been no significant earthquakes in this area in historic times. The plant design is based on Zone 1 requirements of the Uniform Building Code which establishes factors of safety which should be used for various areas depending on earthquake frequency. Zone 1 assumes a low level of earthquake experience.

The investigations of the various parameters, including meteorology, geology and hydrology and the population density and industrial activity of the area indicate that this location is satisfactory and that there can be expected no undue limitations to the operating characteristics of the plant placed on it by its surrounding environs.

It is concluded that the design of the plant will offer adequate protection against any hazard from radioactivity which may be discharged into the water and that the proposed site is suitable for location of the described reactor.

The Advisory Committee on Reactor Safeguards has reviewed the Applicant's Preliminary Hazards Summary Report and in its report to the Chairman of the Atomic Energy Commission, dated March 14, 1960, the ACRS stated that:

"Presupposing continued generally favorable experience with boiling water reactors, the Consumers Power Company proposes to conduct an unusual four and one-half year research and development program sponsored by the Atomic Energy Commission to explore the area of high power densities, optimum conditions for higher power density operation, longer fuel life, and reduced fuel fabrication costs. It is anticipated a power level of 240 MW thermal may eventually be obtained. The Committee recognizes that the detailed design and the experimental program will require additional study later.

"It is the opinion of the Advisory Committee on Reactor Safeguards that a boiling water reactor of this general type and power levels can be constructed at the proposed site with reasonable assurance that it may be operated without undue risk to the health and safety of the public."

Evidence was presented concerning the experience of General Electric Company respecting design and operation of test nuclear reactors, and of Bechtel Corporation respecting construction of various nuclear facilities, both of which organizations have been selected by Consumers for the project it proposes in this proceeding. It is concluded that Consumers, in conjunction with General Electric Company and Bechtel Corporation, is technically qualified to design and construct the proposed facility.

Consumers estimates that the total cost of construction of the Big Rock Point Plant will be $27,762,000, and the Commission's Staff believes the estimate to be reasonable. Consumers plans to finance the plant as an integral part of its normal construction program through the use of funds internally generated and from the sale of securities in the same general manner as other plant additions are financed. Consumers' general financing plan is to maintain the existing capitalization ratios between common stock, preferred stock and debt securities within ratios which are believed to be consistent with acceptable public utility financing criteria. Upon the basis of the present ability shown by Consumers to generate funds and the proposed general plans for financing, it is concluded that Consumers has demonstrated sufficient financial qualifications to undertake the construction permit sought in this proceeding.

In connection with its application for a construction permit, Consumers has requested an allocation by the Commission of 4,791.8 kilograms of contained U-235 to cover the fuel requirement of the reactor through a 40-year operating period. The financial data submitted by Consumers indicates sufficient ability of Consumers to be financially responsible for the maximum quantity of special nuclear material which is expected to be outstanding to Consumers at any one time.

The determination of the existence of reasonable assurance that the utilization facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public is limited by the present record to the conventional type of boiling water reactor with a power density of 30 kilowatts per liter. Many details of the design of Consumers' project remain to be completed. The Staff of the Commission has expressed the view that while it is believed that experiments will develop satisfactory designs, it remains to be demonstrated that the maximum specific power densities proposed can be safely achieved, and that clarification must be provided on the need for in-core instrumentation which may appear to be necessary as performance parameters are better understood. The adequacy of the control rod drive mechanisms may warrant re-examination at a later stage in the design calculations.

In addition to all of the foregoing constituting findings and conclusions, and based upon the entire record of this proceeding including the contentions and statements of the participants, and the proposed findings and conclusions which have been adopted, modified or rejected, as shown herein, the Presiding Officer hereby further *finds*:

1. Consumers Power Company, a corporation organized under the laws of the State of Maine, but engaged solely in business as a utility within the State of Michigan with principal offices at Jackson, is a qualified applicant to receive a construction permit provided by the Atomic Energy Act, as amended.

2. The nuclear reactor project proposed to be constructed by Consumers is a utilization facility within the meaning of Section 104(b) of the Atomic Energy Act, as amended.

3. There is sufficient information available to provide reasonable assurance that the utilization facility of the general type proposed in the application by Consumers can be constructed and operated at the location specified without endangering, and will provide adequate protection to, the health and safety of the public.

4. There is reasonable assurance that, in accordance with the research and development programs outlined in the evidence and the availability of alternate proven features to be incorporated in the reactor should necessity therefore require, the technical information omitted from the application will be supplied.

5. Consumers, General Electric Company, and Bechtel Corporation, collectively, are technically qualified to design, construct, and operate the proposed utilization facility.

6. Consumers is financially qualified and able to engage in the activities proposed in its application in accordance with the Atomic Energy Act, as amended, and the Rules and Regulations of the Commission.

7. The issuance of the construction permit in substantially the form attached hereto will not be inimical to the common defense and security, nor to the health and safety of the public.

WHEREFORE, IT IS ORDERED, subject to review by the Commission upon its own motion, or upon appeal after exceptions, if any, are filed:

A. The Director of the Division of Licensing and Regulation of the Atomic Energy Commission is directed to issue a provisional permit substantially in the form of Annex A made a part hereof.

B. The construction permit shall be provisional to the extent that a license authorizing operation of the facility will not be issued by the Commission unless Consumers has submitted to the Commission (by amendment to the application) the Final Hazards Summary Report (portions of which may be submitted and evaluated from time to time) and the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the specified procedures.

C. The Director of the Division of Licensing and Regulation of the Atomic Energy Commission is further directed to allocate or arrange for the allocation to Consumers, for use in the operation of the reactor 4,791.8 kilograms of uranium 235 contained in uranium in the isotopic ratios specified in Consumers' application.

D. Exceptions, if any, and briefs in support thereof must be filed by May 27, 1960, briefs in opposition thereto shall be filed by May 30, 1960, and if the Commission does not initiate a review on its own motion, and no exceptions are filed, this decision shall, in accordance with the Commission's Rules of Practice, become final on May 28, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

ANNEX A

DOCKET NO. 50-155

IN THE MATTER OF CONSUMERS POWER COMPANY

CONSTRUCTION PERMIT

Pursuant to the Atomic Energy Act of 1954, as amended (hereinafter referred to as "the Act") and Title 10, CFR, Part 50, "Licensing of Production and Utilization Facilities," the Atomic Energy Commission (hereinafter referred to as "the Commission") hereby issues a construction permit to Consumers Power Company to construct a 75 megawatt (electrical) high power density boiling water nuclear reactor as a utilization facility.

This permit shall be deemed to contain and be subject to the conditions specified in Sections 50.54 and 50.55 of said regulations; is subject to all applicable provisions of the Act, and rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. The earliest date for the completion of the facility is July 1, 1962. The latest date for the completion of the facility is July 1, 1963.

B. The site proposed for the facility is the location at Big Rock Point, Charlevoix County, Mich., described in the application.

C. This permit authorizes the applicant to construct the facility in accordance with the application and amendments thereto heretofore filed in this proceeding without further authorization.

D. If the applicant desires Commission approval of any particular technical specification, prior to consideration of a license to operate, it may request that the Commission grant specific approval of any technical specification by appropriate amendment to this permit.

