UNITED STATES ATOMIC ENERGY COMMISSION

IN THE MATTER OF:

ARKANSAS POWER AND LIGHT COMPANY

Docket No. 50-313

THIS DOCUMENT CONTAINS

POOR QUALITY PAGES

Place - Russellville, Arkansas

Date - 30 October 1968

Pages [illegible] - 271

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BEFORE THE

ATOMIC ENERGY COMMISSION

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In the matter of: :

ARKANSAS POWER AND LIGHT COMPANY : Docket No. 50-313

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Young Student Center
Arkansas Polytechnic College
Russellville, Arkansas

Wednesday, 30 October 1968

The conference came on for hearing, pursuant to notice at 10:00 a.m.

BEFORE:

ALGIE A. WELLS, Chairman
DR. LAWRENCE R. QUARLES, Member
R. B. BRIGGS, Member

APPEARANCES:

W. HORACE JEWELL, ESQ., AND PHILIP K. LYON, ESQ., of House, Holmes & Jewell, 1550 Tower Building, Little Rock, Arkansas 72201 and

ROY B. SNAPP, ESQ., 1725 K Street, N.W., Washington, D. C. 20006, on behalf of the Applicant.

THOMAS F. ENGELHARDT, ESQ., Regulatory Staff, Atomic Energy Commission.

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P R O C E E D I N G S

CHAIRMAN WELLS: The hearing will come to order.

This hearing is held in accordance with the Notice of Hearing published on September 20th, 968 found in Federal Register, page 14143. The Notice pertains to an application by the Arkansas Power and Light Company for a provisional construction permit for a pressurized water reactor to be located at a site in Pope County near Russellville, Arkansas.

We are convened in the Young Student Center at the the Arkansas Polytechnic College at approximately ten o'clock. This is the time and place designated in the Notice of Hearing.

The hearing will be conducted by an Atomic Safety and Licensing Board which has been designated by the Atomic Energy Commission. The Board is composed of Dr. Lawrence R. Quarles, Mr. R. B. Briggs, and myself. I am Algie A. Well, and I have been designated Chairman of the Board.

Dr. Lawrence R. Quarles, who is seated at my right, is Dean of the School of Engineering and Applied Science at the University of Virginia. Dr. Quarles holds a Ph. D. from the University of Virginia and has been a member of the faculty of the University since 1935, except for a period of time in which he was Chief Development Engineer of the Oak Ridge National Laboratory where he was in charge of designing and developing controls for homogeneous reactors.

Dr. Quarles has a broad interest in all phases of

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reactor design and has been especially interested over the years in reactor control and instrumentation. He has been on the Atomic Safety and Licensing Board panel for several years, has been a member of more Boards than any other technically qualified member of the panel. Dr. Quarles has participated in more than eleven hearings such as this one.

Mr. Briggs is sitting on my left. He is Director of the Molten Salt Reactor Program at Oak Ridge National Laboratory. Since his graduation from Wayne University in 1941 Mr. Briggs' career has been entirely in the atomic energy field Most of the time he has been engaged in the design and development of a wide variety of reactor types, including the Hanford production reactor.

Since 1962, he has been directly responsible for the technical direction of the Molten Salt Program which is one of the most significant of the Commission's programs, looking forward to the development and the utilization of advanced reactor types.

I am employed by the U. S. Atomic Energy Commission as a permanent Chairman of the Atomic Safety and Licensing Board panel.

The Board welcomes the interest in this hearing which is reflected in the number of people who are here this morning and, for the benefit of those of you who may be attending a hearing of this kind for the first time, I would like to

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make a few observartions which may enable you to follow the proceeding more easily and make the proceeding more interesting and useful to you.

Copies of the Notice of Hearing which I have referred to earlier are available to any of you who may wish to have a copy. I think it would be useful for you to have a copy of this Notice of Hearing because it points out clearly the extent as well as the limitations of the jurisdiction of this Board. I believe copies of the Notice of Hearing can be found at the table probably directly behind the chairs; if any of you would like to take a copy you may do so now.

The Atomic Energy Commission has indicated that it desires that the hearing before the Atomic Safety and Licensing Board be conducted in an informal manner which will permit the presentation of all relevant facts which should be considered by the Board in discharging its responsibility as specified in the Notice of Hearing. At the same time, we hope that this method of proceeding will enable you to acquire an understanding of how the applicant and the regulatory staff of the Atomic Energy Commission, the Advisory Committee on Reactor Safeguards, and finally, this Board itself discharges the responsibility which is theirs with respect to the matters involved.

Now while it is desirable that this hearing be conducted in an informal manner, there are several aspects of a proceeding of this kind which unavoidably require certain

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formalities. At the end of the hearing, the Board is required to make findings and conclusions and to write a decision giving the reasons for these findings and conclusions. The decision must be based on the record made during the hearing, and this record must be available to the public. It is necessary therefore to have a transcript which will be a part of the record.

And at this point I would like to recognize that the Reporter for this hearing is Mr. Bloom. Mr. Bloom, we are glad to have you with us again today, and the Board and all the other participants will give you all possible assistance in reporting the conference. If you need assistance at any time, or the repetition of anything that is said, please let us know.

You will observe that the parties to this proceeding will present their oral and documentary evidence in a careful and precise manner, and in accordance with traditional practices. In the interest of all persons concerned, rules and procedures which are fair and which will enable the proceeding to take place in an orderly, effective manner must be observed.

A pre-hearing conference was held on October 15th, 1968, in Washington, D. C. as was provided for in the Notice of Hearing. If there is no objection, the Board desires that the transcript of the pre-hearing conference be made a part of the record of the hearing so that the transcript will reflect the decisions and agreements made at the conference.

End

Do I hear any objection?

Hearing no objection, the transcript of the pre-hearing conference will be made a part of the record of this hearing. And, of course, if there should be intervenors, this will be done without prejudice to their interests.

During the pre-hearing conference a provisional agenda was submitted to the Board by the applicants and the regulatory staff. This proposed agenda was adopted as a provisional agenda and may be changed with good cause.

There are also copies of this provisional agenda along with the notice of hearing at the table in the back of the room. Again, if any of you are attending a hearing of this kind for the first time, you may find it useful to provide yourself with a copy of the agenda so that you will know in what order the items will be discussed.

Since several people are going to the table to obtain copies of the agenda, we will wait a moment until they have done so.

MR. ENGELHARDT: Mr. Chairman, if there appears to be any shortage of agendas at the table to the rear of the room, I have a small supply of additional copies here at my table which I will make available also.

CHAIRMAN WELLS: Thank you very much.

To continue now with the hearing, I would like to defer the remainder of the opening statements until we have had the appearances of the parties. This is a slight modification of the agenda, but it gives me a rest. And I hope the parties will not object.

MR. W. Horace Jewell. Mr. Edward B. Dillon, Mr. Philip K. Lyon and Mr. Roy B. Snapp have filed notices of appearance on behalf of the applicant.

Mr. Jewell, do you care to make a statement at this point?

MR. JEWELL: Mr. Chairman, I am Horace Jewell, to my left is Philip K. Lyon and to his left is Roy B. Snapp. The three of us are here today appearing for the applicant.

CHAIRMAN WELLS: Thank you, Mr. Jewell.

Mr. Thomas F. Engelhardt has filed a notice of appearance as the representative of the regulatory staff in this hearing. Mr. Engelhardt and his colleagues are prepared to assist the members of the public who may wish to consult with them concerning the regulations and procedures applicable to this hearing. They are also prepared to provide any other appropriate assistance to the public and parties.

If any person desires to discuss with Mr. Engelhardt and his associates any matters, this can be done during the recess which we will have in the course of the morning.

Mr. Engelhardt, do you desire to make an appearance?

MR. ENGELHARDT: Mr. Chairman, I will just identify myself as Thomas F. Engelhardt, counsel for the Atomic Energy Commission regulatory staff of Washington, D. C.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

Three persons have requested the opportunity to make limited appearances and these requests have been granted by a decision of the Board which is reflected in the record of the prehearing conference.

So the Board can be informed if the persons who have been granted permission to make these appearances are in the room, I would like to read their names. It would be appreciated if you would stand as I call your name so that your presence can be recorded.

For this purpose it will not be necessary for you to step to the microphone unless you desire to do so for some reason.

First, Mr. E. F. Wilson, Director, Division of Radiological Health, Arkansas State Board of Health, Little Rock.

I know Mr. Wilson is here because I saw him this morning.

(Mr. Wilson stands.)

CHAIRMAN WELLS: Thank you, Mr. Wilson.

The next is Dr. Howard K. Suzuki, Professor, University of Arkansas Medical Center, Little Rock.

Is Dr. Suzuki here?

(Dr. Suzuki stands.)

CHAIRMAN WELLS: Thank you very much, Dr. Suzuki.

Third, Mr. S. Ladd Davies, Director, Arkansas Pollution Control Commission, Little Rock.

(Mr. Davies stands.)

CHAIRMAN WELLS: Thank you, Mr. Davies.

Mr. Davis, the Board notes in connection with your request for a limited appearance that your statement would be confined to thermal and chemical aspects. This suggests to us that we should probably call your attention to the fact that Section 1715 of the Commission's regulations provides that the Board may permit a limited statement on the issues.

While I do not know exactly what you propose to include in your statement, you will note that as I pointed out and as is pointed out in the notice of hearing, the Commission's jurisdiction does not extend to thermal effects.

Not withstanding this, the Board is prepared to receive your statement, but if it does include matters not germane to the issues, we would respectfully request that this statement be brief.

Is that satisfactory to you?

MR. DAVIES: That is satisfactory.

CHAIRMAN WELLS: Thank you very much, Mr. Davies.

The agenda which we are following as a guide will permit those of you who are making a limited appearance to make a statement later in the morning. This is Item 19 of the agenda.

Although I cannot say exactly when we will reach Item 19 on the agenda, I believe it would be shortly before 12 o'clock.

Now, I understand that Dr. Suzuki has teaching responsibilities at the Medical Center and would like to make his statement as early as possible.

Dr. Suzuki, if it is not convenient to you to wait until, say 11:30, we will arrange for you to make your statement before the mid-morning recess.

DR. SUZUKI: 11:30 will be fine, sir.

CHAIRMAN WELLS: Thank you very much

I will return now to the part of the proceedings which is designated on the agenda as the Chairman's opening remarks. These remarks are rather long but they are designed to acquaint you with the nature of the Board's

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jurisdiction.

The plant we are concerned with today is a nuclear reactor. That is why the hearing is before the Board. It will be used to produce electrical energy and in the case of other plants which produce electricity a number of public agencies may have an interest in one or more aspect of the plant's construction, its operation, and the transmission and sale of electrical energy.

The Atomic Energy Commission, however, regulates only some and not all of the matters involved in the construction and operation of a nuclear reactor. The AEC's regulatory functions are limited by law to essentially two areas, first, public health and safety and, second, the common defense and security.

The issues in this case fall within these two areas. With respect to the first area, public health and safety, the AEC's regulatory interest is further restricted to public health and safety questions relating to the special characteristics of nuclear materials and atomic energy.

These are sometimes referred to as radiological hazards or nuclear hazards. Questions about other aspects of health and safety or other aspects of the plant not falling within the area of radiological health and safety ebl and the common defense and security are not within the AEC's jurisidiction and will not be considered at this hearing.

I have been informed that the loudspeakers on my right, at the right-hand side of the room, the right as I am facing, are not working. If any of you who are sitting on that side of the room desire to move on the other side, you will probably hear better.

Some of you who are present today may have questions concerning aspects of the plant which are not involved in this hearing within the limitations which I have mentioned to you.

I wonder, Mr. Engelhardt, if you would inform us of the several State and local agencies that may have an interest in the non-radiological aspects of the plant?

MR. ENGELHARDT: Yes, sir. I have been informed that the following State agencies would exercise some degree of regulatory authority over the proposed plant. These would be:

The Arkansas State Board of Health;

The Arkansas Public Service Commission;

The Arkansas Pollution Control Commission;

The Arkansas Commissioner of Labor;

The Arkansas Workman's Compensation Commission.

In addition there would be the Arkansas Fish and Game Commission, and the Arkansas State Police would have an interest in the operation of this plant.

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In addition, on the local level, the County Judge of Pope County and the Sheriffs of Pope County and Yell County would have an interest in the operation of this plant.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

Now, in addition to the State and local agencies which Mr. Engelhardt has mentioned, several Federal agencies other than the Atomic Energy Commission may have some interest in certain aspects of the operation of the plant. For example, effects of the discharges which might affect the quality of the water may be controlled by the Federal Water Pollution Control Administration of the Department of the Interior.

Similarly, gaseous discharges, if any, which might result in air pollution other than radiological may be controlled by the Department of Health, Education and Welfare.

A plant which generates electrical energy for transmission in interstate commerce or for sale in interstate commerce is subject to some of the regulatory provisions of the Federal Power Act which is administered by the Federal Power Commission.

In some instances involving public utility holding companies, the Securities and Exchange Commission's requirements rather than those of the Federal Power Commission would be applicable.

Now I have merely outlined the areas of possible interest of other Federal agencies. If any of you are interested

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in these aspects, I suggest that you consult with Mr. Engelhardt, the staff counsel. He can provide you with more details concerning these agencies' jurisdiction and the Federal laws under which they operate.

Now the time has long since passed since the possibility for intervention as set forth in the Notice of Hearing was available. The Board, however, is prepared to consider any request for intervention this morning if good cause can be shown for failure to make the application earlier.

I wonder if there is any one present today that desires to intervene in this hearing as a party and can show good cause for failure to make an application earlier?

(No response.)

CHAIRMAN WELLS: Noting that there is no such person present we will conclude that this hearing will be what is called an uncontested hearing within the meaning of the Commission's regulations.

Now earlier in the proceeding I indicated that there were requests from three persons to make limited appearances. I wonder if there are any persons here this morning who would desire to make a limited appearance?

(No response.)

CHAIRMAN WELLLS: I believe there are none.

So that completes the appearances of Counsel, and the identification of persons who desire to make limited

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appearances.

We now come to Items 4 and 5 of the agenda, and these two items are closely related. They are the opening statements by applicant's Counsel and the summary oral statement by the applicant.

Before giving the applicant's Counsel an opportunity to make an oral statement, however, I would like to point out again for the benefit of any member of the public who did not attend the pre-hearing conference that the applicant has the burden of proof in this proceeding. Another way of saying this is that the applicant is the moving party. The applicant initiated a series of events which led to this hearing by making an application in November 1967 for a construction permit. This application was made with the knowledge of and in conformance with the rules pertaining to the issuance of a construction permit as laid down by the Commission's regulations.

At this point in the hearing there is provided an opportunity for the applicant to inform the Board and the public in a general way of the steps that he has taken and proposes to take to insure the safety of the public by such means as site selection, engineering safeguards, and other safety features of the reactor.

Please bear in mind that the statements that the applicant's Counsel will make at this time are only general and preliminary. He will have an opportunity later to present

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in depth testimony which will be subject to cross-examination by the regulatory staff if they desire to do so, and will be subject to questions by this Board. This will be done in connection with Item 13 of the agenda, I believe, either Item 13 or 14.

Mr. Jewell, we would be very glad to have an opportunity to hear your statement at this time.

OPENING STATEMENT OF HORACE JEWELL ON BEHALF OF THE APPLICANT

MR. JEWELL: Thank you, Mr. Chairman.

If it please the Board, the application which is the subject matter of this hearing was filed by the Arkansas Power and Light Company with the Atomic Energy Commission under the provisions of Section 104(b) of the Atomic Energy Act of 1954, as amended, and pursuant to the Rules and Regulations of the Atomic Energy Commission as set forth in 10 CFT, Part 2, Section 2.101.

By this application, the Arkansas Power and Light Company seeks a permit to construct and a license to operate a nuclear reactor which will be utilized in the generation of electric power and energy. This application, as amended by ten supplements, is quite voluminous and includes three volumes and three supplemental sections. It describes in detail the facility which the Arkansas Power and Light Company proposes to construct and the criteria to which this facility is being designed. A copy of the appl·cation is on the table at the rear of the room for ·e convenience of any members of the public who may desire t inspect it.