E. This permit is provisional to the extent that a license authorizing operation of the facility will not be issued by the Commission unless Consumers Power Company has submitted to the Commission (by amendment to the application) the complete Final Hazards Summary Report (portions of which may be submitted and evaluated from time to time) and the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the specified procedures.

F. Upon the filing of any additional information needed to bring the original application up to date, upon filing of proof of financial protection as required by Section 170 of the Act and the Commission's regulations, and upon a finding by the Commission that construction of the facility has been completed in accordance with the final design as approved by the Commission, and that the facility has been constructed and will operate in conformity with the provisions of the Act and the Rules and Regulations of the Commission, and in the absence of any good cause shown to the Commission why the granting of a license would not be in accordance with the Act, the Commission will issue a Class 104 license to Consumers Power Company pursuant to Section 104(b) of the Act, which license shall expire forty years from the date of issuance of this permit.

G. Pursuant to Section 50.60 of the regulations in Title 10, Chapter 1, CFR, Part 50, the Commission has allocated to Consumers Power Company for use in the operation of the reactor 4,791.8 kilograms of uranium 235 contained in

uranium in the isotopic ratios specified in Consumers Power Company's application. Estimated schedules of special nuclear material transfers to Consumers Power Company and returns to the Commission are contained in Appendix IX of Part A of the application. Shipments by the Commission to Consumers Power Company in accordance with said Appendix IX will be conditioned upon Consumers Power Company's return to the Commission of material substantially in accordance with said Appendix IX.

FOR THE ATOMIC ENERGY COMMISSION,

Date of Issuance: ---

DOCKET NO. 50-10

IN THE MATTER OF COMMONWEALTH EDISON COMPANY

Issued May 16, 1960

APPEARANCES:

Isham, Lincoln, and Beale, Counselors at Law, by *Arthur C. Gehr, Esq.*, for Commonwealth Edison Company, and *William F. Kennedy, Esq.*, for General Electric Company.

Gerald Charnoff, Esq., for the Staff of the Atomic Energy Commission.

Harry Begley, Esq., Special Assistant Attorney General, for the Illinois Commerce Commission.

SECOND SUPPLEMENTAL INTERMEDIATE DECISION

Commonwealth Edison Company, Chicago, Ill. (Edison), filed a report on April 28, 1960, of test operations and results concerning its nuclear utilization facility generally described as the Dresden reactor. This report was filed pursuant to the provisions made in a Supplemental Intermediate Decision issued on November 12, 1959, which authorized test operations and limited power levels up to 315 MW (thermal). The report filed by Edison in April was submitted as a part of the basis for its request for an operating license to permit operation at the Dresden reactor to the full limit of the designed power. The Dresden reactor is the largest of its size proposed to this time with an anticipated ultimate capacity of producing an electrical output of 180,000 KW, corresponding to a reactor thermal power of approximately 630 megawatts. On April 1, 1960, further in accordance with the aforesaid Supplemental Intermediate Decision, an Order was entered providing for a public hearing on the matters to be presented by the report which Edison had theretofore indicated would be filed on April 28, 1960. Hearings were held on May 4 and 6, 1960. No additional appearances were entered than those which had appeared in the earlier hearings held in this proceeding. Evidence was presented by both Edison and the Staff, and in addition, a report of the Advisory Committee of Reactor Safeguards was received into the record.

Following the issuance of the basic Intermediate Decision in this proceeding on September 26, 1959, Edison undertook the nuclear fuel loading into the Dresden reactor and later achieved criticality [1] on October 15, 1959. On November 3, when Edison was analyzing the physics data taken in the 84 fuel assembly loading, it was discovered that one of the nuclear activity control rods had no apparent worth for control, and a visual inspection revealed that a poison blade had separated from its drive mechanism and remained in the

[1] Twenty-eight (28) fuel assemblies were used.

nuclear core when the drive was withdrawn. Edison stopped the nuclear activity in the reactor core which was then dispersed, and all 84 control rod drives were removed for inspection. A number of the seal rings required for proper buffering action at the end of the nuclear scram, or shut-down mechanism, were found to be cracked, and these damaged seals were replaced, and in addition, the pin arrangement was changed in an endeavor to avoid further pin shearing due to any failure of buffer action. On December 12, 1959, the nuclear core was reconstituted by fuel assembly loading, and it was then discovered that another poison blade, which is a part of the nuclear control rod drive, had also separated from its drive mechanism, as a result of the shearing of two pins therein. At this point, all operational activities at the Dresden reactor were suspended, and the nuclear fuel dispersed in order to permit Edison and General Electric to undertake a complete investigation of the control rod problems. Edison reported both of these incidents to the Commission and invited its review of all steps undertaken to resolve the difficulties. Edison made a complete analysis of the entire control rod mechanism and the defects[2] in prior assembly units and arrived at changes believed to be fully corrective. Prior to further nuclear loading of the reactor, Edison determined the effectiveness of the changes by extensive nonnuclear tests without any failure or indication of wear on the pins. Thereafter, nuclear fuel loading was made, and the tests at zero power outlined in the hearing in this proceeding in July and September 1959 were undertaken, followed later by further tests at various power levels, at specified step wise increases,[3] in which confirmation was achieved of the safety parameters of the reactor, and by which Edison attained the presently licensed limit of 315 MW thermal.[4]

From the evaluation made by Edison at this 315 megawatt level,[5] it has extrapolated those results to predict safe operation of this reactor at the full designed power level, but nevertheless, Edison proposes only step wise increases to further power at scales of 60 MW (thermal), or less, as outlined in the original hearings.

In further support of its application for authority to operate the Dresden reactor at power levels up to 630 megawatts, Edison offered evidence from personnel in charge of and responsible for the operation of Dresden and the conduct of the testing program. In addition to a summary of all tests and operations conducted since initial loading of the reactor commenced last October, the evidence gave particular emphasis to a review of the test results which had a bearing on the questions noted in the Supplemental Intermediate Decision respecting the stability, responsiveness to control and measurement of heat flux distribution. The testimony also reviewed in

[2] For instance, Edison's witness stated that the locking nut at the top and bottom of the fuel assembly had been "disturbed" (after reshipment from GE San Jose testing laboratory), and an inspection showed that the welding of those nuts was not done as effectively as it should have been done, so the welding process was repeated.

[3] The first nuclear heating of the reactor took place March 31, 1960.

[4] Edison's witness testified: "Radiation surveys during this period of testing indicated the need of certain additional shielding against scattered neutron and gamma radiations spread for higher power operations in order to meet the radiation limits specified in the preliminary hazards summary report."

[5] The Dresden reactor, at the time of the hearing, was shut down, and the reactor head was removed for the performance of certain gamma scan radiation tests, to check the flux profile of the reactor and for the addition of certain shielding and some equipment, including a shielding plug and new core pressure drop instrumentation. This process may take ". . . a matter of weeks." This decision is rendered at this time in order not to delay the planned program for the operation of the reactor.

considerable detail the inspection and tests of fuel cladding, the problems encountered with the separation of control rod drives from the poison blades, and a review of operational difficulties, both human[6] and mechanical.

During the initial critical testing of the reactor, control rod worths and flux distributions were investigated for various critical configurations of control rods. The principal purpose of this investigation was to determine an optimum control rod configuration to employ during the subsequent startup, nuclear heating, and power escalation phases of operation. Previous calculations had indicated that certain groupings of control rods were to be avoided in operation, since the worth of a rod in such a configuration might become excessive or the flux would be peaked excessively. The critical testing program at essentially zero reactor power confirmed these predictions.

It was further demonstrated in the testing program that the reactor could be made critical by withdrawal of a symmetrical and uniform pattern of rods with adequate observation of flux levels by the out-of-core instruments and, in addition, with such a control rod pattern, flux peaking problems and high worth groupings of rods in the core would be avoided.

The stability tests consisted primarily of rod cycling disturbances, steam drum pressure changes, load changes and noise measurements conducted over a wide range of secondary flows at 1,000 psi and up to 315 megawatts thermal power. These tests and measurements demonstrated that the reactor was stable in both time response and spatial flux distribution in all modes of operation at 1,000 psi and up to 50 percent rated power. All indications point to a continuing wide margin of stability at higher power operations and it appears highly unlikely that any reactor instabilities, other than xenon induced instabilities, will be observed at full power. Evidence of the spatial stability of the reactor flux distribution at half power is demonstrated by the fact that the whole core responded readily to changes made in any part of the core without any indication of flux waves which could be detected on a recorder having a 50-cycle per second band width. Although there was no evidence of transient xenon induced instabilities at half power, it appears that such instabilities may be observed at full power. The evidence was that the behavior and methods of control of xenon induced instabilities are well known, however.

Numerous transient tests were conducted at various power levels with the reactor producing mostly primary steam and also with dual cycle operation in order to test stability of the system and controls. Abrupt changes in steam pressure, steam flow, generator load, coolant flow and feed water temperatures showed excellent responses of the control system and no evidence of instability in either the reactor core or the reactor system.

The only variance in the tests results from predictions was the slightly negative slope rather than the moderate positive slope ex-

[6] Edison's witness stated: "At Dresden, a few human errors have occurred, although in my opinion, the performance of the operating people has been of uniformly high quality, as good or perhaps better than I have observed at any other installation over the seventeen years I have been in the business of reactor operation . . . of the human errors . . . none has resulted in hazards to plant operating personnel, plant equipment, and certainly not to the general public . . . people do make mistakes. It is inescapable, even the good ones."

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pected, for the dual cycle steam flowmap. System stability or safety is not affected by this difference in slope. The only consequence should such a negative slope still appear at rated conditions is some reduction in the expected range of automatic dual cycle control.

The license, in accordance with the report of the Advisory Committee on Reactor Safeguards, requires that for power operations above 315 megawatts thermal there be at least 32 in-core ion chambers operative and appropriately distributed throughout the core. During the hearings, preceding the issuance of the Supplemental Intermediate Decision, there were some evidence of doubt expressed concerning the ability and reliability of this system to perform its function of measuring the heat flux in various core regions. Tests and calibrations of the in-core monitoring system using in-core wire irradiations for base measurements show that of 64 ion chambers installed, at least 56 have been operating satisfactorily since April 9, 1960.[7]

With respect to the status of the remaining eight ion chambers, the testimony of an Edison witness is not in agreement[8] with the testimony of the witness from the AEC Inspection Division. This is possibly attributable to a variance at the date as of which each witness testified. In any case, both witnesses and the witness from Hazards Evaluation Branch concur that the in-core monitoring system has been performing satisfactorily and that an adequate number of monitors are now performing reliably. The long term dependability of the in-core ion chambers being used in the Dresden Nuclear Power Station remains to be established. Further experience as to the dependability of the ion chambers will be gained during the forthcoming rated power tests.

The peak heat flux measured at 315 megawatts was approximately 145,000 btu/ft^2-hr. This indicates that with expected activities of the core flux distribution, it will be possible to operate at a power level of about 630 megawatts within the peak heat flux limitation of 280,000 btu/ft^2-hr established in the technical specifications.

All of the fuel elements in use at Dresden were reinspected in December 1959 using newly developed inspection techniques which permitted the detection of minute cracks in the zircaloy cladding that did not extend all the way through the cladding. As a result of such reinspection, about 3 percent of the fuel rods were replaced. Tests were conducted at the Vallecitos Boiling Water Reactor using Dresden prototype elements in 1958 and 1959. Of 1,200 segments tested, three developed ruptures. None of these three fuel elements had been reinspected by the recently developed instrument used with the Dresden fuel. In addition, 13 segments in the Vallecitos reactor were intentionally ruptured by drilling holes through the

[7] General Electric's witness testified that the Dresden reactor had operated for a sufficient period of time to determine all of the nuclear characteristics of the reactor to 315 MW (thermal).

[8] The difference is largely in the count of the number of operating chambers. Edison's witness testified: "In the beginning when the chambers were first installed, I think we had trouble with one or two strings of chambers that were not reading properly and these were replaced. . . . There are four chambers in any one string or assembly. Of the 16 assemblies, there were two not working properly in the beginning when we went to the first power level and shut down for some shielding additions referred to earlier, these two strings were replaced. Since then, presently we have four chambers that are not operating . . . two that had shorted and two had open circuits . . . the present inoperative condition of four of the 60 chambers."

cladding. These tests established[9] that the fuel operates satisfactorily in excess of the peak heat fluxes permitted for operation at Dresden. The VBWR tests also demonstrated that under actual operating conditions a boiling water reactor can tolerate a reasonable number of ruptured elements, that carryover of fission products does not create a serious problem, and that maintenance following shutdown is not limited by contamination.

Off-gases have been monitored and these measurements indicate that the radioactivity of the fission products in these gases is generally less than 100 microcuries per second which may be compared to the license limitation or allowance of 700,000 microcuries per second. These trace amounts indicate a very low level of tramp uranium (uranium that might have been left on the outside of the fuel element) in the system, and support the conclusion that the integrity of the cladding of the fuel elements has been maintained.

While it appears that ruptures in fuel cladding is a matter of concern with respect to all reactors, safeguards have been incorporated in the Dresden design and license to prevent the release of an undue amount of radioactivity from this source. Because of such safeguards at Dresden, the concern with the integrity of the fuel cladding appears at the present time to be primarily economic.

Additional shielding was installed to confine the radioactivity to local areas in the reactor, where access is not required during reactor operations, and some small amount of additional shielding is to be installed during the current shutdown.

In addition to the foregoing tests and changes made since the date of first criticality of the reactor, and further in reference to the changes made in the control rod drives, the modifications of five control rod drives were made, and proven out by operating each of the five control rods through 500 scram cycles without evidence of recurrence of the difficulty. The corrective measures were reported to the Commission, together with an anlysis of the safeguards against a "stuck-rod" incident. On the basis of these reports and subsequent successful control rod operation, the Applicant, the Hazards Evaluation Branch, and the Advisory Committee on Reactor Safeguards have concurred that causes of separation appear to have been solved. There is also agreement that in the unlikely event a poison blade becomes separated from the drive mechanism and sticks in the core, and then falls out unexpectedly, no significant hazard to the health and safety of the public would result.