One ·· he primary purposes of this public hearing is to inform the public generally as to the nature of what is being proposed by the Arkansas Power and Light Company, and the close scrutiny which has been given it by the Atomic Energy Commission in an effort to perform its primary function of protecting the public against radiation.

Without attempting to pre-empt the function of this Board and of this hearing, Arkansas Power and Light Company has been diligently engaged in its own effort to fully inform the public since making its decision to enter the field of nuclear generation. There has been a great deal of publicity through news stories, appearances at civic clubs, and personal contacts with those persons who have shown an interest in the application. Copies of the complete application and all amendments have at all times si·ce the original filing with the Atomic Energy Commission been on file in the Office of the Honorable Wayne Norton, County Judge of Pope County, Arkansas, the Honorable Winthrop Rockefeller, Governor of the State of Arkansas, the Hornorable Joe Purcell, Attorney General of the State of Arkansas, the Arkansas Public Service Commission, and the Arkansas Pollution Control Commission.

I. addition, copies have at all times been available at the offices of the company in Pine Bluff and in Little Rock.

A second function of this public hearing is to consider the answers to certain questions which are set forth in the Notice of Hearing. Those questions are:

Question No. 1: Whether in accordance with the provisions of Section 50.35(a) of the Atomic Energy Commission's Rules and Regulations:

a) The applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and enginerring criteria for the design, and has identified the major features of components incorporated therein for the protection of the health and safety of the public;

b) Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration will be supplied in the final safety analysis report;

c) Safety features or components, if any, which require research and development, have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and

d) On the basis of the foregoing, there is reasonable assurance that (1) such safety questions will be satisfactorily resolved at or before the latest date stated in the

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application for completion of the proposed facility; and (2) taking into consideration the site criteria contained in Part 100 of the Rules and Regulations of the Atomic Energy Commission, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

Question 2: Whether the applicant is technically qualified to design and construct the proposed facility.

Question 3: Whether the applicant is financially qualified to design and construct the proposed facility.

Question 4: Whether the issuance of a permit for the construction of the facility will be inimical to the common defense and security or to the health and safety of the public.

The Director of Regulation of the Atomic Energy Commission has announced his intention to make an affirmative finding on the first three of these questions and a negative finding on the fourth question. Inasmuch as this is an uncontested hearing as announced by the Chairman, the objective of the Board in this hearing will be to consider the issues of whether the application and the record of this proceeding contains sufficient information, and the review by the Commission's regulatory staff has been adequate, to support the findings which the Director of Regulations proposes to make and the provisional construction permit which he proposes to issue.

These issues are those upon which the Chairman

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has properly pointed out the burden of proof rests on the applicant, and we will undertake to satisfy that burden of proof.

The first witness for the Arkansas Power and Light Company will be Mr. A. B. Coen, the Vice President-Treasurer and Secretary of Arkansas Power and Light Company. Mr. Coen will testify to facts showing the financial ability of the company to construct and operate and then to shut down the proposed facility. testimony will also cover facts involved in the question concerning the common defense and security of the United States as affected by this facility.

The second primary witness for the applicant will be Mr. Harlan T. Holmes, the Nuclear Project Manager of the company. Mr. Holmes will introduce as his primary testimony, a Summary Description of the application of the company in this proceeding. This summary will include a description and some detail of the proposed facility and its site.

The summary also will set forth the technical qualifications of the Arkansas Power and Light Company as a company to construct and operate the nuclear plant.

In addition, Mr. Holmes will also present and answers to certain questions which were previously propounded by members of the Board to the applicant and the staff at the pre-hearing conference. He will be assisted in answering on cross-examination and in answering further questions from the eb10 members of the Board by a panel of expert witnesses composed of Dr. Knox Broom, Nuclear Specialist with Middle South Services, Inc.; Mr. Harry P. Marsh, Mr. Paul Schmitz, and Mr. Howard W. Wahl, of Bechtel Corporation; and Mr. James McFarland, Mr. Robert E.Wascher and Mr. William R. Smith, of the Babcock and Wilcox Company.

In addition to these experts, the applicant will have available for answering technical questions a panel of back-up witnesses who will be identified as they are needed.

Mr. Chairman, this concludes my portion of the opening statement, and with the permission of the Board, I would like to call Mr. Harlan T. Holmes to give the summary description of the plant provided for as required under Section 3(c) of Appendix A to the Rules of Practice of the Atomic Energy Commission.

CHAIRMAN WELLS: Thank you, Mr. Jewell.

I take it that it will not be necessary for Mr. Holmes to be sworn until your other witnesses are sworn?

MR. JEWELL: I do not believe it is necessary for him to be sworn at this time.

CHAIRMAN WELLS: Thank you. We will be very glad to hear from Mr. Holmes.

End

OPENING STATEMENT OF HARLAN T. HOLMES

ON BEHALF OF THE APPLICANT

MR. HOLMES: The Russellville Nuclear Unit is a pressurized water reactor. The Nuclear Steam Suply System being supplied by the Babcock & Wilcox Company (B&W) is similar in design concept to others now in operation or under construction under AEC Licenses.

The evolution of B&W's reactor system proceeded on a schedule allowing in 1955 for them to contract for Indian Point No. 1 which went into service in 1963. In 1957, they entered into contract for the entire Propulsion Plant for the Nuclear Ship Savannah which was placed into service in 1962. Duke Power Company's Oconee Units 1, 2 and 3 were closely followed by Metropolitan-Edison's Three Mile Island No. 1, Florida Power Corporation's Crystal River Unit No. 1, Sacramento Public Utility District's Rancho Seco Unit and this Russelville Nuclear Unit. This is basically the evolution of the heretofore licensed reactors by B&W similar to the Russellville Nuclear Unit.

The reactor will operate initially at a nominal core power level of 2452 thermal megawatts. All physics and core thermal hydraulics information submitted in support of our Application is based on a core design for operation at that level.

It is expected, however, that the nuclear steam supply system will be capable of an ultimate output of 2584 megawatts thermal -- including 16 mwt contribution from the reactor primary coolant pumps. The facility systems, engineered safeguards and containment are designed consistent with safe operation at this ultimate power level. In addition, accident analyses presented have been made on the basis of the ultimate power level.

The reactor will be refueled with slightly enriched uranium dioxide pellets contained in zircaloy tubes. Control of reactivity will be provided by a combination of neutron absorbers and movable control rods. The neutron absorber, boric acid, is dissolved in the reactor coolant for the purpose of controlling the long-term reactivity changes of the core and provide cold shutdown. Silver-Indium-Cadmium control rods clad in stainless steel are employed to control short-term changes in reactivity levels and to provide fast shutdown capability.

Incore instrumentation, consisting of self-powered neutron detectors, will be located at pre-selected locations within the core. This instrumentation will monitor core performance. The fuel core will be supported within a heavy-walled steel reactor vessel, through which reactor coolant water will be pumped to remove the heat generated within the core.

This thermal energy will be transferred to two once-through steam generators. The steam produced will be used to drive a steam turbine-generator, the capability of which initially will be about 855 gross megawatts of electricity. Ultimately, it is expected that the unit will have a gross electrical capability of about 885 megawatts.

There are numerous systems, components and features incorporated into the plant for the protection of the health and safety of the public. The first line of protection against the release of fission products from the reactor is the fuel pellets themselves, with their high capability for retaining fission products within their own physical structure.

The fuel pellets are inserted in zirconium metal tubes which are designed and selected to withstand without failure much higher temperatures and pressures than those to which they will be subjected, thus preventing the escape of fission products. In the event of fuel tube failure for whatever case with a release of any contained fission products, these fission products would remain within the liquid reactor coolant system contained in the primary coolant piping loops all of which are within the reactor building containment structure.

Finally, the reactor buildings containment structure encloses and contains the entire reactor coolant system to limit the release of radioactive fluids and vapors to the environment in the unlikely event of an accident. In the Russellville Nuclear Unit the reactor coolant system will be housed in a prestressed, post-tensioned concrete reactor containment building in the shape of a cylinder. The inside diameter of the building is 116 feet and the inside height will be 206 feet. The reactor containment building will rest on an integral concrete slab approximately 9 feet thick.

The building will be lined internally with 3/8-inch welded steel plate to provide vapor tightness. The reactor building containment structure is designed to limit radioactivity release, in even of an accident, to values well below 10 CFR 100 guidelines published by the Atomic Energy Commission in the Federal Register.

The Russellville Nuclear Unit is being designed to rigid codes and standards to assure reliable safe operation without adverse effect from any release on the environment. It is a prime requirement that this plant operate continuously to supply reliable electrical power to the Company's customers inArkansas. We are making every effort to assure that the design, the manufacturing of equipment, the construction and the operation of the facility meet the highest standards for reliability and safety.

In the design, protection of the public is assured by the following engineered safeguards systems:

1. Redundant systems are provided which inject sufficient borated water directly into the reactor vessel to assure adequate cooling of the core and thus limiting any damage to the reactor fuel.

2. Two separate and redundant reactor building emergency cooling systems designed to cool gases and condense steam that might be introduced into the building in the event of an accident. These systems will limit the building pressure to less than its design pressure and will return the pressure to normal.

3. The reactor building containment structure is designed to safely contain the maximum pressure buildup resulting from complete rupture of the largest reactor coolant pipe.

Each of these safeguard systems include redundant components to assure their functioning as intended. These engineered safety systems will effectively protect the public from any credible accident in the Russellville Nuclear Unit.

The Russellville Nuclear Unit site is located on a peninsula in Dardanelle Reservoir in Pope county, Arkansas, about six miles west-northwest from Russellville and about two miles southeast from London. Arkansas Power and Light Company owns the site, which consists of approximately

1,000 acres. The plant will be the center of an exclusion area of .65 mile in which habitation will be prohibited and which will all be controlled by Arkansas Power & Light Company. A low population zone of four miles radius has been established.

The surface drainage of water at the site is toward Dardanelle Reservoir, and strata of impervious clay and rock prevent surface water from penetrating to the subsurface water. There are no potable water supplies which could be affected by the plant at this location.

The plant structure which will be founded on underlying dense shale, will be capable of supporting the loads to be imposed upon it.

The site is located in a relatively quiescent area seismically.

The nuclear generating unit will be protected against winds and floods.

The location of the plant on the peninsula provides direct access for intake and discharge cooling water supply and discharge of the nuclear plant.

This plant site is characterized by very favorable conditions of hydrology, geology, seismology and meteorology.

Nuclear power plants are quite similar to the fossil-fired steam plants Arkansas Power & Light Company is now operating. The fundamental difference is in the energy source used to make the steam to drive the turbines.

In this nuclear plant the energy source is enriched uranium dioxide pellets contained in metal tubes, located within the nuclear reactor vessel. In Arkansas Power & Light's conventional plants, oil and natural gas are used as energy sources. Both types of plants have steam generators to produce the steam for the turbines which directly drive the electric generators.

CHAIRMAN WELLS: Thank you very much.

In his opening statement Mr. Jewell very succinctly, I think, described the role of the AEC's regulatory staff, the Advisory Committee on Reactor Safeguards in examining and evaluating the application.

I would like now to ask staff counsel if he desires to make a statement.

MR. ENGELHARDT: Yes, sir. I would like to make a brief opening statement and then ask one of our technical witnesses to supplement that statement with some weather information.

OPENING STATEMENT OF THOMAS F. ENGELHARDT

XXXXXXX ON BEHALF OF THE REGULATORY STAFF

MR. ENGELHARDT: As the Chairman has indicated, the provisions of the Atomic Energy Act of 1954, as amended, has been available here in the local community.

The first stage of the review of an application for a permit to construct a nuclear reactor such as the Russellville plant involves a safety evaluation of the application.

The application is evaluated by technical specialists on the Commission's regulatory staff and their expert consultants and also by the Commission's Advisory Committee on Reactor Safeguards.

The Advisory Committee is an independent committee established by Congress to advise the Commission on matters of nuclear power reactor safety and other matters involving different types of facilities.

It is composed of scientists and engineers who are specialists in various disciplines important to reactor safety. The report of the Advisory Committee on Reactor Safeguards and the technical evaluation of the safety considerations relevant to the proposed facility prepared by the AEC's regulatory staff called a safety evaluation are made public prior to the hearing on the construction permit application.

Before a construction permit is issued to an applicant, the Commission must first find that there is reasonable assurance that the applicant will comply with the Commission's regulations and that the health and safety will not be endangered and the applicant is technically and financially qualified to engage in the proposed activity, but the Commission's interest in the facility does not end with issuance of a construction permit. The facility is subject to periodic inspection by Commission inspectors to assure that the facility is constructed in accordance with the provisions of the construction permit.

Before an operating license may be issued, the applicant must file with the Commission an application for such a license which contains detailed information regarding the completed design of the facility, a detailed description of the facility and its components and operating plans.

This application is given the same detailed review by the regulatory staff and the Advisory Committee on Reactor Safeguards as was the case at the construction permit stage.

Also, before issuing an operating license for a nuclear reactor, the Commission publishes in the Federal Register a notice of its intent to do so and offers any interested person an opportunity to request a hearing.

If the Commission determines that the proposed operation of the facility involves safety problems of unusual significance, it may on its own initiative require another public hearing.

This same procedure is followed with respect to requested amendments to either a construction permit or a license.

In the hearing today Mr. Charles Long, Mr. Paul Check and Mr. Albert Schwencer will testify for the Atomic Energy Commission regulatory staff on the technical apects of the application and will be available for cross examination.

Their testimony is contained in the safety evaluation which I have previously described. The staff will also offer testimony of Mr. Charles A. Lovejoy of the Office of the Comptroller of the Atomic Energy Commission with respect to the financial qualification of this applicant.

The staff's safety evaluation is available to anyone attending this proceeding on the table at the rear of this room. And members of the public are welcome to take copies with them.

Copies of Mr. Lovejoy's testimony and the statements of the professional qualifications of Mr. Long, Mr. Check and Mr. Schwencer are also available at the table at the rear of this room.

As Mr. Jewell, counsel for the applicant, has indicated, several witnesses will be presented by the applicant at this hearing. The staff has extensively questioned the applicants during the course of the extensive review of this application. The answers to many of these questions are found in the various amendments to the application.

Consequently, the staff will have a very few, if any, additional questions to ask of witnesses for the applicant in this proceeding.

This concludes my opening statement, Mr. Chairman. However, with the Board's permission, I would like to request Mr. Schwencer to supplement this statement by describing the staff's evaluation of the application, the reasons for the conclusion reached by the staff and summarizing the various steps taken by the staff and the ACRS in their review of the application

I might add that copies of Mr. Schwencer's statement are available at the rear of the room. And copies will be made available for the Board at this time.

CHAIRMAN WELLS: We would be very glad to hear Mr. Schwencer's statement. I am inclined to believe, however that since we have been in session for almost an hour that it would be easier for the Board to concentrate on the statement and perhaps the members of the public if we had a short recess before you give your statement.

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If that is agreeable to the parties, we will have a recess, shall we say, until 10 minutes after 11.

(Recess.)

CHAIRMAN WELLS: The hearing will come to order.

Before hearing Mr. Schwencer's statement I would like to be sure that those of you in the room can hear what is being said. I am told the amplification system is not working on the righthand side of the room. I wonder if Mr. Yore, who is sitting in the back row on the left can hear?

MR. YORE: Very well on this side.

CHAIRMAN WELLS: Thank you very much.

Before we adjourned for the recess we were about to hear the statement of Mr. Albert Schwencer on behalf of the Atomic Energy regulatory staff.

Mr. Schwencer, we will be very glad to have your statement at this time.

SUMMARY STATEMENT ON BEHALF OF THE ATOMIC ENERGY COMMISSION STAFF BY MR. ALBERT SCHWENCER.

MR. SCHWENCER: The Akansas Power & Light Company applied to the Atomic Energy Commission on November 29, 1967, for a permit to construct and operate a pressurized water reactor, known as the Russellville Nuclear Unit, at a site in Pope County, Arkansas.