Edison will observe check procedures designed to reveal conditions which may lead to separation of a control rod from its drive mechanism. These procedures include the performance of a pull test on each blade to assure that each blade and its drive mechanism is properly connected prior to the next startup of the reactor.

One of Edison's witnesses, who is manager of the Dresden site operations, and who has had extensive experience in the nuclear field since 1943, with 16 years of reactor operating experience during which he directed the startup of several reactor facilities, analyzed the operating experience at Dresden during the test program to 315 megawatts thermal power level. His analysis included a discussion of the approximate total of 50 scrams, or automatic shutdowns, of

[9] The evidence did not include reference to the characteristics of the 1,184 fuel elements constituting the balance of those tested at Vallecitos.

the reactor. This witness classified the scrams into two categories: as "spurious" and "real" scrams in reference to both equipment malfunctions, and the human errors which were encountered. In the instance of each scram, the safety system proved to be operable and effectively prevented any hazard to the public and plant personnel. Such experiences serve to demonstrate the successful operation of the fail-safe design of the safety system and that the safety equipment functions as intended to safeguard the operation personnel, plant and the public generally. Instances in which the real [10] scrams occurred were:

December 12, 1959—Investigation revealed that the scram resulted from tripping of one of the micro-micro ammeter safety channels with reactor on a rising period although the period was not sufficiently fast to cause a spurious scram. This scram was attributed to operator error.

April 13, 1960—Through error, a control rod in the reactor was not fully inserted at the timing checks of the control drive speed.

April 17, 1960—The linkage in one of the regulating valves supplying a part of the water for normal movement of the control rod drives broke and a number of the control rods drifted into the reactor toward the fully inserted or safe condition. This causes the power level of the reactor to decrease rapidly. Edison stated that no significant hazard was involved.

While the foregoing were classified as "real" scrams, the spurious scrams included the following:

March 7, 1960—Testing was in progress and there was an unplanned placement of a void assembly in a water hole, making the reactivity in that local area of the core higher than had been planned. It was stated that no hazard was involved.

October 14, 1959—Prior to achieving criticality, there was apparently an error in the computation of needed number of fuel assemblies. Edison's witness stated that: ". . . geometric factors explained why the remaining channels did not accurately predict criticality. Four fuel assemblies were added to the core. Subsequent analysis revealed that the four were indeed less than one half of the number required to reach criticality. No hazard was involved."

In addition, Edison's witness stated that some human errors developed certain problems which have been recorded in log books at the Dresden site and: "The AEC Inspection Division people who periodically visit Dresden have access to all records, therefore, probably are aware that such difficulties do occur." Some of these problems included (1) the movement of more than one control rod in the reactor, at different times, due to difficulties which have arisen in connection with the use of solenoid valves and, (2) pressure reducing valves in the water supply system have evidenced a number of operational difficulties. Edison's witness concluded with the statement that: ". . . this review of human errors and equipment failures . . . is only fair to state again that human errors have been few and equipment failures rare, considering the size and complexity of the plant. In no instance has human error or equipment failure resulted in conditions hazardous to plant operating personnel, plant equipment, or to the public."

In each instance of the real scrams the safety system performed its designed function, and corrective measures were taken. Edison stated that the occurrence of "spurious" or "nuisance" scrams was anticipated and the number was not unexpected nor excessive. Because of the importance of avoiding unnecessary shutdowns in the

[10] The word "real" is used to distinguish scrams from those arising from spurious signals.

generation of electricity, and in view of the anticipated occurrence of "spurious" or "nuisance" scrams,[11] the plant design provides for the installation of some sensoring devices in coincidence. In this connection, it will be noted that the ACRS letter of May 18, 1959 stated: "All of these ion chambers will be connected to the scram circuit and may be paired in coincidence circuits to reduce "spurious scrams."

Witnesses on behalf of both Edison and the Staff testified that the training program of Edison personnel in station operation would not be complete until some period of time after operation of Dresden at rated power had been achieved. In this connection, there was evidence that direction of plant operations would not be turned over by General Electric to Edison until such training program had been completed to the satisfaction of both parties. In any case, counsel for Edison stipulated that Edison would not accept direction of plant operations without prior written approval by the Commission.

The Advisory Committee on Reactor Safeguards has reviewed the reports of operations of the Dresden Nuclear Power Station up to 315 MW (thermal), or to half power and has reported as follows:

"It is the consensus of the Committee that the initial difficulty with the control system appears to have been solved. Although the possibility of future difficulties with this system cannot be ruled out, it does not appear likely that such difficulties, if they should occur, will create any significant hazards to the health and safety of the public.

"The Committee sees nothing in the reported Dresden operational experience up to half power that would preclude continuing the approach to full power operation under carefully planned test and operational procedures which will assure a prompt shutdown of the reactor in the event of significant departures from expected performance of the reactor or its control equipment. The step-by-step procedures proposed by the Applicant are in agreement with this concept.

"The Committee wishes to emphasize that the Dresden Power Plant operations for some time to come must be considered to be developmental in nature. The large size of the reactor and the relatively novel aspects of some of its control features go considerably beyond prior experience with boiling water reactors. Further operations should be conducted with this firmly in mind, until such a time as there has been a substantial period of satisfactory operation at full power, the Committee advises that visitor access to the plant be restricted to those having an official reason for such visits."

In reference to the visitor access reference in the ACRS letter, the record shows that during the construction period Edison permitted

[11] A member of the AEC Inspection Division testified he made a series of inspections of the Dresden reactor from October 1959 through April 1960. His testimony was largely of two categories: (1) that inspections had been made, "records . . . were reviewed as well as other operating logs."; and (2) opinions of compliance. No detailed testimony was given of what was discovered upon the occasions of the inspections. This inspector witness had not analyzed all of the approximately, "or less than", 50 scram incidents which occurred to the reactor, although he did analyze the "real" scrams. He stated: "I have no plans to analyze the 50 scrams if that is what the number is." He further testified that he had no specific instructions as to the scope of his inspection, and that inspection work was largely a matter of personal judgment. He was guided, he said, by (a) the amount of time he had available, (b) his opinion . . . organization's ability to perform safely, (c) his opinion of the adequacy of the design of the facility inspected. The AEC Hazards Evaluation Branch witness stated he ordinarily would prefer to have an independent audit of all reactor scrams. He testified as follows: "I would think that for the AEC to fulfill its responsibilities and encompassed in my evaluation of what has been taking place here, I would think it valuable for some one from the AEC to perform an independent check on the operation, including such review of such things as the spurious scrams. I have every reason to believe that the Inspection Division of the Atomic Energy Commission is doing just this. . . . I would like to have a report by a trained individual conducting an independent audit." He concluded by an affirmative to the question: "The fact that you have not received the reports or the details of these spurious scrams, Dr. Booth, notwithstanding the fact that you have not received these reports, you are still quite confident that this reactor can be safely operated up through the rated power test in a steady state operation."

large numbers of visitors on the site for guided tours of the noncritical facility. This policy was changed when initial fuel loading for criticality commenced, and since that time visitations have been restricted to 12 to 15 per month on the average. Edison stated that most of the visitors since the time of initial loading have been guests of the AEC and the State Department, but there were also invitees of General Electric for business reasons. Edison, itself, included some persons as visitors whose presence may not have been directly related to the operations. The Manager of the Dresden Site Operations stated that the number of visitors had not, in his opinion, been large enough to disturb or interfere with operation of the plant.

Because of the novel aspects of the reactor control features, the lack of evidence as to the long term dependability of the in-core ion chambers, the design power level of 630 megawatts (thermal), and the fact that the program to train Edison personnel in safe operation of the Dresden Nuclear Power Station, is not now complete and will not be complete, until some period of time after conclusion of the rated power tests, the Staff has proposed the issuance of an amendment to License No. DPR–2 which would authorize the conduct of the rated power tests and steady-state operation at 630 megawatts (thermal) in accordance with technical specifications attached to the license. The Staff has proposed that the results of the rated power tests and the steady-state operations at full power, as well as the status of the training program, be considered at a further public hearing to be held after completion of the rated power tests and prior to turnover by General Electric Company of the direction of the station operations to Edison. At that hearing, the Staff proposed that consideration be given to the issuance of an operating license for the full term of years requested by Edison. While Edison agreed that it would not accept full responsibility for the Dresden operations, and thus release General Electric from that direction, until the Commission had approved such a change in operation, Edison stated that it did not desire and therefore opposed a public hearing concerning the rated power tests proposed to be undertaken beyond the 315 MW (thermal) level, the steady-state operations at full power, and the training program. Edison, however, expressly emphasized that if a hearing were deemed advisable by the Commission, Edison would cooperate in every way in such a proceeding and it implied it would make its presentation in the most expeditious manner possible.

In the determination of matters of safety, all data that can be assembled should be presented, and a public hearing permits a complete review by many interested persons of all aspects that enter into that consideration. It is concluded that a public hearing is required in this proceeding respecting the foregoing items, and any other aspects that may develop, either in the particular operations at Dresden, or in any respect related to such Dresden operations. In *Westinghouse Electric Company* (WTR), Docket No. 50–22, the Commission stated:

> ". . . before a final operating license can be issued by this Commission all substantial and material facts pertaining to the existing status of the facilities proposed to be licensed and the proposed operation, including particularly safety, must be developed in full upon the record."*

*(Ed.) *See* Order Reopening Proceedings, May 6, 1959, p. 118, *supra*.

The results of the rated power tests, particularly as they bear on the dependability of the in-core ion chambers and the stability of the reactor at full power operation, and the satisfactory completion of the training program are "substantial and material facts pertaining to . . . safety" and accordingly should be developed in full upon a record prepared in a public hearing.

In addition to all of the foregoing constituting findings and conclusions, and based upon the entire record in this proceeding including all proposals and statements of the participants, which have been adopted, modified, or rejected, as shown herein, the Presiding Officer hereby further finds as follows:

1. Commonwealth Edison Company, licensee herein pursuant to License No. DPR–2, has duly filed its report concerning tests, characteristics, operations, and attainment of a power level of 315 MW (thermal) at its Dresden nuclear utilization reactor facility in accordance and in compliance with the First Supplemental Intermediate Decision in this proceeding issued November 12, 1959.

2. Due and sufficient notice has been given of the filing on April 28, 1960, of such report by Edison, and a duly convened public hearing has been held to consider all aspects of such report and the presence or absence of any good cause concerning the grant of further authority for operations to the fully designed power level of the Dresden reactor and for the requested term of years.

3. Good cause exists to grant authority to Edison to increase its power level operations of its Dresden reactor to 630 MW (thermal) in accordance with the terms and conditions of the findings herein and the order of this decision, and good cause further requires that Edison shall not operate the Dresden reactor in excess of or at a steady-state power level of 630 MW (thermal) without further order of the Commission.

4. License No. DPR–2 should be further amended by deleting: "and subject to the Order of the Commission in reference to operations of the nuclear facility at the 315 megawatt (thermal) power level" from paragraph 2 thereof, and by deleting paragraphs 5, 6, and 7 thereof in their entirety and substituting therefore a new paragraph 5 to read as follows:

> "5. This license shall become effective as of the date of issuance and shall expire upon issuance of a final decision by the Commission following a public hearing to be held respecting, and within 90 days after filing of, the report of rated power tests, required to be filed under paragraph 3. c. (2) of this license. Said report shall also include a description of the status of the program designed to train the Commonwealth Edison personnel in the direction and operation of the Dresden Nuclear Reactor Utilization Facility. Such hearing will be held upon 15 days' notice to the public and no later than 90 days following the filing of the aforesaid written report of rated power tests by Commonwealth Edison Company with the Secretary of the Commission."

5. In accordance with the terms and provisions of the Supplemental Intermediate Decision issued November 12, 1959, and with good cause present this decision and order herein should become immediately effective subject to:

A. The review hereof and further decision by the Commission upon exceptions hereto filed by any party within twenty days hereafter, pursuant to the Commission's Rules of Practice, and,

B. Such further order as the Commission may enter upon such exceptions or upon its own motion within 45 days after the issuance of this decision.

WHEREFORE, IT IS ORDERED, in accordance with the terms and provisions of the Supplemental Intermediate Decision issued November 12, 1959, and further in accordance with the Rules of Practice of the Commission:

A. Commonwealth Edison Company, Chicago, Ill., is granted authority and amendment to its existing License No. DPR-2 to authorize power operations at its Dresden nuclear reactor utilization facility to a power level of 630 MW (thermal) in accordance with the findings and provisions of this decision and order, but shall not operate the aforesaid Dresden facility in excess of or at a steady-state power level of 630 MW (thermal) without further order of the Commission.

B. This decision and order is effective immediately subject to:

1. Review hereof and further decision by the Commission upon exceptions hereto filed by any party within twenty days or on or before June 6, 1960, objections, with briefs thereto, filed by June 8, 1960, pursuant to the Commission's Rules of Practice.

2. Such further order as the Commission may enter upon such exceptions or upon its own motion at any time within 45 days, or June 30, 1960, after the issuance of this decision.

C. The Director of the Division of Licensing and Regulation shall forthwith issue an amendment to Edison's License DPR-2, to insert: "as revised August 12, 1959" after "February 6, 1959", and to insert "and June 17, 1959" after "May 15, 1959" and delete "and" which appears before "May 15, 1959", and in accordance with the findings and conditions of this Order, to amend said License by including specifically the amendment that:

> "This License DPR-2 shall become effective as of the date of issuance and shall expire upon issuance of a final decision by the Commission following a public hearing to be held respecting, and within 90 days after filing of, the report of rated power tests, required to be filed under paragraph 3. c. (2) of this License. Said report shall include a description of the status of the program designed to train the Commonwealth Edison Company personnel in the direction and operation of the Dresden nuclear reactor utilization facility. Such hearing will be held upon 15 days notice to the public and no later than 90 days following the filing of the aforesaid written report of rated power tests by Edison with the Secretary of the Commission.

SAMUEL W. JENSCH,
Presiding Officer.