The technical safety evaluation of the proposed nuclear generating station has been performed by the Commission's regulatory staff, based on the applicant's Preliminary Safety Analysis Report and ten subsequent amendments which were submitted as part of the application. During the course of our evaluation, we have visited the proposed reactor site and have held a number of meetings with the applicant and its representatives, including those from the Bechtel Corporation, its architect-engineer and manager of construction and those from the Babcock & Wilcox Company, its reactor and nuclear steam supply system contractor.

We were assisted in this evaluation and received reports from our consultants on special aspects of the application, including seismicity, U.S. Coast and Geodetic Survey, geology and hydrology from the U.S. Geological Survey, meteorology from the Environmental Science Service Administration, and for environmental considerations, the U.S. Fish and Wildlife Service. We also received a report from Nathan M. Newmark Consulting Engineering Services, a structural design consultant.

The proposed pressurized water reactor to be designed and furnished by the Babcock & Wilcox Company is expected to operate initially at core power levels up to 2452 Mw thermal. The applicant anticipates, however, that the reactor ultimately will be capable of operating at a core power level of 2568 megawatts thermal.

Accordingly, the applicant and we evaluated the engineered safety features of the reactor, and accident consequences at a power level of 2568 megawatt thermal, and evaluated the thermal-hydraulic characteristics of the reactor onthe basis of core power level of 2452 megawatts thermal.

Before operation at any power level above 2452 megawatts thermal can be authorized, however, the Commission is required to perform a safety evaluation to assure that the facility, including the core, can be operated safely at the high power level.

Our safety evaluation encompassed an examination of the proposed plant layout, structural design, and plant operating characteristics with special attention to those aspects concerned with public health and safety relative to radiological effects.

In this respect, we considered the nuclear, thermal-hydraulic, and mechanical design characteristics of the reactor core and found them to be appropriate under all anticipated modes of operation.

We reviewed the proposed instrumentation and control systems and found them to be acceptable for the construction permit stage. We have considered the overall mechanical layout of the plant, including provisions for shielding and missile protection and conclude that the appropriate measures have been taken in these respects.

The nuclear steam supply system design and the overall containment design of the Russellville plant are very similar to those of the three Oconee plants currently under construction by the Duke Power Company.

Since the initial filing of its application, the applicant has made three significant changes in the design of the plant: The containment building design was revised to provide for three instead of six vertical buttresses and for a 240-degree span instead of a 120-degree span of horizontal tendons, the emergency core cooling system was revised to provide more complete separation and better protection against failures, and the electrical system was redesigned to provide automatic selection of offsite power for emergency conditions.

In addition, the applicant made the following significant changes in the plant design as a result of the regulatory staff evaluation: Installed a chemical addition iodine removal system to the containment sprays to assure that any offsite radiation exposure does not exceed 10 CFR 100 limits, and added an on-site pond of water to provide a backup source of emergency cooling water.

The applicant has also agreed to replace 1200 feet of an existing gas line which traverses the site with piping which meets the current ASA gas pipeline code and to provide isolation capability so that in the event of a break the gas line can be shut off, and also to perform tests on the containment structure's liner and tendon anchorages to confirm the adequacy of their design. We have found the above additions and design changes to be acceptable and compatible with the Commission's General Design Criteria.

We have evaluated the consequences of potential accidents which could involve the release of radioactivity from the Russellville Nuclear Unit and have concluded that in the unlikely event of any of these accidents, the potential doses from the release of radioactivity would not exceed the guidelines set forth in 10 CFR Part 100 of the Commission's regulations

The applicant has identified further research and development work on a number of items which will be performed during the detailed design of the plant.

Each of these items has been identified in the application and in our safety evaluation. In our opinion, this research and development program will provide the data necessary to construct the facility in accordance with the criteria and specifications set forth in the application.

The Arkansas Power & Light Company will be responsible for the design and construction of the plant and will operate the plant. They have contracted with the Bechtel Corporation to furnish architect-engineering services including design of the reactor containment structures

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and to serve as construction manager.

We have examined the technical qualifications of the applicant and its principal contractors as well as the applicant's planning for the conduct of operations, both normal and emergency.

The Division of Compliance, a part of the regulatory staff, will conduct continuing inspections of the plant during its construction to assure that it will be constructed in accordance with the provisions of the provisional construction permit.

The Advisory Committee on Reactor Safeguards performed an independent review of the application for the proposed plant and provided comments and recommendations to the Commission in its September 12, 1968, report to the Chairman of the Atomic Energy Commission. We have considered each of these and will be guided by all of them in our continuing review of the Russellville Nuclear Unit.

The ACRS report concludes that with due consideration to the various items mentioned therein, "... the proposed plant can be constructed at the Russellville site with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

We have concluded, based on our review and evaluation of the Arkansas Power and Light Company's application, that appropriate findings can be made on each of the issues set forth in the Notice of Hearing issued in this proceeding.

The construction permit sought for this plant would be the first step in the Commission's regulatory process which would continue throughout the lifetime of the plant.

Prior to issuing an operating license for this plant, the final design will be thoroughly evaluated by the regulatory staff of the Division of Reactor Licensing and the Advisory Committee on Reactor Safeguards in a manner similar to the review process at this, the construction permit stage, in order to determine that all of the Commission's safety requirements have been satisfied.

The plant would then we operated only in accordance with the Commission's regulations and under the continued scrutiny of the Commission's regulatory staff throughout the plant lifetime.

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CHAIRMAN WELLS: Thank you, Mr. Schwencer.

MR. ENGELHARDT: Mr. Chairman, that completes the staff's opening statements.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

We come now to Item 8 of the agenda, which is the introduction of application, amendments and correspondence as Joint Exhibits.

Do you desire to make the introduction of the application, Mr. Engelhardt?

MR. ENGELHARDT: Yes, sir.

The Regulatory Staff of the Atomic Energy Commission offers for identification Joint Exhibit A, which is a three-page document containing a complete description and listing an index of 14 documents relevant to this proceeding which have been filed or may be part of the public record of the Commission in connection with this proceeding.

The 14 documents which are described and identified in Joint Exhibit A are to be incorporated by reference into the evidentiary record of this proceeding for any and all use by the parties and by the Board in this proceeding. Copies of the documents have been previously made available to the members of the Board and the applicant. Additional copies will now be transmitted to the Board and to the applicant.

(Documents distributed.)

MR. ENGELHARDT: I would now like to request that

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the applicant join me in offering in evidence Joint Exhibit A, which I have just described, and request that it be copied into the transcript of this proceeding as if read.

MR. JEWELL: Mr. Chairman, the applicant joins the staff in the request just made.

CHAIRMAN WELLS: Thank you very much. The record will so indicate.

(The document referred to follows:)

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

RUSSELLVILLE NUCLEAR UNIT NO. 1

RECORD FOR HEARING

INDEX

Item No.	Description	Date
1.	Application for Licenses, consisting of: General Information with Exhibits 1, 2, 3 & 4, Volumes I and II of Preliminary Safety Analysis Report (PSAR)	11/24/67
2.	Supplement No. 1 to Application for Licenses, containing information concerning the 70 criteria proposed as an amendment to 10 CFR Part 50	1/22/68
3.	Amendment No. 1 (later redesignated Supplement No. 2) to Application for Licenses, consisting of: (1) revised pages for Preliminary Safety Analysis Report (PSAR) to reflect changes in design. (2) Appendix 2-F presenting a safety investigation on Dardanelle Lock and Dam	2/8/68

Item No.	Description	Date
4.	AEC letter requesting additional technical data.	4/3/68
5.	Supplement No. 3 to Application for Licenses, (1) redesignating Amendment No. 1 as Supplement No. 2 to application, and (2) furnishing answers to questions raised in AEC letter of April 3, 1968	5/3/68
6.	AEC letter requesting additional technical data	5/6/68
7.	Supplement No. 4 to Application for Licenses, consisting of: (1) answers to questions raised in AEC letter of May 3, 1968 (2) revised pages for Supplement No. 3 (3) revisions to certain pages of PSAR (corrections)	6/5/68
8.	Supplement No. 5 to Application for Licenses, consisting of: (1) additional information on reactor containment structure (2) Appendix J to Supplement No. 5, on Post Tensioning, (PROPRIETARY INFORMATION), dated 6/28/68	7/3/68

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Item No.	DESCRIPTION	Date
	Correction to Supplement No. 5	7/10/68
9.	Supplement No. 6 to Application for Licenses, consisting of: (1) supplemental information in response to informal questions raised by AEC staff (2) revised pages for PSAR	7/11/68
	Correction to Supplement No. 6	7/15/68
10.	Supplement No. 7 to Application for Licenses, containing supplemental information in response to informal questions raised by AEC staff	8/15/68
11	Supplement No. 8 to Application for Licenses, consisting of: (1) corrections to Supplement No. 7 (2) supplemental information in response to informal questions raised by AEC staff.	8/26/68
12.	Supplement No. 10 to Application for Licenses, consisting of: (1) updated financial and organization data (2) revised pages for PSAR TO incorporate corrections.	9/6/68

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Item No.	Description	Date
	Letter from Roy B. Snapp confirming that the revised pages for PSAR submitted with Supplement No. 10 are to correct semantic and numerical errors and do not alter substance of the PSAR	9/12/68
14	AEC letter granting Applicant's request for withholding from public disclosure the contents of Appendix J of Supplement No. 5 to the Preliminary Safety Analysis Report	9/16/68

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MR. ENGELHARDT: Mr. Chairman, I believe that that completes Item 8 on the agenda, at least to the extent that I have any further knowledge of what might be included.

CHAIRMAN WELLS: Does the applicant have anything to add?

MR. JEWELL: No, sir.

CHAIRMAN WELLS: Thank you very much.

Then we come to Item 10 on the agenda, Item 9 not being applicable because there are no intervenors.

Item 10 provides for the statements by persons making or desiring to make limited appearances.

We have Mr. E. F. Wilson, the Director of the Division of Radiological Health, Arkansas State Board of Health, who is the first person to request the opportunity to make a limited statement.

We will be very glad to hear from you, Mr. Wilson.

xzxgx STATEMENT OF E. F, WILSON, DIRECTOR OF THE DIVISION OF RADIOLOGICAL HEALTH OF THE ARKANSAS DEPARTMENT OF HEALTH

MR. WILSON: Thank you, Mr. Chairman.

Mr. Chairman and members of the Board. I am E. F. Wilson, Director of the Division of Radiological Health of the Arkansas Department of Health. My appearnace at the proceeding today is on behalf of J. T. Herron, M. D., State Health Officer.

The State of Arkansas entered into an agreement with the Atomic Energy Commission on July 1, 1963, pursuant to our Act 8 of the Second Extraordinary Session of 1961. As a result of the agreement, the State of Arkansas assumed all regulatory control of By-Product, Source and Special Nuclear Material in less than critical mass quantities. The law additionally designated the State Board of Health as the control agency with the Division of Radiological Health responsible for the administration of the program.

The Board of Health recognizes that regulation of the construction and operation of utilization and production facilities is excluded from the agreement; however, the Board of Health is responsible for the protection of the environmental areas surrounding any nuclear facility. The primary interest of the Board of Health therefore lies in those on-site operations that release or could release radioactive material to off-site areas resulting in radiation exposure to the general population.

It is the two-fold purpose of this statement to support the Arkansas Power & Light Company in their application for a construction permit and to define those specific responsibilities the Board of Health will assume during the construction and operational phase of the facility.

As a basis for support of the application, the Division of Radiological Health conducted a public health

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analysis of the planned Russellville Nuclear Unit. The analysis included a visit in April, 1968, to the proposed site by the representatives of the Division of Radiological Health and the Nuclear Facilities Section, National Center for Radiological Health, U.S. Public Health Service, in addition to a staff review of the following documentation:

1. The Preliminary Safety Analysis Report, Volumes I and II, by the Arkansas Power & Light Company.

2. The Public Health Evaluation of the Russellville, Nuclear Unit by Nuclear Facilities Section, U.S. Public Health Service.

3. Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission.

As a result of this analysis, the Board of Health has assumed the following responsibilities:

1. The design and implementation of an Environmental Surveillance Program around the Russellville Nuclear unit.

2. The development and utilization of a Division of Radiological Health, Contingency Plan.

3. As specified in Act 386 of 1957, the Division of Radiological Health will review the adequacy of present Rules and Regulations and

the Board of Health will initiate, if deemed necessary, any changes to the Rules and Rgulations to adequately meet the above responsibilities.

The Division of Radiological Health will implement the Environmental Surveillance Program during the construction phase of the facility. The program will be designed to collect and evaluate a sufficient number of samples to establish the background levels indigenous to the site environs.

Emphasis will be placed on, but not limited to, the critical pathways to man, ie., aquatic biota, air and milk. Sampling locations and frequency will be selected in accordance with current guidelines. The Environmental Surveillance Program will continue following the start of plant operation and will be updated in design with the objective of verifying the adequacy of the in-plant monitoring and source control by the Arkansas Power and Light Company in addition to the detection of abnormal releases to the environment.

Copies of the proposed Environmental Surveillance Program, as well as the results obtained, will be furnished to the Arkansas Power and Light Company and other interested State and Fedeal agencies on a regular basis.

The Arkansas Power and Light Company should

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furnish the Division of Radiological Health a copy of the proposed On-Site Environmental Surveillance Program for review. Additionally, the subsequent results should be furnished on a quarterly basis.

The Division of Radiological Health will continually review the need for additional surveillance based on the data obtained from the original programs; however, in the event of a radiological emergency, additional surveillance measures will be instituted according to the Division of Radiological Health Contingency Plan.

A Radiological Contingency Plan will be prepared by the Division of Radiological Health and will be utilized by the Department of Health in the event of a radiological emergency. Required pre-planning, coordination and testing, in anticipation of all contingencies, will be completed prior to the start of plant operation.

The Arkansas Power and Light Company has agreed to coordinate with the Division of Radiological Health to jointly define the conditions, either real or suspected, that would require the notification of the Division of Radiological Health. Additionally, the State Department of Health will retain, on a consulting basis, a physician who has received specialized training in nuclear medicine and will coordinate with local hospitals to insure the rms5

availability of adequate hospital facilities.

The Division of Radiological Health has concluded in the public health analysis of the proposed Russellville Nuclear Unit that the construction and eventual operation of this facility does not represent an undue risk to the health and safety of the general public.

CHAIRMAN WELLS: Thank you very much, Mr. Wilson.

The Board would now be glad to hear from Mr. A. K. Suzuki, Professor of Medicine at the University of Arkansas.

STATEMENT OF HOWARD K. SUZIKI, PROFESSOR OF ANATOMY, UNIVERSITY OF ARKANSAS SCHOOL OF MEDICINE

DR. SUZUKI: Mr. Chairman, Members of the Board:

My name is Howard Suzuki. I am Professor of Anatomy at the University of Arkansas School of Medicine. And I am also Chairman of the Arkansas Conservation Council which represents a number of State organizations interested in the natural resources of the State.

This will be a joint statement between myself and Dr. Joe F. Nix, who is Associate Professor of Chemistry at Warshaw Baptist University in Arkadelphia and also Co-Chairman of the Soil and Water Committee of the Arkansas Wildlife Federation.

The impending nuclear power generation facility to be constructed on Dardanelle Reservoir will affect Arkansans for many years to come. The need for this facility is not questioned, nor would one want to retard the expension of this industry.

Radiation hazards from uncontrolled nuclear chain reactions or the release of radioactive by-products into the environment seems extremely remote.

The effect of the discharge of 1,700 cubic feet per

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second of heated effluent on the econology of the Dardanelle Reservoir remains unanswered. Since elevation of the environmental temperature and associated phenomena can be detrimental to the ecological systems of a reservoir, it is important that the thermal pollution aspects of the Russellville nuclear unit be considered.

As interested conservationists, we have been in contact with the Arkansas Power and Light Company and have accompanied representatives of AP and L on an inspection tour of an operating nuclear power plant in Haddam Neck, Connecticut. We have also observed the model of Dardanelle Reservoir which has been constructed at Hydro-Science Engineering Company in San Jose, California.