DOCKET NO. 50-29

IN THE MATTER OF YANKEE ATOMIC ELECTRIC COMPANY

Issued May 18, 1960

ORDER

At a session of the Atomic Energy Commission held at Washington, D.C., on the 18th day of May 1960, Commissioners John S. Graham, John F. Floberg, John H. Williams and Robert E. Wilson present, upon consideration of a motion of Yankee Atomic Electric Company in the above-entitled docket dated May 11, 1960, which was certified to the Commission by order of the Hearing Examiner dated May 16, 1960, pursuant to Section 2.748 of the Commission's Rules of Practice, it is hereby ORDERED that:

In the discretion of the Presiding Officer any intermediate decision and order for the issuance if any, of a license, provisional or otherwise, in the proceeding in this docket initiated by Notice of Hearing dated April 21, 1960, may provide that such decision and order shall become effective immediately upon issuance, subject to (1) the review thereof and further decision by the Commission upon exceptions thereto filed by any party within twenty (20) days after issuance of such intermediate decision, pursuant to the Commission's Rules of Practice, and (2) such further order as the Commission may enter upon such exceptions or upon its own motion within forty-five (45) days after the issuance of such intermediate decision; *provided, however*, that in the absence of any further Commission order pursuant to the foregoing, the intermediate decision of the Presiding Officer shall become the final decision of the Commission at the end of such 45-day period.

ATOMIC ENERGY COMMISSION.
By WOODFORD B. MCCOOL, *Secretary.*

BYPRODUCT MATERIAL LICENSES No. 4-1695-2B59 AND No. 4-5744-1

IN THE MATTER OF ADVANCE INDUSTRIAL X-RAY LABORATORIES, A DIVISION OF AIR FRAME INSPECTION, INCORPORATED, RESPONDENT, AND ADVANCE INDUSTRIAL X-RAY LABORATORIES, INC., APPLICANT

Issued May 20, 1960

ORDER EXTENDING TEMPORARY LICENSE

By telegram dated May 19, 1960, Advance Industrial X-Ray Laboratories, Inc. filed their request for an additional extension of 30-days of the temporary byproduct material license heretofore issued by the Commission following a hearing held on September 16 and 17, 1959. In the request for extension, Advance stated that additional time is necessary for the Commission to examine documents submitted by the applicant on May 13, 1960 in connection with the securing of one-year license from the Commission. The staff of the Commission has indicated that it has no objection to the grant of a 30-day extension.

The Presiding Officer *finds*:

1. Good cause has been presented by Advance for the grant of a 30-day extension of the temporary license, pending examination by the Staff of the Commission of information submitted by Advance.

The Presiding Officer *orders*:

A. An extension of time is granted and the term of the license issued to Advance is extended from May 16, 1960 to June 16, 1960.

SAMUEL W. JENSCH,
Presiding Officer.

DOCKET No. PP-3*

IN THE MATTER OF CONSUMERS PUBLIC POWER DISTRICT POWER DEMONSTRATION REACTOR PROJECT

Issued June 6, 1960

APPEARANCES:

John A. Roscia, Esq. and *William L. Clark, Esq.*, for North American Aviation, Inc.

Richard D. Wilson, Esq. and *Clarence A. Davis, Esq.*, for Consumers Public Power District.

John W. Merryman, Esq., for Bechtel Corporation.

W. H. Wright, Esq., for Peter Kiewit Sons', Inc.

Maurice Axelrad, Esq., for Staff of the Atomic Energy Commission.

INTERMEDIATE DECISION

Consumers Public Power District with headquarters at Columbus, Nebr., (Consumers) has executed a contract with the Atomic Energy Commission whereby the latter will construct and authorize Consumers to operate a nuclear power demonstration reactor for purposes of electric power generation. The reactor is proposed to be located at Hallam, Nebraska, which is located approximately nineteen miles south of Lincoln. The reactor is a 240-megawatt (thermal) facility, (often referred to as HNPF as an abbreviation for Hallam Nuclear Power Facility) which is sodium cooled, graphite moderated and uses uranium alloyed with molybdenum as a fuel.[1] The Commission has proposed this reactor as a part of its announced program to demonstrate the usefulness of nuclear facilities for purposes of electrical power generation. The Commission has executed separate contracts with North American Aviation, Inc. through its Atomics International Division [2] (AI), and with Bechtel Corporation [3] for the construction of designated portions of the nuclear plant. The Commission has executed a contract with Peter Kiewit Sons', Inc. of Omaha, for construction of the portions of the plant not constructed by North American and Bechtel Corporation.

On March 4, 1960, the Commission issued its Notice providing for a hearing on the safety aspects of the proposed nuclear facility,

*(Ed.) Docket number later changed to 115-3.

[1] The original plans for HNPF included using UO_2 as the reactor fuel, but in the latter part of 1958, it was decided that the initial fuel loading in the reactor would be uranium alloyed with molybdenum as a fuel.

[2] The Atomic Energy Commission authorized Atomics International to proceed under its existing general research contract for the Hallam facility. The exact dollar amount of the cost of the AI work for this facility is not readily discernible in this record. By an amendment to the existing contract, known as Modification No. 1, dated November 1958, the Commission agreed to a fixed fee of $360,000 to AI based upon an estimated cost of $13,587,771 for the Hallam facility. It is understood that the estimated cost is subject to revision. By Supplement No. 5 to Modification No. 1 additional funds in the amount of $6,373,490 were provided for carrying out the work of the contract.

[3] This contract is for architect-engineer services and provides for payment of actual costs and a fee for services rendered.

which was held on April 8, 1960 at Germantown, Md. The Commission prescribed the following issues for consideration:

1. Whether there is information sufficient to provide reasonable assurance that a nuclear reactor of the general type proposed can be constructed and operated at the proposed location within undue risk to the health and safety of the public;

2. Whether there is reasonable assurance that technical information required to complete the safety analysis of this facility will be supplied;

3. Whether North American Aviation, Inc., Peter Kiewit Sons', Inc. and Bechtel Corporation are technically qualified to carry out their responsibilities in the design and construction of the reactor; and

4. Whether construction of the reactor will be inimical to the common defense and security or to the health and safety of the public.

Appearances were made at the hearing by North American Aviation, Inc., Bechtel Corporation, Peter Kiewit Sons', Inc., Consumers Public Power District,[4] and by the Staff of the Commission. No persons sought to intervene or to otherwise participate in the proceeding. Proposed findings and conclusions were filed by the Staff on April 29, 1960, and the consent to such Staff proposals by the other participants was received on May 9, 1960. Also, on April 29, 1960, the Staff made a motion to incorporate into the record of the proceeding an affidavit of clarification of testimony from a staff witness. No objection was filed by any of the participants to this procedure. The affidavit does not purport to constitute new or additional evidence, but is limited solely to matters further explanatory of testimony already given. Without establishing any precedent for such procedure, the Staff motion is granted to permit the affidavit to be made a part of this record.

The contract between Consumers and the Commission provides as heretofore indicated, that the Commission will furnish an operable reactor,[5] while Consumers will furnish a site for the reactor, and will operate and maintain the reactor as well as furnish certain materials, equipment, facilities, and services.[6]

The site for the HNPF is in a sparsely populated part of Lancaster County, which is in southeastern Nebraska. Farms and small communities characterize the area within several miles of the site. Consumers owns an entire mile square section upon which the reactor will be located, slightly southeast of the section center. The nearest site boundary is about one-quarter mile from the reactor. A railroad right-of-way passes through one edge of the site about 800 feet from the reactor. The nearest major highway is about 4 miles to the east of the site. There are no inhabited dwellings

[4] In accordance with the Notice of Hearing and Rules of Practice of the Commission, answers were filed respecting the prescribed issues by the participants, except the Staff.