At this stage of the studies, the extent and effect of stratified flow which may be induced by the heated effluent has been determined by engineering projections and has not been confirmed by the model studies. If the presence of extensive stratified flow should allow a high degree of recirculation, the resulting temperature fise may be in excess of predicted values. The effect of an excessive temperature rise on even a section of Dardanelle Reservoir could cause severe damage to the fishery resources of these waters.

It appears that the results of the model study will be forthcoming in the immediate future. These studies should answer many of the questions concerning the magnitude of the eb3 temperature rise, the possibility of recirculation, and the extent of stratified flow.

Past experience has taught that the effect of the discharge of large quantities of heated effluent into natural water systems varies greatly from situation to situation and that final judgment can only be made after operation of the facility has begun.

We believe that Arkansas Power and Light Company has the desire and means to construct this power generation facility in a manner which would minimize the detrimental effect of the heated effluent.

In his statement delivered to the Committee on Natural Resources at the Southern Governors' Conference earlier this year, Governor Rockefeller stated: "Scientific capability, economic values, and aesthetics should be balanced in the relativity of pollution . . ."

We would like to stress the importance of the delicate balance of the econological systems in Dardanelle Reservoir as a basic natural resource which possesses economic as well as aesthetic values.

We make this statement in spite of the fact that we do understand that the Atomic Energy Commission has not had authority in this area.

Thank you, sir.

CHAIRMAN WELLS: Thank you very much, Dr. Suzuki.

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As you have pointed out, the Commission and this Board does not have jurisdiction over thermal effects, but we think that it has been useful for you to make this statement and we are glad to have it included in the record.

Mr. S. Ladd Davies, Director of the Arkansas Pollution Control Commission, has asked to make a limited appearance.

We are very glad to hear from you, Mr. Davies.

xxxx STATEMENT OF S. LADD DAVIES, DIRECTOR, ARKANSAS POLLUTION CONTROL COMMISSION:

MR. DAVIES: Thank you.

I am S. Ladd Davies, Director of the staff of the Arkansas Pollution Control Commission. I would like to read this into the hearing.

Secretary, United States Atomic Energy Commission,
Washington, D. C.

Dear Sir:

As Director of the Arkansas Pollution Control Commission I wish to report on the status of the application of Arkansas Power and Light Company for a disposal permit for its nuclear plant to be located on the Arkansas River near Dardanelle. Under the Arkansas Water and Air Pollution Control Act it is required that a permit be obtained from the Commission before any effluent is discharged to any of the waters of the State. Thus, the Commission must approve the

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proposed treatment or disposal system before the system is constructed and the effluent discharged. Detailed engineering plans and specifications must be submitted to the Commission as a part of the permit application.

Pursuant to these provisions of the Arkansas Water and Air Pollution Control Act, AP&L, on November 14, 1967, submitted to the Commission its application for a disposal permit for the proposed nuclear plant. Plans and specifications prepared by the engineering firm of Bechtel Corporation accompanied the application. Essentially the application seeks approval by the Commission of the proposed method of handling cooling water which will be taken from the river in enormous quantities, run through the plant for condenser cooling, and then discharged back into the reservoir. The volume of water will be in the order of 1700 cubic feet per second. The pollution effect to be avoided is the elevation of the normal temperature of the river to a level which might have adverse effects on the ecology of the river and on the use of the water for other beneficial uses.

The Commission promulgated Regulation No. 2 establishing water quality criteria for interstate streams on May 26, 1967. These criteria were subsequently approved by the Secretary of Interior pursuant to the provisions of the Water Quality Act of 1965. Regulation No. 2 contains the following temperatur criteria:

"Temperature -- The maximum temperature shall not be elevated above 20 degrees Centigrade in trout streams, 30 degrees Centigrade in smallmouth bass streams, and 35 degrees Centigrade in other streams. The temperature of a stream as determined by natural conditions shall not be increased or decreased more than 5 degrees Fahrenheit by discharges thereto."

The following p' isions of Regulation No. 2 are also pertinent. Again quote:

"The water quality criteria herein contained shall not be construed as permitting any waste amenable to treatment or control to be discharged into any waters of the State of Arkansas without reasonable treatment of control. The Arkansas Water and Air Pollution Control Act provides, among other things, that it shall be unlawful for any person to discharge any waste into any waters of the State without having first obtained a written permit from the Commission. A disposal permit may not be issued unless there is submitted to the Commission plans and specifications for a disposal system adequate to treat or control the wastes so as not to cause water pollution as defined in the Act. Such treatment or control must be

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consistent with the state of the art and best practicable industry standards, the minimum requirements being secondary treatment or equivalent, giving due regard to the quality and flow of the receiving waters, the present, future and potential uses of such waters, economic feasibility, and other relevant factors."

Following its submission and extended study of the AP&L waste disposal application was made by the Commission staff. Conferences were held between our technical people and those of AP&L and Bechtel Corporation. As a result of these studies and cooperative action, the application and plans and specifications were modified by AP&L in various respects calculated to improve the disposal arrangement, that is, to improve diffusion and dissipation characteristics in the receiving stream. The amended plans and specifications are still under review by the Commission staff. At our suggestion, AP&L has caused to be constructed a working model to test the conclusions reached by Bechtel Corporation as to temperature, diffusion of the effluent, currents, and so forth. The model is being built and constructed by the Hydro-Research-Science Company in San Jose, California, and it represents a cost in the order of $125,000.

Studies of currents and diffusion of the effluent in the receiving stream have already been initiated and

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members of the staff and the Commission have observed the model in operation. Temperature studies will begin in about 60 days.

Should the conclusions of the Bechtel study be confirmed, then it is probable that a disposal permit will be issued by the Commission based on the plans and specifications presently under submission. However, it is also probable that the permit will be conditioned upon meeting specified performance criteria so that, if in actual operation the permissible temperatures, for example, are exceeded, then AP&L will be required to make such changes or additions as may be necessary to meet the criteria.

This means that the system initially constructed must be so designed as to permit changes and additions which will meet the criteria. AP&L has indicated its willingness to make such changes and additions as may prove necessary in actual operation. In this and other respects, AP&L has cooperated with the Commission and recognized its obligation to handle its cooling water so that it will not degrade the quality of the receiving waters and prevent their use for other beneficial purposes.

The Commission has and will continue to furnish such relevant information concerning this matter to the Secretary of the Interior and the Federal Water Pollution Control Administration as they may need to discharge their

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responsibilities under the Federal Water Pollution Control Act.

In conclusion I can report that considerable progress has been made by the Commission in processing the application of AP&L and I am optimistic that a disposal permit will be issued in due course which will, on the one hand, fully protect the quality of water and, on the other, permit this great plant, with all of its economic and social benefits, to be constructed.

Yours very truly, S. Ladd Davies, Director.

CHAIRMAN WELLS: Thank you, Mr. Davies.

Again I will emphasize that while thermal effects do not come under the jurisdication of this Board, we are very glad to use this hearing as a vehicle for this excellent statement by Dr. Suzuki and Mr. Davies so that the public in this area will be informed of the progress made with respect to these items.

That completes now the statements by persons desiring to make limited appearances, and we come to the matter of identification, and swearing of applicant's panel of principal witnesses.

Do you desire to identify your panel at this time, Mr. Jewell, and have them sworn?

MR. JEWELL: Mr. Chairman, one of the witnesses will be Mr. Coen, the Chief Financial Officer of the company, eb10 who testimony is not germane to that which the panel will be discussing. Mr. Coen would like to get away from here if it possible, and I wonder if we would be permitted to put Mr. Coen on the stand at this time?

CHAIRMAN WELLS: That is agreeable to the Board, Mr. Jewell.

MR. JEWELL: Mr. Coen, will you stand and be sworn, please?

Whereupon,

A. B. COEN

was called as a witness on behalf of the Applicant and, having been first duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. JEWELL:

Q Will you please state your name, address, and position with Arkansas Power and Light Company?

A I am A. B. Coen. My business address is Arkansas Power and Light Company, 6th and Pine Street, Pine Bluff, Arkansas. I am Vice President, Treasurer and Secretary of Arkansas Power and Light Company.

Q Mr. Coen, please describe your background in education and experience which qualified you to occupy your present position.

A I graduated from high school in Mississippi and from Bowling Green Business University, Bowling Green, Kentucky, and

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have over forty years experience in the Treasury Department of Arkansas Power and Light Company.

Q What is the nature of the buisness of Arkansas Power and Light Company?

A The company is a public utility engaged in generating, transmitting and distributing electric power and energy for sale to the public, primarily in the State of Arkansas.

Q Will you please summarize the extent of the operations and facilities of the company?

A Arkansas Power and Light Company owns electric facilities in 61 of the 75 counties in Arkansas. This area includes approximately 18,200 square miles and has a population of about 1,150,000. The company serves approximately 350,000 customers. It supplies electricity directly to retail customers in 231 incorporated municipalities. It sells power and energy at wholesale to 18 customers.

The peak demand on the company's system in 1968 was 1,927,000 kilowatts, which is an increase of 12.7 percent over the maximum experienced in 1967. In the five-year period ended December 31st, 1967, the company experienced a growth in annual gross electric revenues of approximately 41 percent. During the same period of time, the average annual kilowatt hour sales per residential customer increased from 3,083 kilowatt hours to 4,695 kilowatt hours, an increase of 52.3 percent.

Arkansas Power and Light Company's properties at the

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end of 1967 included approximately 3,336 circuit miles of transmission lines, of which about 3,315 miles carry 100,000 volts or more; and approximately 23,000 miles of distribution lines. The company had 168 distribution substations with rated transformer capacity aggregating 2,683,886 kilovolt-amperes.

At September 30th, 1968, the total installed generating capability of the company was 1,717,000 kilowatts, consisting of five steam electric plants with a total installed capability of 1,642,000 kilowatts, two hydroelectric plants with a total installed capability of 69,000 kilowatts and diesel electric units having a total capability of 6,000 kilowatts. The company now has under construction an additional steam electric generating unit which will have a net capability of 530,000 kilowatts when it is completed in 1969.

Q Mr. Coen, will you please explain the relationship between Arkansas Power and Light Company and Middle South Utilities, Inc.?

A Middle South Utilities is a public utility holding company. It does not have any utility operations of its own. It owns the common stock of its four principal operating subsidiaries, Arkansas Power and Light Company, Louisiana Power and Light Company, Mississippi Power and Light Company and New Orleans Public Service Inc. Each of the operating companies is independent of the others and provides its own financing, except that Middle South buys the common stock of

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each company when additional common equity funds are needed.

The four companies do, however, plan and operate their generating and transmission facilities very much as if they were a single integrated system. This enables each company to install learger and more economical generating units than would be possible if each acted separately. For example, when this nuclear unit is completed, it will represent approximately 27 percent of the total generating capacity of Arkansas Power and Light Company, but it will be only about 9 percent of the capability of the Middle South System.

Q Mr. Coen, what are your responsibilities as Treasurer of Arkansas Power and Light Company?

A As Treasurer of the Company, I am the chief accounting and financial officer and am responsible for the preparation of the operating and cash budget, the custody and disbursement of company funds, and the raising of both short-term and long-term capital.

Q Are you familiar with the accounting procedures and the books and records of Arkansas Power and Light Company in general and particularly the financial statements filed in the company's application to the Atomic Energy Commission for authority to construct and operate the Russellville Nuclear Unit?

A Yes, I am. All of the financial books and records of Arkansas Power and Light Company are prepared and kept under

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my custody. The financial statements filed with the Application to the Atomic Energy Commission were prepared under my supervision and were taken from the books of accounts of Arkansas Power and Light Company. These books of account are ekpt in accordance with the Federal Power Commission's Uniform System of Accounts and accounting procedures prescribed by the Arkansas Public Service Commission.

I am familiar with the financial statements and statistics filed with the Application including thoese which are embodied in the Company's Annual Report to Stockholders for 1966 and Annual Report to Stockholders for 1967.

Q Do these financial statements and statistics present fairly the financial position of Arkansas Power and Light Compan as of the dates which they respectively bear?

A They do.

Q Has there been any material change in the financial condition of Arkansas Power and Light Company since July 31, 1968, the date of the latest financial statements attached to the Application?

A No. Arkansas Power and Light Company's financial condition is essentially the same.

Q Mr. Coen, what are Arkansas Power and Light Company's estimated construction budgets for the years 1968 through 1972, inclusive?

A The company's budgeted construction expenditures for the five-year period, 1968 - 1972, including the cost of nuclear fuel, are as follows:

1968	$ 52,382,000
1969	63,700,000
1970	72,744,000
1971	111,796,000
1972	50,060,000

It must be understood that these construction budgets are tentative and must be accepted as being subject to change. They do, however, include all of the major items of property which we now expect to construct during these years.

Q Will you state how Arkansas Power and Light Company plans to finance the construction of the Russellville Nuclear Unit?

A Arkansas Power and Light Company expects to finance Russellville Nuclear Unit as an integral part of its normal construction program for plants and necessary attendant facilities. This will involve the use of funds internally generated and funds derived from the sale of various securities in the same manner as is done with conventional plant facilities and additions.

Our present estimates indicate that the construction costs for the nuclear unit, including the initial cost of fuel, will be $169,000,000. In my opinion, based upon Arkansas Power and Light Company's past record of earnings, depreciation

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accruals, and cash dividend distributions, and assuming the continuation of the current level of earnings, it is reasonable to expect that a substantial portion of the cost of this unit will be provided by internal sources such as earnings and depreciation.

Furthermore, it is my opinion, that in view of the size of Arkansas Power and Light Company's resources, the strength of its financial position, its earnings record, and the regard held for its A-rated bonds and its preferred stock in the financial markets, we would expect that the company would have little difficulty in selling sufficient securities in the form of preferred stock and first mortgage bonds, or whichever type of security would be the most prudent at the time to provide the remaining funds needed to finance the contemplated nuclear plant construction.

In addition, of course, we will from time to time sell additional common stock to Middle South Utilities when that method of financing appears appropriate. The amount and type of securities to be issued cannot be determined at this time. The proper type of security and the proper amounts of securities will be issued from time to time to maintain sound capitalization ratios.

Q Mr. Coen, can you state the facts which would most strongly reflect the fact that Arkansas Power and Light Company is sound financially and has the financial qualifications to

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construct and operate the Russellville Nuclear Unit?

A The first mortgage bonds and sinking fund debentures represent approximately 58 percent of the total capitalization of the company and the proprietary capital represents approximately 42 percent of the total capitalization. The numb er of times interest was earned after Federal income taxes for the 12-month period ended July 31st, 1968, was 2.65. The number of times total preferred stock dividends were earned during the same 12-month period was 7.77. The company's current Dun and Bradstreet credit rating is AaA1. Moody's Investor Service rates the company's first mortgage bonds as A (high-medium grade).

Q How would Arkansas Power and Light Company finance a permanent shutdown of the nuclear generating plant?

A I have been advised by our Nuclear Project Manager that upon the company's construction and operation of the Russellville Nuclear Unit pursuant to the construction license to be issued by the Atolmic Energy Commission, the plant will be safe to the public as required by the Atomic Energy Act. Therefore, when the nuclear plant is ultimately shut down, the relatively small expense that will be necessary to continue the safe condition of the plant will be so small with reference to the annual general revenues that such expenditure may be readily financed by the Company either through internal cash generation or as a part of a normal permanent financing program.

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Q Mr. Coen, would you briefly detail for this Board the plans for, and the present status of, Arkansas Power and Light Company's efforts to obtain all required property and liability insurance for the Russellville Nuclear Unit, as well as for its nuclear fuel?

A The insurance section of our company has been thoroughly investigating this insurance and has been consulting with Rebsamen and East, Inc., of Little Rock, Arkansas, who will be our principal agent and consultant on matters pertaining to nuclear liability and nuclear property insurance for this plant.

Arkansas Power and Light Company will fully comply with the requirements of the Atomic Energy Act of 1954, as amended, and the applicable Rules and Regulations of the Atomic Energy Commission.