[5] The Atomic Energy Commission agreed to bear the cost of preconstruction research and development in the established amount of $18,000,000 and postconstruction costs estimated at $8,000,000, and covering some fuel costs and extraordinary maintenance. This contract provides that the primary responsibility for the Hallam Nuclear Power Facility rests with the Atomic Energy Commission.

[6] Specifically, the Commission has agreed to furnish to Consumers a sodium-cooled graphite-moderated reactor designed to produce 710,000 pounds of steam per hour at a temperature and pressure adequate to produce 75,000 net electric kw of power, and Consumers will operate and maintain, after full power has been attained, both the turbo generator and reactor facilities.

within ½ mile of the reactor. The nearest population center is the village of Hallam, population 172, about 1½ miles south of the site. The nearest large population centers are Lincoln (population 115,000), 19 miles to the north, and Beatrice (population 13,000), 19 miles to the south. The population density within 10 miles of the site is about 21.12 persons per square mile.

The site for the reactor facility was selected by first considering the general area in which an additional electric power supply is required. Next, the site was selected upon the basis of the existence of an adequate water supply. Finally were considered the influence of population distribution, hydrological characteristics, climatology, and seismological characteristics so as to minimize the radiation exposure to the public should a radioactive incident occur.

The terrain at the site is characterized by the rolling hills and valleys typical of the Loess Sections of the Great Plains region of the United States. Ground elevation at the reactor facility is approximately 1,440 feet above sea level. The site is near the high point of terrain which gradually recedes to the north and east to form the valley of the Platte River.

Surface drainage from the immediate area of the site is controlled by retention dams erected for soil conservation purposes, and flows northerly into Salt Creek, a tributary of the Platte River. Drainage water from the area, insofar as is known, is not used for industrial or domestic purposes for approximately 250 miles downstream.[7] Water that enters the soil at the site would move very slowly through a 200-foot surface layer of glacial clay to a gravel layer, and once in the gravel layer would move laterally in a generally east-southeast direction at an estimated rate of only 300 feet per year. Therefore, any water percolating into the soil at the site would appear to have ample opportunity for decay and dilution before it might be withdrawn for domestic purposes.

At Hallam there are only infrequent unfavorable atmospheric diffusion conditions, except for the nocturnal inversions, which do not persist for long periods. In the general area of the site, the prevailing winds are from the northwest during the winter months and from the southeast during the summer months. On the average, Lincoln is exposed to winds from the direction of the reactor site 5.5 percent of the year, Crete 4.3 percent of the year, and Beatrice 8 percent of the year. Tornadoes occur occasionally in eastern Nebraska. The Hallam area of southeast Nebraska comes within the definition of Seismic Zone I of the Uniform Building Code, which is defined as an area where infrequent seismic disturbance may be expected to cause minor damage. The site for the HNPF has many favorable features and is believed to be an advantageous location for a nuclear reactor.

As stated, the HNPF will be a 240 MWT (75 MWE) sodium-cooled graphite-moderated reactor fueled with stainless steel clad elements containing slightly enriched uranium—10 percent molybdenum alloy.[8] The reactor core consists of the fuel elements sus-

[7] The State of Nebraska Department of Health believes that there are no persons, communities, or industries using any of the streams draining the reactor site for either domestic or industrial purposes for approximately 250 miles downstream of the site.

[8] The Hallam facility is represented as reflecting advanced design which produces steam pressure, temperatures and plant efficiences exceeding those of contemporary reactor systems.

pended in a closely packed array of canned graphite moderator elements, scalloped at the corners so that each three adjoining elements form a channel between them which runs axially through the core. The fuel elements and control rods are suspended in these channels and the main sodium flow through the core is upward through them. Each fuel element consists of a cluster of 19 fuel rods held vertically in a zirconium alloy process tube. Each fuel rod consists of the fuel material in a column enclosed in a thin walled, stainless steel jacket. The thin gap between the fuel material and the jacket is filled with sodium. A ½-inch layer of sodium covers the fuel material at the top and above this is a helium volume provided to contain fission product gases which might diffuse out of the fuel as it becomes irradiated.

The reactor core assembly is contained in a stainless steel reactor vessel of welded construction. At the top, a stainless steel bellows assembly seals the reactor vessel to the upper cavity liner to prevent sodium vapors and the helium atmosphere from escaping, at the same time allowing for thermal expansion and contraction of the reactor vessel. The reactor vessel is surrounded by the thermal shield, the function of which is to attenuate gamma and neutron radiation and thus to prevent the concrete biological shield from overheating. The reactor outer vessel is a carbon steel tank surrounding the thermal shield and covered on the outside by thermal insulating material. The outer vessel is located in the reactor cavity which is completely enclosed by a concrete biological shield with a gas tight lining forming the secondary gas confinement barrier. A loading face shield in the form of a dense concrete cylindrical plug encased in steel is located directly above the reactor core.

The sodium heat transfer system consists of three independent circuits each directly connected to the reactor vessel. Each circuit consists of a radioactive primary sodium loop which transfers thermal energy from the core to an intermediate heat exchanger and a nonradioactive secondary sodium loop which carries the heat from the heat exchanger to a steam generator. This arrangement separates the steam system from the radioactive primary system. Three steam generators are provided, one for each of the secondary sodium loops. Each consists of an evaporator, moisture eliminator and superheater. Construction is of the shell-and-tube type utilizing duplex tubes to prevent interleakage between sodium and water or steam. A third fluid system using helium is provided to monitor the duplex tubes for leaks. The helium pressure is set at a pressure intermediate of those of the sodium and steam systems so that a suspected leak would be evidenced by a change of pressure in the helium system. Tests performed at MSA Research Corporation [9] on a model of these steam generators demonstrated that the gas monitoring system operated effectively.

The control rod system is provided to regulate the reactor power level and to effect emergency setback or shutdown when required. The rods can be operated automatically by the plant control and protective system or manually by the operator at the control console. The control rods have a rate-limited motion to ensure that

[9] A subsidiary of Mines Safety Appliance Company.

excessive reactivity cannot be inserted by erroneous action. In the event that other protective system action cannot adequately maintain plant safety, gravity fall of the control rods into the reactor core will provide scram action.

The reactor instrumentation can be divided into nuclear instrumentation, sodium instrumentation, plant control system, plant protective system, and radiation monitoring system. The nuclear instrumentation circuits measure and indicate, or record, neutron flux over the entire operating range of the reactor, starting from source level. The sodium instrumentation measures and indicates, or records, pressure flows, temperatures, and liquid levels in the sodium systems. The plant control system operates the reactor plant (by adjusting control rods, pump speeds, etc.) in accordance with plant load demand either by manual, semiautomatic auctuation, or automatically. The protective system, consisting of alarm, setback, and scram circuits, safeguards the reactor against malfunctions and errors which might otherwise create hazardous conditions. The radiation monitoring system measures radiation levels throughout the plant.