As a condition to the granting of the operating license for the Russellville Nuclear Unit, Arkansas Power and Light Company will purchase nuclear liability insurance and nuclear property insurance in the amounts of $74,000,000 and $74,000,000, respectively, from the available nuclear insurance pools, or in such other amount or amounts as may be lawful at the time. Upon delivery of the nuclear fuel elements to the plant site, but prior to their being loaded into the reactor, Arkansas Power and Light Company will purchase nuclear liability insurance in the amount of $1,000,000 from one of the

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available nuclear insurance pools.

I might add that in addition to the foregoing nuclear insurance which Arkansas Power and Light Company will purchase, it will also enter into an Indemnification Agreement with the Atomic Energy Commission for the protection of the public in the amount of $486,000,000, or such other amount as may be prescribed by law.

Q In your opinion, can Arkansas Power and Light Company finance the Russellville Nuclear Unit without jeopardizing the financial integrity and structure of the company?

A Yes. Construction of the Russellville Nuclear Unit can be financed without any material adverse change in the financial structure of the company.

Q Mr. Coen, in your opinion, does Arkansas Power and Light Company have now, and is it reasonable to assume that it will have in the future, the resources to construct and operate the Russellville Nuclear Unit in an appropriate manner and to pay all charges and expenses therefor?

A Yes. I have become somewhat familiar with the nature of the company's new type nuclear generating station, together with the expenses that will be incurred during its construction. I am confident that the company has sufficient resources to carry out this enterprise.

Q Mr. Coen, are any of the officers or directors of Arkansas Power and Light Company either residents or citizens

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of a foreign country?

A No. They are all citizens and residents of the United States.

Q Who owns the controlling voting stock of Arkansas Power and Light Company?

A The only class of stock of Arkansas Power and Light Company which has general voting rights is the common stock. All of this common stock is owned by Middle South Utilities, Inc. As long as we continue to be a subsidiary of Middle South, that company will own all of our common stock.

Q Are any of the officers or directors of Middle South Utilities, Inc. residents or citizens of a foreign country?

A No. They are all residents and citizens of the United States.

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MR. JEWELL: This concludes Mr. Coen's direct testimony.

CHAIRMAN WELLS: Mr. Jewell, do you desire to have your other witnesses sworn at this time?

MR. JEWELL: Yes, sir.

Will there be any cross examination of Mr. Coen?

CHAIRMAN WELLS: Does the regulatory staff desire to make any cross examination?

MR. ENGELHARDT: Mr. Chairman, the regulatory staff has had available the prepared testimony of Mr. Coen which Mr. Coen read into the record this morning and has had had an opportunity last even to review this and determine that it has no cross examination questions to raise of Mr. Coen.

CHAIRMAN WELLS: Thank you very much.

May we proceed to your other witnesses.

MR. JEWELL: May Mr. Coen be excused?

CHAIRMAN WELLS: You are excused, Mr. Coen. Thank you very much.

(Witness excused.)

MR. JEWELL: The primary panel of witnesses for the applicant will be composed of Mr. Harlan T. Holmes -- will these gentlemen stand and come forward to these seats here.

CHAIRMAN WELLS: Will the gentlemen who compose the panel of witnesses please come forward here?

MR. JEWELL: Mr. Chairman, the panel consists of Mr. Harlan T. Holmes, Dr. Knox M. Broom Mr. James McFarland, Mr. Robert E. Wascher, Mr. William R. Smith, Mr. Harry P. Marsh, Mr. R. Paul Schmitz, Mr. Howard W. Wahl.

Would it be appropriate for all of these gentlemen to be sworn at the same time?

CHAIRMAN WELLS: Yes.

Now that you gentlemen have seated yourself in the proper order, I wonder if you would be kind enough to stand and take the oath.

Whereupon,

HARLAN T. HOLMES

KNOX M. BROOM

JAMES MC FARLAND

ROBERT E. WASCHER

WILLIAM R. SMITH

HARRY P. MARSH

R. PAUL SCHMITZ

and

HOWARD W. WAHL

were called as witnesses on behalf of the applicant and, having been first duly sworn were examined and testified as follows:

CHAIRMAN WELLS: You may be seated.

Am I right in assuming that the Board has or will have the qualifications, the technical qualifications, of your witnesses?

MR. JEWELL: We have delivered these to the Board last evening -- the qualifications.

CHAIRMAN WELLS: Thank you very much, Mr. Jewell, will you proceed.

MR. JEWELL: Mr. Chairman, would it be permissible for me to have witnesses identify their own statements of their qualifications and insert those qualifications in the record as if read without actually reading them.

CHAIRMAN WELLS: That will be satisfactory, Mr. Jewell.

DIRECT TESTIMONY

MR. JEWELL: I will ask Mr. Holmes, have you prepared the statement of your educational and professional qualifications.

MR. HOLMES: Yes.

MR. JEWELL: Are those qualifications set forth in a document entitled "Educational and Professional Qualifications of Harlan T. Holmes, Nuclear Project Manager, Arkansas Power and Light Company?"

MR. HOLMES: Yes.

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ATOMIC ENERGY COMMISSION

IN THE MATTER OF	)	
	)	
ARKANSAS POWER & LIGHT COMPANY	)	Docket No. 50-313
	)	
(Russellville Nuclear Unit)	)	

EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
HARLAN T. HOLMES
NUCLEAR PROJECT MANAGER
ARKANSAS POWER & LIGHT COMPANY

1. My name is Harlan T. Holmes. My residence address is 1105 North Bryan, Little Rock, Arkansas. I am employed by Arkansas Power & Light Company as Assistant Manager of the Production Department and Nuclear Project Manager.

2. I graduated from the University of Arkansas with a degree of Bachelor of Science in Mechanical Engineering in 1944 and with a degree of Bachelor of Science in Electrical Engineering in 1947. During the past two years I have continued some post graduate study in

Nuclear Engineering at the University of Arkansas' Graduate Institute of Technology.

3. In 1947 I was first employed by Arkansas Power & Light Company as a Cadet Engineer and worked in that position until 1950.

4. I was promoted to Assistant Plant Superintendent of the Harvey Couch Generating Plant of Arkansas Power & Light Company in 1950. In this capacity I supervised the operation, maintenance and repair of this steam-electric generating facility. I continued in this position until 1953.

5. In 1953 I was appointed Assistant Manager of the Production Department of Arkansas Power & Light Company and have continued in that position to the present time. In this capacity I have had responsibility for plant efficiency and performance tests and for construction supervision and testing at all of the Company's generating plants.

6. In 1967 I was named Nuclear Project Manager for Arkansas Power & Light Company. Since that time I have

had the chief responsibility for the design and licensing of the Russellville Nuclear Unit and have devoted substantially full time to this project.

7. I am a member of the Edison Electric Institute's Committee on Nuclear Fuels and have participated in the work of the Plutonium Task Force engaged in a plutonium survey and development of studies on plutonium recycle in thermal reactors. I am a member of the American Society of Mechanical Engineers, the Institute of Electrical and Electronic Engineers, American Nuclear Society and am the company alternate representative to Atomic Industrial Forum. I am a registered professional Engineer in Arkansas.

MR. JEWELL: I will ask Dr. Knox Mr. Broom, have you prepared a statement on your educational and professional qualifications?

DR. BROOM: Yes.

MR. JEWELL: Are those qualifications set forth in a document entitled "Educational and Professional Qualificati ns of Knox . Broom, Jr., Nuclear Specialist, Middle South Services, Inc.?"

DR. BROOM: Yes.

MR. JEWELL: Are all the statements and facts contained in that statement true and correct?

DR. BROOM: Yes.

MR. JEWELL: Mr. Chairman, I would like to have this statement of qualifications inserted in the transcript of evidence bodily and treated as the evidence of this witness as if read.

CHAIRMAN WELLS: It is so ordered.

(The document follows.)

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
KNOX M. BROOM, JR.
NUCLEAR SPECIALIST
MIDDLE SOUTH SERVICES, INC.

1. My name is Knox M. Broom, Jr. My home address is 1767 Pace Blvd., New Orleans, Louisiana. I am employed by Middle South Services,Inc. in the Engineering Department as a Nuclear Specialist.

2. I received a Bachelor of Arts Degree from the University of Southern Mississippi in Chemistry and Mathematics in 1958. In 1961 and 1963, respectively, I received my M.S. and Ph.D. degrees from the University of Arkansas in Nuclear Chemistry. My theses work was on high energy fission of U^{238} and Th^{232}.

3. From June 1963 to July 1966 I was a Senior Chemist

with the Atomics International Division of North American Aviation, Inc., Canoga Park, California. My responsibilities included nuclear fuel burnup analysis, activation analysis, fuel performance evaluations, and consulting work for the Hallam, Piqua, and SNAP reactor projects. I also supervised the Radiochemistry and Nuclear Spectroscopy Laboratories. I was member of the Burnup Task Force of the American Society for Testing and Materials.

4. In 1961 I joined the U. S. Atomic Energy Commission, Germantown, Maryland in the Fuels and Materials Branch of the Division of Reactor Development and Technology. As a technical administrator I supervised research and development contracts primarily on fast breeder reactor fuels. I was a member of the Plutonium Research Coordinating Committee and was U. S. Coordinator for the Libby-Cockcroft Exchange on Plutonium Recycle.

5. In September 1967 I joined Middle South Services Inc. in my present position. My primary responsibility is to provide technical assistance to any company in the

Middle South System engaged in nuclear activities. At present I am working full-time on the technical and licensing aspects of the Russellville Nuclear Unit. I am a member of the Design Review Board for the project.

6. I am a member of Sigma Xi, the American Nuclear Society, and the American Chemical Society.

MR. JEWELL: I will ask this question of Mr. R. Paul Smith.

Mr. Schmitz, have you prepared a statement of your educational and professional qualifications?

MR. SCHMITZ: Yes, I have.

MR. JEWELL: Are those qualifications set forth in a document entitled, "Educational and Professional Qualifications of R. Paul Schmitz, Chief Nuclear Engineer, Power and Industrial Division, Bechtel Corporation?"

MR. SCHMITZ: Yes.

MR. JEWELL: Are all the statements and facts contained in that statement true and correct?

MR. SCHMITZ: Yes, they are.

MR. JEWELL: Mr. Chairman, I would like to have this statement of qualificatins inserted bodily into the record and be a part of the testimony of this witness as if read.

CHAIRMAN WELLS: It is agreed.

(the document follows.)

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
R. PAUL SCHMITZ
CHIEF NUCLEAR ENGINEER
POWER AND INDUSTRIAL DIVISION
BECHTEL CORPORATION

1. My name is R. Paul Schmitz. My residence is 715 Butternut, San Rafael, California. I am Chief Nuclear Engineer for the Power and Industrial Division in San Francisco.

2. I graduated from the Missouri School of Mines in 1950 with a Bachelor's Degree in Chemical Engineering. In 1959, I was awarded a Masters Degree in Engineering Administration from the George Washington University. I have been associated with the nuclear industry since 1950.

3. During the four-year period, from 1950 to 1954, I was employed by the General Electric Company at the Hanford Atomic Products Operation in Richland, Washington. This assignment involved the analysis and testing of the fuel and coolant systems for the Hanford Production Reactors.

4. In 1956 through 1959, I was employed by the U. S. Atomic Energy Commission's Division of Reactor Development with responsibility for coordination and direction for technical aspects of AEC sponsored organic cooled civilian reactor projects.

5. In 1959, I was employed by the Bechtel Corporation as an engineer in the Scientific Development Department. My major assignments were with the project design groups for the Peach Bottom Atomic Power Station Unit 1, the San Onofre Nuclear Generating Station, the Space Environmental Test Chamber for the NASA Houston Manned Spacecraft Center, the Fast Reactor Test Facility for Argonne National Laboratory and the Muhleberg Nuclear Unit in Switzerland.

6. In 1967 I transferred to the Power and Industrial Division and became the Chief Nuclear Engineer for the San Francisco Office.

7. I am registered as a Professional Engineer in the State of Missouri and am a member of the American Nuclear Society and the Health Physics Society.

MR. JEWELL: I will ask Mr. Harry P. Marsh.

Mr. Marsh, have you prepared a statement of your educational and professional qualifications?

MR. MARSH: Yes.

MR. JEWELL: Is that statement embodied in an instrument entitled "Educational and Professional Qualifications, Harry P. Marsh, Project Engineer, Power and Industrial Division, Bechtel Corporation?"

MR. MARSH: Yes.

MR. JEWELL: Are all the statements and facts contained in that instrument true and correct?

MR. MARSH: They are.

MR. JEWELL: I would like to have this witness' statement inserted bodily into the transcript of the testimony and treated as part of the testimony of this witness as if read.

CHAIRMAN WELLS: It is agreed.

(The document follows.)

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
HARRY P. MARSH
PROJECT ENGINEER
POWER AND INDUSTRIAL DIVISION
BECHTEL CORPORATION

1. My name is Harry P. Marsh. My residence is 1 Cowper Avenue, Kensington, California 94707. I am employed by the Bechtel Corporation, Power and Industrial Division, San Francisco, California.

2. I graduated from the University of California at Berkeley in 1943 with a Bachelor of Science Degree in Mechanical Engineering, Heat Power Option.

3. Upon graduation I worked for two months for the U. S. Corps of Engineers, San Francisco District, in soil

mechanics testing in the field and laboratory.

4. The balance of 1943 I worked with the Chas. M. Bailey Co., San Francisco, as a sales engineer in the field of steam specialties.

5. Through the years of 1944 and 1945 I worked with the Permanent Metals Corp. as a field engineer involved in all phases of the construction and testing of Liberty and Victory Ships.

6. In March 1946 I joined the Bechtel Corporation as an Assistant Engineer involved in the specification and selection of heat exchangers, pumps, mechanical drive turbines, instruments and control valves for petroleum refineries and process steam generating plants.

7. From July 1952 to July 1954 I resided in England as an Instrument Engineer for the Refinery Division of Bechtel Corp. involved in the design and construction of the Aden Refinery for the Anglo-Iranian Oil Co.

8. From July 1954 to present I have been in the Power and Industrial Division of the Bechtel Corp. involved in the design, procurement, construction and start-up of

six conventional steam electric power plants ranging in size from 25,000 KW to 550,000 KW supercritical units. I was made Assistant Project Engineer in September 1955 and Project Engineer in April 1957.

9. I have been a registered Professional Engineer in Mechanical Engineering in the State of California since June 30, 1948, certificate number 3905, and similarly in the State of Arkansas since November 27, 1967, certificate number 2664.

MR. JEWELL: I will address this question to Mr. Howard W. Wahl.

Mr. Wahl, have you prepared a statement of your educational and professional qualifications?

MR. WAHL: Yes, I have.

MR. JEWELL: Is that statement embodied in the instrument entitled "Educational and Professional Qualifications, Howard W. Wahl, Project Engineer, Containment Design Group, Power and Industrial Division, Bechtel Corporation?"

MR. WAHL: Yes.

MR. JEWELL: Are the statements and facts contained in that statement true and correct?

MR. WAHL: Yes, they are.

MR. JEWELL: Mr. Chairman, I would like to have this statement inserted in the transcript of the testimony and treated as evidence of this witness to the same extent as if read.

CHAIRMAN WELLS: Agreed.

(The document follows.)

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
HOWARD W. WAHL
PROJECT ENGINEER
CONTAINMENT DESIGN GROUP
POWER AND INDUSTRIAL DIVISION
BECHTEL CORPORATION

1. My name is Howard W. Wahl. My residence is 865 Solana Drive, Lafayette, California. I am employed by Bechtel Corporation, San Francisco, California.

2. I graduated from the University of Washington in 1956 with a Bachelor of Science degree in Civil Engineering. Upon graduation I joined the Power and Industrial Division of Bechtel Corporation.

3. My experience includes civil and structural design on

the Dresden, Humboldt Bay and Peach Bottom nuclear power plants as well as fossil fueled plants. Upon completion of the design phase of the Humboldt Bay plant I served as the jobsite civil field engineer for a period of nearly two years.

4. Other responsibilities have included:

   a. Contributing to the AEC Reactor Containment Handbook ORNL-N51C-5.

   b. Structural design on the FARET Project and the Savannah River Power Conversion Study.

   c. Civil and structural portion of numerous commercial nuclear power plant proposals and studies for both U. S. and foreign power companies.

5. I am now the Project Engineer of the Containment Design Group, Power and Industrial Division. This Design Group is responsible for the formulation of design criteria, the structural analysis and design, and the material specifications for the post-tensioned concrete containment structures for the Palisades Plant, Turkey Point Units #3 and #4 and Point Beach Units #1 and #2.

The group also serves as technical consultant on the above items for Arkansas Nuclear One, Oconee Units #1, #2 and #3 and the Rancho Seco Plant.

6. I am a member of the American Society of Civil Engineers and a Registered Professional Civil Engineer in the State of California.

MR. JEWELL: I will direct this question to Mr. James McFarland.

Mr. McFarland, have you prepared a statement of your educational and professional qualifications?

MR. MC FARLAND: Yes, I have.

MR. JEWELL: Are those statements included in a document entitled "Educational and Professional Qualifications, James McFarland, Project Manager, Nuclear Power Generating Department, Power Generation Division, the Babcock and Wilcox Company?"

MR. MC FARLAND: Yes.

MR. JEWELL: Are the statements contained therein true and correct?

MR. MC FARLAND: Yes, they are.

MR. JEWELL: Mr. Chairman, I request that this statement be incorporated bodily into the transcript of the testimony and treated as the testimony of Mr. Mc Farland the same as if read.

CHAIRMAN WELLS: Agreed.

(The document follows.)

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
JAMES McFARLAND,
PROJECT MANAGER,
NUCLEAR POWER GENERATING DEPARTMENT
POWER GENERATION DIVISION
THE BABCOCK & WILCOX COMPANY

1. My name is James McFarland. My residence address is 2105 Burnt Bridge Road, Lynchburg, Virginia, 24503. I am employed by The Babcock & Wilcox Company, Power Generation Division, Nuclear Power Generation Department, as a Project Manager.

2. I served in the U. S. Navy Reserve from February 1943 through February 1946, Ensign rank - honorable discharge.

3. I was graduated from Carnegie Institute of Technology in 1948 with a Degree in Mechanical Engineering.

4. In 1948 I began working for The Babcock & Wilcox Company as a Student Engineer. A year later I was assigned to the Field Engineering Section of the Boiler Division as a Service Engineer in the Pittsburgh district.

5. In 1956 I transferred to B&W's Boiler Division headquarters at Barberton, Ohio working as a Contract Supervisor.

6. In 1967 I transferred to the Nuclear Power Generation Department as an Assistant Project Manager and later that year was assigned to the Russellville Nuclear Unit as a Project Manager for B&W.

7. I am a registered Professional Engineer in the States of Ohio and Pennsylvania.

MR. JEWELL: Mr. Robert E. Wascher.

Mr. Wascher, have you prepared a statement of your educational and professional qualifications?

MR. WASCHER: Yes, I have.

MR. JEWELL: Is that statement contained in an instrument entitled "Educational and Professional Qualifications. Robert E. Wascher, Manager, Nuclear Safety Engineering Section, Nuclear Power Generation Department, Power Generation Division, The Babcock and Wilcox Company?"

MR. WASCHER: Yes, it is.

MR. JEWELL: Are the facts set out in that statement true and correct?

MR. WASCHER: Yes, they are.

MR. JEWELL: Mr. Chairman, I would like to have this statement embodied in the transcript of the testimony in this case and treated as the testimony of this witness the same as if read.

CHAIRMAN WELLS: Agreed.

(The document follows.)

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ATOMIC ENERGY COMMISSION

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
ROBERT E. WASCHER
MANAGER, NUCLEAR SAFETY ENGINEERING SECTION
NUCLEAR POWER GENERATION DEPARTMENT
POWER GENERATION DIVISION
THE BABCOK & WILCOX COMPANY

1. My name is Robert E. Wascher. My residence is 1916 Eastwood Lane, Lynchburg, Virginia, 24503. I am employed by The Babcock & Wilcox Company, Power Generation Division, in the Nuclear Power Generation Department.

2. I graduated from the Illinois Institute of Technology in 1952 with a Bachelor of Science Degree in Mechanical Engineering. In 1953 I graduated from the Oak Ridge School of Reactor Technology.

3. Upon graduation, I joined the Oak Ridge National

Laboratory as an Associate Development Engineer responsible for the development of mechanical components for homogeneous nuclear reactors.

4. In 1955 I was commissioned an officer in the U. S. Navy. During my naval service, I served as the Navy Liaison Officer in the Army Package Power Reactor Program. I was also assigned to the Navy's Bureau of Yards and Docks with responsibility for nuclear engineering problems of the Bureau.

5. In 1958 I joined The Babcock & Wilcox Company as a Nuclear Engineer with responsibility for the safety analysis of the Consolidated Edison Company's Indian Point No. 1 Nuclear Plant. In 1959 I was appointed Supervisor of the Safety Analysis Group with responsibility for safety analysis of nuclear plants designed by B&W. In 1964 I became Chief of the Operational Analysis Section with responsibility for reactor and system dynamic analysis, reactor control analysis, plant performance, and safety analysis.

6. In 1965 I was appointed Manager of the Nuclear Safety Section, my present position. In this position I am

responsible for safety and licensing of the plants designed by B&W.

7. During 1964 and 1965 I was Chairman of the N.S. Savannah Safety Committee, a committee responsible for periodic review of the operation of the N.S. Savannah. From 1962 to 1966 I was also Chairman of B&W's Nuclear Development Center Safety Review Board. In 1966, I was appointed to the Atomic Energy Commission's Advisory Task Force on Power Reactor Emergency Cooling. In addition, I am a member of the American Nuclear Society and the Atomic Industrial Forum's Safety Steering Committee.

8. I am a registered Professional Engineer in the State of Virginia.

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MR. JEWELL: Mr. William R. Smith.

Mr. Smith, have you prepared a statement of your educational and professional qualifications?

MR. SMITH: Yes, I have.

MR. JEWELL: Is that statement embodied in an instrument entitled "Educational and Professional Qualifications, William R. Smith, Supervisor, Licensing Group, Nuclear Safety Engineering Section, Nuclear Power Generation Department, The Babcock & Wilcox Company?"

MR. SMITH: Yes.

MR. JEWELL: Are the facts set forth in that statement true and correct?

MR. SMITH: Yes, they are.

MR. JEWELL: Mr. Chairman, I would request that this document identified by the witness be embodied into the record of the testimony and treated as the testimony of this witness the same as if read.

CHAIRMAN WELLS: Agreed.

(The document follows.)

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EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
WILLIAM R. SMITH
SUPERVISOR, LICENSING GROUP
NUCLEAR SAFETY ENGINEERING SECTION
NUCLEAR POWER GENERATION DEPARTMENT
THE BABCOCK & WILCOX COMPANY

1. My name is William R. Smith. My residence is 3750 Woodside Avenue, Lynchburg, Virginia, 24503. I am employed by The Babcock & Wilcox Company, Power Generation Division, in the Nuclear Power Generation Department.

2. I graduated from the United States Naval Academy in 1945 with a Bachelor of Science Degree in Marine Engineering and was commissioned an officer in the U. S. Navy. I served through 1947 as a junior gunnery officer at sea and as a Radiological Safety Officer at

the San Francisco Naval Radiological Defense Laboratory.

3. In 1948 I resigned my commission in the U. S. Navy and joined the Health Physics organization being formed at the Gaseous Diffusion Plant in Oak Ridge. In 1950 I was made supervisor of the Health Physics Group at that plant.

4. In 1951 I was recalled to active duty with the U. S. Naval Reserve and assigned duties of a classified nature in the Special Weapons program.

5. In 1955 I joined The Babcock & Wilcox Company as a nuclear engineer in shielding design. I participated in the shield design work for the Consolidated Edison Company's Indian Point No. 1 Nuclear Plant and was assigned as lead engineer for shield design for the Nuclear Merchant Ship Reactor for N. S. Savannah. In 1960 I was appointed Supervisor of the Shielding Design Group with responsibility for basic reactor radiation analysis and shield design activities for plants designed by B&W.

6. In 1963 I was assigned to six months of specialized training as nuclear advisor for the anticipated foreign

voyages of N. S. Savannah. I subsequently sailed in this capacity on the first two foreign voyages of the ship under its general agency charter, and later on its first foreign voyage in commercial service on lease, with responsibilities for advising the master with respect to nuclear and regulatory aspects of reactor plant operation.

7. In 1965, I was transferred to the B&W Atomic Energy Division's Marketing Department as coordinator for Marine Markets, with responsibility for coordinating sales and promotional efforts with engineering design in B&W's maritime reactor activities.

8. In 1967 I was assigned as Supervisor, Licensing Group, Nuclear Safety Engineering Section, with direct responsibility for all licensing activities in reactor plants designed by B&W.

9. In 1965 - 1967 I served as a member of The Babcock & Wilcox Company Nuclear Development Center Isotopes Committee. I have served as Vice Chairman of the North Carolina-Virginia chapter of the American Nuclear

Society. I am a member of the American Nuclear Society, the Society of Naval Architects and Marine Engineers, Panel M-13 (Atomic Energy) of the Ships Machinery Committee of SNAME, and of the Reactor Safety Committee of the Atomic Industrial Forum.

MR. JEWELL: Mr. Chairman, that completes the qualifications of the panel for the applicant.

CHAIRMAN WELLS: Are you prepared to proceed on Item 13 of our agenda?

MR. JEWELL: Yes, sir.

CHAIRMAN WELLS: Very well.

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MR. JEWELL: Mr. Harlan Holmes has been sworn and he will be our first witness.

Mr. Holmes, have you in accordance with the rules of practice of the Atomic Energy Commission, caused to be prepared under your supervision a Summary Description of the Application of Arkansas Power and Light Company in this case?

MR. HOLMES: Yes, I have.

MR. JEWELL: Is that description contained in the document which is entitled "Summary Description of Application for Reactor Construction Permit and Operating License" dated September 28th, 1966?

MR. HOLMES: Yes, it is.

MR. JEWELL: Mr. Holmes, are all of the facts set forth in this document true and correct?

MR. HOLMES: Yes, they are.

MR. JEWELL: Do you now adopt this Summary Description as part of your testimony in this case?

MR. HOLMES: Yes, I do.

MR. JEWELL: Mr. Chairman, a copy of this Summary Description has previously been handed to all members of the Board and to the staff and Counsel for the staff. We would like to ask permission at this time that it be inserted bodily into the transcript of the record and be treated as the testimony of Mr. Holmes, to the same extent as if he read it word for word in this hearing.

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CHAIRMAN WELLS: It is agreed. The document will be so inserted.

(Summary Description follows:)

BEFORE THE

UNITED STATES ATOMIC ENERGY COMMISSION

In the Matter of

ARKANSAS POWER & LIGHT COMPANY

RUSSELLVILLE NUCLEAR UNIT   Docket No. 50-313

SUMMARY DESCRIPTION OF APPLICATION FOR REACTOR CONSTRUCTION PERMIT AND OPERATING LICENSE

September 28, 1968

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1. INTRODUCTION

This document is a Summary Description of the Application, as supplemented by Supplements 1 through 10, of Arkansas Power & Light Company (referred to as "the Applicant") for a construction permit and facility license to construct and operate the Russellville Nuclear Unit on a peninsula in Dardanelle Reservoir on the Arkansas River in Pope County, Arkansas. This Summary Description includes information on the site and environment, a description of the Russellville Nuclear Unit, analyses of the safety aspects of the plant, a summary of quality assurance procedures, a summary of the research and development programs necessary for the final design, the technical qualifications of the Applicant and its principal contractors and considerations relating to the common defense and security of the United States.

This Summary Description will constitute a portion of the prepared testimony of the Applicant to be presented at its hearing before the Atomic Safety and Licensing Board and is therefore being sponsored by an Arkansas Power & Light Company witness, Mr. Harlan T. Holmes, Assistant Manager of Production and Nuclear Project Manager.

To assist Mr. Holmes in answering questions on cross-examination by the Board or another party, several technical witnesses representing the Applicant, its engineers and contractors will make up a panel of technical expert witnesses whose unprepared testimony will become a part of the Applicant's testimony before the Board.

The Russellville nuclear generating unit will employ a pressurized water nuclear steam supply system furnished by The Babcock & Wilcox Company (referred to as "B&W") and is similar in design to the nuclear steam supply systems which are being furnished by B&W to Duke Power Company for its Oconee Nuclear Station (AEC Docket Nos. 50-269, -270 and -287), Metropolitan Edison Company for the Three Mile Island Nuclear Station (AEC Docket No. 50-289), Florida Power Corporation for the Crystal River Plant Unit 3 (AEC Docket No. 50-302) and Sacramento Municipal Utility District for its Rancho Seco Nuclear Generating Station, Unit No. 1 (AEC Docket No. 50-312). A construction permit authorizing construction of the Oconee facilities was issued in

November 1967 and a construction permit authorizing construction of the Three Mile Island Nuclear Station was issued in May 1968, both pursuant to Section 104 (b) of the Atomic Energy Act of 1954, as amended. The nuclear steam supply system will operate initially at core power levels up to 2452 MWt, which corresponds to a gross electrical output of about 850 MWe. An ultimate core output of 2568 MWt is expected, and all steam and power conversion equipment is designed accordingly. All plant safety systems, including containment and engineered safeguards, are designed and evaluated for operation at this higher power level. The higher power level is also used in the analyses of postulated accidents to establish the suitability of the site under the guidelines set forth in 10 CFR 100.

The Applicant's construction permit application including the supplements thereto, has been reviewed by staff of the Atomic Energy Commission, which has prepared a safety analysis of the Application. The Advisory Committee on Reactor Safeguards (referred to as "ACRS") has also reviewed the Application, as amended through Supplement No. 9, and reported its findings to the

Chairman of the U. S. Atomic Energy Commission in a letter dated September 12, 1968. The ACRS concluded "the proposed reactor can be constructed at the Russellville site with reasonable assurance that it can be operated without undue risk to the health and safety of the public." The AEC staff concluded similarly.

The principal architectural and engineering criteria which will govern the plant design are set forth in Section 1.4 of the Volume I and Supplement No. 1 of the Applicant's Preliminary Safety Analysis Report. These criteria together with the engineered safeguards and other incorporated systems provide assurance that the proposed Russellville Nuclear Unit can and will be constructed and operated at the proposed location without undue risk to the health and safety of the public.

2. DESCRIPTION OF SITE AND ENVIRONMENTAL CHARACTERISTICS WHICH INFLUENCE DESIGN

2.1 Location

The Russellville Nuclear Unit will be constructed in the Southwestern part of Pope County, State of Arkansas. The site of the unit is located six miles West North-West of Russellville and 57 miles Northwest of Little Rock, as shown in Figure 1, Appendix B. The site and immediate vicinity are shown in Figure 2, Appendix B.

All land comprising the site will be controlled to the extent necessary by Arkansas Power & Light Company. This area includes certain portions of the bed and banks of Dardanelle Reservoir which are owned by the United States. An easement has been obtained which entitles the Applicant to exclude all persons from these areas during periods when Applicant feels it is advisable.[(1)] Land use is shown in Figure 3, Appendix B and dairy animal population is shown in Figure 4, Appendix B.

2.2. Population

The site exclusion area, which is under control of the Applicant, has a minimum radius of 0.65 mile. The distance to the boundary of the low population zone has been established as four miles.[2] The nearest population center of 25,000 or more is Hot Springs, located 55 miles South of the site. There are no population centers of 25,000 or more located within a 50-mile radius of the site.

It is expected that the Dardanelle Reservoir will be a major contributing factor to the part-time population within a five mile radius. It is anticipated that the 75 miles (approximately) of shoreline of the Dardanelle Reservoir and Arkansas River will be developed as recreational areas and week-end and holiday population will increase.[3] Figure 5, Appendix B shows this estimated transient population within five miles of the plant site in 2012.