A helium system maintains an inert-gas atmosphere in the reactor, in the piping and equipment in contact with sodium, and in the reactor cavity. A nitrogen system maintains an inert-gas atmosphere in the pipe and heat exchanger cells, the cold trap cells, the carbon trap cells, the primary fill tank cell, the moderator coolant pump cells, the maintenance cell, and the primary service pump cell. Nitrogen is also used to provide cooling for the reactor loading face shield, primary system sodium valves, vapor traps, freeze seals, primary cold traps, and primary plugging meters. Circulating air is used to cool the secondary system sodium valves, cold traps, and freeze traps.

A radioactive vent system is provided to control the disposal of any gas that might be radioactive. A radioactive liquid waste system is provided to safely collect, store and reduce in volume aqueous liquid wastes that may be radioactive.

The reactor building houses the reactor, the heat transfer system, reactor service systems and equipment, and miscellaneous maintenance and service areas. The building design conforms to all applicable building codes.[10] The building is of steel frame construction and the exterior walls are covered with insulated steel panels joined to form a reasonably tight enclosure. The reactor building ventilation system is designed to satisfy general heating and ventilation requirements and to minimize the possibility of contaminating the entire plant as a result of an incident in which radioactivity is released to the air. The design objective with respect to the confinement of the building atmosphere is to maintain a negative pressure, relative to outdoor barometric pressure, to insure that leakage of air is inward toward potential radioactive areas.

There are three general problems that are peculiar to reactor systems using sodium as a primary coolant. First, the primary cool-

[10] Applicable building codes do not require any particular construction as protection against possible tornadoes, and there is no evidence that the building proposed would not be damaged to some extent by tornadoes. However, it is believed that the containment cells within the massive concrete structure in the lower part of the building would remain intact, despite any tornadoes, and prevent the release of radioactivity to the environment.

ant becomes highly activated with sodium-24 as it passes through the core. This means that the entire primary sodium system must be heavily shielded to protect personnel operating the reactor. Second, sodium water reactions must be guarded against because of their violent nature and the possibility that such a reaction, if it occurred in primary sodium, could rupture the system and lead to a large release of activated sodium and possibly fission products. Third, sodium burns rapidly in air, and hence must be confined in an atmosphere that prevents the possibility of fires.

The containment scheme utilized in the HNPF is based upon the fact that the primary and secondary sodium systems operate essentially at atmospheric pressure. Hence, structures confining the sodium systems need be designed to withstand only a small overpressure which would result from heating the atmosphere in a confined space by release of hot sodium following a major system rupture. The cells containing the primary sodium system are sufficient to withstand at least six psig internal pressure. This is estimated to be double the pressure increase which would result from a major rupture of primary sodium piping. The possibility that water and air could enter the confinement spaces and cause sodium water reactions or combustion of sodium with resulting greater pressures is precluded by the fact that the cells are made air tight by a continuous lining of steel sheet. Sodium water contact is further prevented by the elimination of all water lines from the vicinity of sodium systems. The massive structures of the sides and tops of the cells provide the necessary shielding for the activated sodium.

For each primary sodium coolant loop, there is a double barrier against the possibility of a release of sodium to the reactor building. The first barrier is the sodium piping; the second is the cell. Over the reactor vessel, there is only a single barrier, the loading face shield. This shield is a massive reinforced structure sealed about its periphery by a continuous frozen metal seal. It is capable of withstanding 58 psig, which is believed to be beyond any credible pressure buildup in the primary sodium system. The reactor building serves as a final barrier against the release of activity to the environment. As previously noted, the building will be maintained at a slightly negative pressure by the ventilation exhaust system. Any airborne material released inside the building will have to pass through filters prior to exhaust through the stack. The containment capability of the confinement cavities for the primary sodium system is adequate to contain the primary sodium in the event of any credible accident. The leakage rate of these cells is sufficiently low that even in the event of an accident involving a major pipe break and partial melting of the core, the off-site exposure would not be greater than the limits specified in Part 20 of the regulations shown in 10 C.F.R.

The design of components of the secondary sodium loops has received particular attention because of the possibility that a loop could be pressurized in excess of its design pressure, if a steam generator tube should fail and allow the higher steam pressure to be imposed on the system. Such an event could lead to rupture of a secondary sodium loop in a heat exchanger cell. This could cause pressure buildup in the cell sufficient to rupture the cell. If these

circumstances were to result in damage to the primary loop within the cell, primary sodium might escape to the reactor building.

This course of events is considered remote since the steam generator tubes are double walled, with helium monitor gas in the space between them. Leaks developing in either the sodium side or the water side of the tubes could be detected by a change in helium pressure. The evidence indicates that it is difficult to conceive of a means by which simultaneous failure of both walls of a steam generator tube could occur. Further in the event of an overpressure condition that could cause rupture of the secondary sodium system, such a rupture would most likely occur outside of the confinement cells. Thus, the containment capability would not be breeched by this type of incident.

The HNPF design is in most basic respects patterned after that of its prototype, the Sodium Reactor Experiment (SRE),[11] a Commission-owned reactor constructed and operated by Atomics International as part of a program to develop a sodium graphite power reactor.[12] This reactor has been operated since April 1957, and a total of 2,409 MWT of heat has been generated. The design, construction, safety features and operating procedures of the HNPF are based on the SRE experience.

During a power run at the SRE in July 1959, the reactor suffered considerable fuel element damage.[13] AI has concluded that the damage was a result of contamination of the sodium system by tetralin, a hydrocarbon material used as an auxiliary coolant for certain components. Flow restrictions were caused by tetralin decomposition products deposited on the fuel elements and the reduced flow resulted in high local fuel cladding temperatures and resultant failure of the cladding. As a consequence of the SRE incident, AI made certain changes in the HNPF design involving primarily (1) complete elimination of the tetralin auxiliary coolant and (2) provision of means for monitoring the cover gas for fission products and the sodium for carbon contamination. It is the opinion of the technical witnesses that these design modifications eliminate the conditions responsible for the SRE difficulties without introducing any additional hazards. Further, the HNPF design now provides means which the witnesses believe are adequate to detect impending trouble of a kind similar to that which preceded the SRE fuel failure in sufficient time to prevent any serious failure of fuel in the HNPF.

SRE experience indicates that sodium graphite reactors in general can be expected to be stable and easily controlled. At the present time it is not definitely known whether the overall power coefficient of the Hallam reactor will be positive or negative.[14] The graphite temperature coefficient is positive, but other components contribute

[11] The SRE facility cost $9,000,000 and has attained a temperature of 1,000°F which was at that time the highest steam temperature produced from a nuclear reactor.

[12] Atomics International will be responsible for the preoperational testing, startup, low power testing, and initial full power attainment by the Hallam facility.

[13] There was damage to 13 of the 43 fuel elements which had been installed.

[14] The Staff witness stated: "I think we would like to know whether it is positive or negative at the time of operation to ascertain that the reactor procedures and operating techniques adequately consider this problem. However, at the design stage we are not sufficiently concerned about this problem to insist upon a resolution of it at this time." In response to a question whether there should be some testing of the reactor after it had been constructed in order to ascertain the temperature coefficient, this witness answered in the affirmative.

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