2.3 Meteorology

The site meteorology has been extensively investigated to provide an assessment of environmental consequences of routine and accidental releases of radioactivity. The climate of the Arkansas River Valley in the region of the site is primarily continental in character. The Boston

Mountains, with elevations up to 2700 feet and oriented generally east-west on the north side of the valley, have an influence on the annual precipitation. The annual precipitation on the south slope is on the order of 2.4 inches greater than in the valley. Within the valley, in an east-west direction, the climatology is homogeneous. A study was made of the site atmospheric diffusion characteristics, utilizing conservative meteorological conditions.[4] A meteorological program for the site was initiated in the Fall of 1967.

2.4 Surface Water Hydrology

In connection with the safety aspects of the proposed nuclear power plant, surface water investigations were made. These included the source and dependability of the cooling water supply, magnitudes of possible floods and possible failure of upstream dams.[5]

The plant will require 1700 cfs cooling water. This water will be taken from the Dardanelle Reservoir downstream east of the plant. The discharge will flow into the Arkansas River southwest of the plant. No domestic water supply is taken downstream of the plant to the mouth of the Arkansas River.[5]

The minimum pool elevation in Dardanelle Reservoir is 336 feet. The highest experienced flood occurred in 1943, with a peak flow of 683,000 cfs. The levees along the river channel in this area are generally designed for flow of 830,000 cfs. The Dardanelle Dam is designed to hold a water level no higher than 338 feet and to discharge 900,000 cfs. The maximum probable flood level was computed by the Corps of Engineers as 1,500,000 cfs with 358 feet flood level. Failure of Ozark Dam, immediately upstream from Dardanelle Dam, during a maximum probable flood would result in a maximum 361 foot water level at the site. Nominal plant grade elevation will be 353 and ground floor elevation for the building will be 354. During a maximum probable flood the plant will be shut down. All Class I structures are designed to resist this flood and all Class I equipment is either located above elevation 361 ft. or protected from flooding by the Class I structures. (Access to the plant would be by boat and/or helicopter.) The minimum daily average flow computed by the Corps of Engineers during the driest critical month of the year is 4,000 cfs.[5]

2.5 Ground Water Hydrology

The site is located on compact clayey soil overlying dense shale bedrock and adjacent to the Dardanelle Reservoir. This clayey overburden is generally impermeable and hence ground water is not available. Ground water is available in the bedrock fracture systems. It is confined water which flows toward the reservoir under a relatively flat gradient.[6]

Water discharged at the surface and ponded will percolate very slowly downward through the clayey soil overburden while migrating toward the reservoir. In the unlikely event of an accident, the clayey soils at the site will react with any dissolved radionuclides and inhibit their migration.[6]

The proximity of the site to the Dardanelle Reservoir will not adversely affect construction conditions. Domestic wells obtain supplies from confined water in bedrock which is under pressure. Thus infiltration from the surface is not a problem.[6]

2.6 Geology

The recent exploration program which included core, auger, and wash-bore holes in addition to geologic mapping, a geophysical survey, and testing program were sufficient to delineate the foundation conditions relative to construction of the proposed plant. The exploration and testing program enabled construction design criteria to be formulated.[7]

Critical structures will utilize the underlying Pennsylvanian McAlester formation shale bedrock as foundation material. Other structures may be placed on the overlying clayey material. These materials are adequate for properly designed structures and should present no unusual construction problems.[7]

2.7 Seismology

No active or recent faulting has been mapped in the area of the proposed site. The London and Prairie View faults located five and six miles, respectively, from the site are the closest known faults.[8]

The proposed reactor structures will utilize the shale bedrock as a foundation. This rock has good strength properties and will result in no amplification of ground motion from an earthquake.[8]

The area is not seismically active; however, the effects of earthquakes from distant sources may be experienced at the site. The New Madrid earthquake of 1811-1812, the epicenters of which were located about 220 miles north-east of the site, is the type which would be felt at the site. The maximum epicentral intensity for this event was estimated at XII which probably decreased to about VI in the area of the site.

Therefore, because of the above described site conditions and seismic history of the area, the maximum probable intensity of VII is assigned to the site. This value is conservative and corresponds to a design spectrum of 0.10g for plant design with a factor of 0.20g for safe shutdown.[8]

2.8 Dardanelle Lock and Dam

Dardanelle Lock and Dam forms the Dardanelle Reservoir which provides cooling water for the Plant. An investigation was performed to determine if this structure would withstand the "Maximum Earthquake" of 0.2g without losing its functional integrity. This investigation included a stability and structural analysis of the following components:[9]

a. Non-Overflow Section

b. Generator Section

c. Overflow Section

d. Lock Gates and Tainter Gates

e. Lock Walls

f. Earthfill Section

The investigation indicated that the "Maximum Earthquake" could cause some distress and limited damage, but the dam would not lose its functional integrity, and the normal control of pool level would not be interrupted.[9]

An emergency cooling water pond of about 100 acre feet will be dug at the location shown on Figure 2, Exhibit B, to provide cooling water in the unlikely event of destruction of Dardanelle Dam.

2.9 Environmental Radiation Monitoring

Environmental radiation monitoring programs will be conducted at the site with assistance from the State Health Department to establish existing background radiation levels and to detect any changes which may occur. Lake water, air, milk, lake bottom, soil and silt, vegetation and fish samples will be collected and analyzed for gross alpha and gross beta-gamma activity. If any significant amount of activity is found, the samples will be analyzed for specific radionuclides. Sampling points will be located both on-site and off-site.[10] This monitoring program has begun and will continue after operations begin.

3. DESCRIPTION OF RUSSELLVILLE NUCLEAR UNIT

3.1. Introduction

A description of plant features and layout, as well as an evaluation of plant safety are set forth in the Application, as supplemented. The plant description emphasizes the concepts, guidelines and criteria which will govern final design. The station will consist of a reactor building, an auxiliary building (including control room and radwaste area), a turbine structure, a fuel storage building, a shop and storeroom, an administration building, a cooling water pond, a switchyard and various other auxiliary structures and equipment. A plot plan of the Russellville Nuclear Unit, indicating the general station layout, is shown in Figure 2 of Appendix B. Table 1-2 in the Application sets forth a comparison of the design parameters of the proposed Russellville Nuclear Unit with the Duke Power Company's Oconee Units 1, 2, and 3; Florida Power and Light Company's Turkey Point Units 3 and 4; and Florida Power Corporation's Crystal River Plant Unit 3. The following is a summary of the principal features of the plant which are significant with respect to safety considerations:

3.2 Reactor and Primary Coolant System

The reactor for the Russellville Nuclear Unit is of the pressurized water type. It has an initial rating of 2452 MWt, corresponding to a gross electrical output of about 850 MWe.[11] The nominal operating pressure for the reactor is 2185 psig, with an average temperature of 579 F. The reactor coolant system is designed for 2500 psig pressure and 650 F temperature.[12]

The reactor core is approximately 129 inches in diameter, with an active height of 144 inches.[13] It is made up of 177 fuel assemblies, each consisting of a 15 by 15 array of rods enclosed in a square, stainless steel, perforated envelope. The array of rods consists of 208 zircaloy tubes containing uranium dioxide, 16 control rod guide tubes and a center tube available for an in-core instrumentation assembly.[14] There are approximately 201,520 pounds of uranium dioxide in the core.[12]

The thermal and hydraulic design limits of the core are conservative and are consistent with those of other pressurized water reactors currently in operation or under construction.[12, 15]

Core reactivity is controlled by a combination of 69 movable control rod assemblies and a neutron absorber dissolved in the coolant. The control rods are an alloy of silver-indium-cadmium encapsulated in stainless steel. The dissolved neutron absorber is boric acid.[16]

The control rods are used for short-term reactivity control associated with the changes in power level and also with changes in fuel burn-up between periodic adjustments of dissolved boron concentration.[17] The reactor can be shut down by the movable control rods from any power level at any time.[18] Each movable control rod assembly contains 16 control pins, and is actuated by a separate control rod drive mechanism mounted on the top head of the reactor vessel. Upon trip, the 69 control rod assemblies fall into the core by gravity.[19]

Systems are provided so that the concentration of dissolved neutron absorber in the reactor may be adjusted to maintain the reactor shutdown at room temperature and to provide a safe shutdown margin during refueling.[20] The concentration of dissolved absorber is reduced to compensate for long-term reactivity changes, burn-up of fuel and buildup of fission products over the core cycle.

The core is contained within a cylindrical reactor vessel having the dimensions of 14 feet 3 inches inside diameter and 37 feet 4 inches in overall inside height. The vessel has a spherically-dished bottom head with a bolted, removable, spherically-dished top head.[21] The reactor vessel is constructed of carbon steel with all interior surfaces clad with austenitic stainless steel. The reactor vessel is manufactured under close quality control, and several types of nondestructive tests are performed during fabrication. These tests include radiography of welds, ultrasonic testing, magnetic particle examination and dye pentrant testing.[22] During operation, specimens of reactor vessel materials will be placed in the reactor near the inside surface of the reactor vessel. These specimens are subject to irradiation similar to that to which the shell of the reactor vessel is exposed. They will be removed periodically and tested to ascertain the effects of radiation on the reactor vessel material.[23]

Two coolant loops are connected to the reactor vessel by nozzles located near the top of the vessel. Each loop contains one steam generator, two motor-driven coolant pumps and the interconnecting piping. The reactor coolant piping is carbon steel clad on the inside surface with austenitic stainless steel.[24] Reactor coolant is pumped from the reactor through each steam generator and back to the reactor inlet by two 88,000 gpm centrifugal pumps located at the outlet of each steam generator.[25]

The steam generator is a vertical, straight-tube-and-shell heat exchanger which produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes, and steam is generated on the shell side.[26]

The reactor coolant pumps are vertical single-speed, shaft-sealed units having bottom suction and horizontal discharge. Each pump has a separate single-speed top-mounted motor, which is connected to the pump by a shaft coupling.[25]

The pressurizer, a vertical surge tank approximately half-filled with reactor coolant and half-filled with steam, is connected to the reactor coolant system to control system pressure. The operating pressure of the system is maintained by operating electric immersion heaters to increase pressure or by spraying reactor coolant water into the steam within the pressurizer tank to reduce pressure. Self-actuated safety relief valves connected to the pressurizer prevent overpressurization of the reactor coolant system.[27]

3.3 Reactor Building

The reactor building is designed to completely enclose the reactor coolant system and portions of the auxiliary and engineered safeguards systems (see Figure 6, Appendix B). It is a reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system, consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. The building will have three buttresses to which tendons will be anchored instead of six in order to facilitate the arrangement of penetrations and of other equipment within the building. The foundation slab is conventionally re-inforced with high-strength reinforcing steel. The entire structure is lined with welded steel plate, 1/4-inch minimum thickness, to provide vapor tightness. The foundation mat will be bearing on rock and will be approximately 9 feet thick.

The building is designed to sustain safely all internal and external loading conditions which may reasonably be expected to occur during the life of the station or which could result from the postulated design base accident to the reactor's primary coolant system. The tendon system used in the structure is of the unbonded type with a protective compound used to prevent corrosion. Prior to construction, a test will be conducted on the liner plate anchorages to verify certain factors of design analyses.

The reactor building is so designed that, with the engineered safeguards systems provided, any leakage of radioactive materials to the environment will result in doses well within AEC's 10 CFR 100 guidelines for any of the postulated accidents. The integrated leak rate at design pressure will not exceed two-tenths of one percent by volume, within 24 hours.(28)

Prior to operation, the reactor building will be subjected to a structural integrity test and leak rate test. The structural integrity test will be conducted at 115% of design pressure. Periodic leak rate tests will be performed to assure integrity at the reactor building. A tendon surveillance capability will be available to provide assurance that the tendons are free from harmful corrosion and that excessive steel relaxation has not taken place.

3.4 Engineered Safeguards

Engineered safeguards are provided to fulfill the following functions in the unlikely event of an accident:

a. Minimize the release of fission products from the fuel to the reactor building atmosphere

b. Ensure reactor building integrity and reduce the driving force for building leakage

c. Remove fission products from the reactor building atmosphere.

The engineered safeguards systems can be grouped into an emergency core cooling system, reactor building cooling systems and fission products control systems.

The emergency core cooling systems contain both passive flooding and pumping equipment. The passive flooding equipment consists of two pressurized core flooding tanks which automatically discharge borated water into the reactor vessel in the event the reactor system pressure drops below 600 psi. The pumping equipment consists of two completely independent sub-systems. Each sub-system contains both a high pressure and a low pressure injection pump. Either sub-system, in conjunction with the core flooding tanks, is capable of protecting the core for any size leak up to and including the double-ended rupture of the largest reactor coolant pipe. Either sub-system can supply coolant directly from the borated water storage tank or by recirculation from the reactor building sump through heat exchangers which cool it before it is returned to cool the core.[30]

The reactor building cooling system, which is made up of two separate and independent heat removal systems, limits the pressure in the reactor building following a loss-of-coolant accident. One system contains three separate fan and cooler units. The other system contains redundant spray headers which spray low temperature borated water into the reactor building to cool it. Each of these systems independently has the heat removal capability to maintain the reactor building pressure below its design pressure.[31]

Control of fission products following a loss-of-coolant accident is provided by the reactor building itself and by a second separate engineered safety feature for limiting release of fission products from the reactor building. The second means for fission product control is the iodine removal spray system which utilizes sodium thiosulphate mixed in the reactor building spray water to absorb the iodine released from the reactor during an accident and renders it unavailable for leakage from the reactor building. The reactor building and the iodine removal chemical spray system will limit radiation doses at the exclusion radius and low population zone boundary to values within the 10 CFR 100 guideline values.[32] In addition, room has been provided for charcoal filters if it is subsequently determined that they are needed.

3.5 Instrumentation and Control

A complete and dependable network of instrumentation and controls will be provided to ensure safe operations of Russellville Nuclear Unit. The reactor protective system monitors parameters related to safe operation and shuts down the reactor if an operating limit is reached.[33] This will be accomplished by interrupting power to the control rod drive clutches and allowing the control rods to drop into the reactor core.[34] Alarms[35] are provided to alert the operator to abnormal operating conditions, and interlocks[36] are provided to prevent abnormal operations which could lead to potentially unsafe conditions.

The nuclear instrumentation system monitors reactor power from start-up level through 125 percent of full power operation. There are separate overlapping instrumentation channels for the start-up power range, the intermediate approach to power range, and the power operation range.[37] A control system automatically monitors reactor system conditions and the load requirements on the turbine-generator unit, and adjusts reactor power, steam generator feedwater flow and the turbine throttle for safe, efficient operation.[38]

The engineered safeguards protective system monitors plant conditions and automatically initiates operation of the engineered safeguards systems, if required.[39]

Following proven power station design philosophy, all control stations, switches, controllers and indicators necessary to start-up, operate and shutdown the nuclear unit will be placed in the centrally located control room. There will be sufficient information display and alarm monitoring to ensure safe and reliable operation under normal and accident conditions. Design is such as to permit shutting down the reactor from outside the control room.

The report of the Advisory Committee on Reactor Safeguards for the Russellville Nuclear Unit indicated that the instrumentation design should be reviewed for common failure modes, and that it should be shown that the interconnection of control and safety circuitry will not significantly affect safety considering the possibility of systematic component failures. During the detailed design of the instrumentation systems their immunity to common failure modes will be evaluated. The possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single failure criterion, will be taken into account in the evaluation. The instrumentation signals sent to control and safety circuits from common transmitters are made fully independent by the use of isolation amplifiers. The effectiveness of these devices has been demonstrated by analysis and by actual test of prototype equipment as described in Supplement 3 to the PSAR, Question 6.4.

3.6 Electrical Systems

The design of the electrical systems for the Russellville Nuclear Unit is based on providing the required electrical equipment and power sources to ensure safe, reliable operation and safe, orderly shutdown of the unit under any normal or emergency conditions. Four sources of power, each possessing various degrees of redundancy, are available to ensure a supply of electrical energy to the station safety systems under any accident conditions, including the loss-of-coolant accident, as outlined below:

a. Two 500-kv transmission lines can supply power for the station auxiliary load through Start-Up Transformer No. 1 connected to the 22 kv tertiary of the 500 kv-161 kv bus tie autotransformer.

b. Start-Up Transformer No. 2 will provide an alternate off-site power source from the 161 kv ring bus, supplied by two 161 kv transmission lines.

c. The main generator will continue to supply the station auxiliary load upon abrupt separation from the 500 kv and 161 kv systems.

d. Upon loss of all sources of power described in (a), (b) and (c) above, power will be supplied from the two automatic, fast start-up diesel engine generators. These are sized so that either can carry the required engineered safeguards load.

The unit will generate electric power at 22 kv, which will be fed through an isolated phase bus to the unit main transformer where it will be stepped up to 500 kv transmission voltage and delivered to the switchyard. The 500 kv switchyard, in turn, is linked to the existing 500 kv transmission network by two 500 kv circuits, and is tied to the 161 kv system by a bus tie autotransformer.

3.7 Auxiliary Systems

Auxiliary systems are provided to supply reactor coolant makeup and seal water, to cool the reactor during shutdown, to cool components, to ventilate station spaces, to handle fuel and to cool spent fuel.

Reactor coolant makeup and seal water is supplied by the makeup and purification system. This system, which also serves the engineered safeguards function of providing high pressure emergency core coolant, maintains the proper coolant inventory in the primary system, maintains the seal water flow, adjusts the concentration of dissolved neutron absorber in the reactor coolant and maintains proper water chemistry.[40]

The decay heat removal system cools the reactor when the reactor system is depressurized for maintenance or refueling.[41] This same system serves the engineered safeguards functions of providing low pressure emergency core coolant and of recirculating borated water to cool the core in the unlikely event of a loss-of-coolant accident.

The chemical addition and sampling system adds boric acid to the reactor coolant system for reactivity control, potassium hydroxide for pH control, and hydrogen and hydrazine for oxygen control. This system is also used to take reactor coolant and steam generator water samples.[42]

The cooling water systems maintain temperatures throughout the equipment and structures of the station.[43]

Appropriate normal ventilation systems are provided in the station.[44]

A fuel handling system[45] provides the means for safe, reliable handling of fuel from the time it enters the station as new fuel until it is shipped from the station as used fuel. Irradiated fuel is handled under water at all times until after it is placed into a shipping cask. The water provides a radiation shield as well as a reliable source of cooling for the irradiated fuel assemblies. A spent fuel cooling system maintains the temperature and purity of the spent fuel storage pool water within acceptable limits.[46]

3.8 Steam and Power Conversion System

The steam and power conversion system is designed to remove the heat energy generated in the reactor core by producing steam in the two steam generators. This heat energy is converted to electrical energy by the turbine-generator. A cooling water system utilizing Dardanelle Reservoir water will be used to dissipate the thermal energy rejected by the turbine condenser. This cycle, including the necessary equipment to achieve safe and reliable operation, is similar in concept and design to turbine-generator cycles in successful use for many years.

3.9 Radioactivity Control Systems

Radioactive gaseous, liquid and solid wastes in the station are handled by the waste disposal systems. These systems contain the equipment necessary to safely collect, process and prepare for disposal the radioactive wastes which result from reactor operation. These systems are designed to minimize the release of radioactive material from the station to the environment and will maintain releases below the limits of 10 CFR 20.

A process radiation monitoring system monitors effluent released to the environment and provides an early warning of possible equipment malfunction or potential radiological hazard. The radiation monitoring system includes a combination of continuous-automatic-monitoring and periodic sampling.

Shielding throughout the station ensures that radiation doses to the general public and to operating personnel during normal operation are well within the limits of 10 CFR 20.

4. SAFETY ANALYSES

Potential malfunctions or equipment failures have been analyzed to provide a safety evaluation of the Russellville Nuclear Unit. This evaluation demonstrates that the public will not be exposed to radiation in excess of the limits established in the AEC's regulation for siting requirements, 10 CFR 100, even in the very unlikely event that one of the accidents postulated in the Application should occur.[47]

Two categories of malfunctions or equipment failures have been analyzed: those in which the core and coolant boundaries are protected, and those in which one of these boundaries is not effective and standby safeguards are required. The core and coolant boundary protection analysis shows that in the event any of the postulated malfunctions were to occur, the normal protection systems operate to maintain the integrity of the core and of the coolant boundary.[48] The standby safeguards analysis demonstrates the capability of the engineered safeguards systems to assure protection of the public for postulated malfunctions in which the normal protective systems may not maintain the integrity of the core and coolant boundary.[49] These analyses show that for all credible malfunctions the radiation exposure to the general public is well below the limits prescribed in 10 CFR 100.

Of the postulated equipment failures, a loss-of-coolant accident is the most severe. Emergency core cooling equipment is provided to prevent clad and fuel damage that would interfere with continued core cooling for reactor coolant system failures up to and including the complete severance of the largest reactor coolant pipe. The core cooling system ensures that the core will remain in place and intact.[50] The reactor building spray or emergency cooling units maintain the integrity of the reactor building.[51] The iodine removal sprays in conjunction with the reactor building assure that the public is protected from radiation and radioactive material.[52] Emergency electrical power is available on-site to ensure operation of these systems even if all external sources of electric power to the plant are assumed to be unavailable at the time of the accident.[53]

Results of the safety analyses show that, even in the unlikely event of a loss-of-coolant accident, no core melting will occur.[52] However, in order to demonstrate that the operation of a nuclear power station at the proposed site does not present any undue hazard to the general public, a hypothetical accident has been analyzed involving release of 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids in the fission product inventory. The analysis evaluated both the direct radiation exposure and the potential total dose to the thyroid from the inhalation of fission products which are assumed to leak from the reactor building. The low leakage rate of the reactor building and the iodine removal spray system reduce the potential radiation dose to the thyroid to below the 10 CFR 100 guidelines even in the event of such a hypothetical occurrence.[54]

5. TESTS, INSPECTIONS, AND QUALITY CONTROL

Pressure containing components of the reactor coolant system will be designed, fabricated, inspected and tested in accordance with Section III, Nuclear Vessels, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The piping will meet the applicable provisions of Power Piping USA Standards and associated nuclear code cases. Non-destructive testing, including radiography, ultrasonic, magnetic particle, and liquid penetration examinations will be performed during fabrication of the nuclear vessels.

Auxiliary systems and equipment will be designed, fabricated and tested to the appropriate provisions of recognized codes and standards of organizations such as the American Society of Mechanical Engineers, American Society for Testing Materials, USA Standards Institute and Institute of Electrical and Electronics Engineers.

A comprehensive field testing program will be conducted to ensure that equipment and systems perform in accordance with design criteria.

The reactor building will be designed and built in accordance with applicable portions of the Building Code Requirements for Reinforced Concrete, ACI 318-63: Specification for Structural Concrete for Buildings, ACI 301-66; AISC Manual of Steel Construction; ASME Boiler and Pressure Vessel Code, Sections III, VIII, and IX. Materials and workmanship will be inspected to ensure compliance with appropriate codes, specifications, and standards. Materials to be inspected and tested include concrete, liner plate, prestressing system materials, hatches, penetrations, structural and reinforcing steel.

The reactor building will be structurally tested at 115 percent of design pressure by pneumatic test. In addition, it will be leak tested to ensure compliance with a maximum allowable gross leak rate of 0.2 percent by volume per 24 hours at the design pressure. Provisions have been included for in-service pressure testing of equipment and personnel hatches and other penetrations.

Consideration has been given to the inspectability of the reactor coolant system in the design and arrangement of components. Access for inspection of the reactor coolant system includes access for visual examination by direct or remote means.

The Applicant's contractors and major equipment suppliers will provide required quality control functions, procedures and techniques to assure manufacture and construction in accord with the plant design and specifications furnished to the Applicant by its architect/engineers, Bechtel Corporation. B&W has an extensive quality assurance program organized and functioning with respect to both equipment of its own manufacture and equipment purchased by it from other vendors. The general contractor has not been finally selected, but this contractor will be required to provide a satisfactory quality assurance program.

In addition, Bechtel Corporation, in its construction management function, will provide a complete quality assurance program covering tests and inspection both in suppliers' shops and on the site of construction and erection.

Applicant has a quality assurance organization which is separate and independent from its vendors, contractors and construction manager. Through its own employees and independent consultants it will monitor the adequacy of quality control procedures followed in the design, fabrication, construction, erection, transportation and testing of reactor components, equipment and structures.

6. RESEARCH AND DEVELOPMENT PROGRAMS

The nuclear steam supply system for Russellville is similar in concept to several projects already in operation, under construction or recently licensed by the Atomic Energy Commission. The preliminary design is based on technical data which has been developed in the nuclear industry and on data developed by B&W which is specifically related to the Russellville Nuclear Unit design. To complete the final detail design of some components additional technical information will be obtained.

The following are the areas of the plant design in which additional technical data will be developed to finalize design details.

a. Once-Through Steam Generator

The design of the once-through steam generator is based on experimental work on boiling heat transfer and data obtained by B&W in full length model tests of the unit. The testing of a prototype unit has been completed but is not yet documented. It included performance, mechanical, vibration and blowdown tests, and control system development. The results have confirmed the analytical predictions of performance, and sufficient data on the performance and structural design has been obtained from operation of the test models to finalize the design of the steam generators.[55]

b. <u>Control Rod Drive Unit</u>

The design of the control rod drive mechanisms is based on a principle which has been used in operating reactors and which has been extensively tested by B&W. Test programs have included full scale prototype testing under no-flow conditions, full scale prototype testing at operating conditions, including flow, and components testing. Testing of a prototype mechanism was carried out for a full-life cycle of strokes and trips, and major design parameters were confirmed. Life cycle testing has been repeated using a miter gear of improved material and showed satisfactory performance. Data from these test programs are being incorporated into the final design of the control rod, its guide structure and the control rod drive mechanism.[56]

c. <u>In-Core Neutron Detectors</u>

The performance and longevity of the self-powered detectors are being demonstrated by detectors installed in the Babcock and Wilcox Test Reactor and in the Big Rock Point Nuclear Power Plant.[57] The tests have demonstrated that the detectors perform successfully. Tests are being continued in order to demonstrate detector longevity. At the present time, the Big Rock Point detectors have accumulated operational experience equivalent to approximately three and one-half years of full power operation in the Russellville Nuclear Unit reactor.

d. <u>Core Thermal and Hydraulic Design</u>

The PSAR as originally submitted contained, in Section 3, an evaluation of the core thermal capability in which the heat transfer limits were predicted based on a correlation of experimental DNB (Departure from Nuclear Boiling) data developed by The Babcock & Wilcox Company. In order to completely substantiate the B&W correlation additional research and development data is necessary. These requirements are described in the PSAR.[58]

Subsequent to submittal of the original PSAR, core thermal performance was also evaluated using the W-3 correlation for predicting DNB. This correlation is available in the literature and has been used and found acceptable in establishing thermal design limits for other large pressurized water reactors. The thermal evaluation using the W-3 correlation is also presented in the PSAR and its supplements. With the use of this correlation, vessel model flow tests are necessary to substantiate operation of the plant within acceptable thermal limits. Flow testing which demonstrated acceptable flow distribution for the rated power level without internal vent valves in the model has been completed. Flow testing with internal vent valves installed and with open internal vent valves must still be performed.

e. <u>Emergency Core Cooling and Internal Vent Valves</u>

Analytical evaluation of the effects of blowdown forces on the internals and of the performance of the internal vent valves installed in the core support shield to insure adequate covering of the core by emergency coolant is in progress. A prototype of these valves is being tested to demonstrate their operating characteristics.[59]

f. Fuel Failure

A study, including testing, is underway to assure that there are no failure mechanisms which might interfere with the ability of the emergency core cooling systems to accomplish their objectives. The results of the work to date demonstrate the ability of the design to accommodate potential fuel failure mechanisms. This work will be continued to assure that fuel rod failures will not significantly affect the ability of the emergency core cooling system to prevent clad melting.[60]

g. Xenon Oscillations

The possibility of the occurrence of xenon oscillations throughout core life is being evaluated. If it is determined that such oscillations may occur, appropriate design changes to eliminate or control the oscillations will be incorporated.[61] The design of a means to eliminate or control such oscillations is being carried out in parallel with the studies of the possibility of such oscillations.

h. <u>Chemical Spray Additive</u>

One of the radiological protection systems of the Russellville Nuclear Unit provides for spraying chemical solutions into the reactor building to remove iodine under accident conditions. Testing to demonstrate the ability of the chemical sprays to remove and retain iodine effectively, and to demonstrate solution stability and chemical compatibility with plant materials is in progress.[62]

7. TECHNICAL QUALIFICATIONS

7.1 Arkansas Power & Light Company

Applicant has over 45 years experience in the design, construction and operation of electric generating plants. Personnel of the Engineering Department of the Company have supervised and made final decisions on the design and construction of its generating plants. It has been the practice of the Company, however, to retain independent engineers to design and manage the construction of its generating plants under the supervision of the Company's engineers. The Production Department, which is a part of the Engineering Department of the Company, operates all of the generating plants with its personnel.

On October 1, 1968, Applicant operated five steam electric generating plants containing a total of 12 units with a net capability of 1,659,000 kilowatts, two hydro-electric stations with a capability of 69,000 kilowatts and diesel generating units with a total capability of 6,000 kilowatts, for a total net electric generating capability of 1,734,000 kilowatts. At the present time the Company is constructing one additional generating unit, a

530,000 kilowatt gas-fired unit, which is scheduled to be completed in 1969.

Applicant was one of the founders in 1957 of Southwest Atomic Energy Associates which was created to conduct research in nuclear fuels. In addition to other projects, SAEA is now one of the participants in the Southwest Experimental Fast Oxide Reactor Facility near Fayetteville, Arkansas, which is expected to begin operations in December 1968. Various officers and employees of Applicant have actively participated in the activities of SAEA and SEFOR since 1957 as trustees, officers, committee members and observers. Applicant has also been a member of and a contributor to High Temperature Reactor Development Associates, Inc. and has participated in the sponsoring of the HTRDA operation at Peach Bottom, Pennsylvania. Applicant is also a contributor to and participant in Southern Inter-State Nuclear Board and the Atomic Industrial Forum.

Applicant recognizes the importance of the early training of sufficient personnel to assure adequate operating manpower, which is the subject of a comment in the ACRS letter. Applicant will initially train enough employees so that there not only will be enough trained employees for regular work on each shift, but also there will be adequately trained personnel to substitute during illness, vacations and other absences. Applicant's training program for operators is described in full in PSAR, Volume II, Appendix 1A, Section 1.7. This program will include 600 hours of classroom work in nuclear engineering and reactor theory, three to five months of training in operations at an existing plant or on a simulator, about two months instruction on the design characteristics of reactor systems furnished by The Babcock & Wilcox Company and approximately seven months of on-the-job training at the Russellville Nuclear Unit.

7.2 Bechtel Corporation

Bechtel Corporation has been retained by AP&L as Architect/Engineer and Manager of Construction for the Russellville project.

Working closely with AP&L, Bechtel is responsible for project studies and conceptual design, specification of material and services, project detailed design, construction management, quality control programs and assistance in plant testing and start-up.

Bechtel Corporation has been continuously engaged in construction or engineering activities since 1898. For the last 20 years, Bechtel has been active in the fields of petroleum, power generation and distribution, harbor development, mining and metallurgy, and chemical and industrial processing.

Since the close of World War II, Bechtel has been responsible for the design of over 165 power generating units, representing more than 38 million kilowatts of new generating capacity, which includes units of the largest and most modern types. Of this number, more than 11 million KWe is produced by 20 nuclear-fueled units.

For over 18 years, Bechtel has been engaged in the study, design and construction of nuclear installations. Their experience includes design or construction, or both, of such facilities as accelerators, nuclear research laboratories hot cells, experimental reactors and nuclear fuel processing plants, as well as nuclear power plants. A summary of experience is listed in the Application.

7.3 Babcock and Wilcox Company

B&W's participation in the development of nuclear power dates from the Manhattan Project. B&W's broad nuclear activities include applied research to develop fundamental data; design and manufacture of nuclear systems, cores, and components; and design, manufacture, and erection of complete nuclear steam generating systems. Through the B&W Company's several divisions, a wide range of equipment for nuclear application is designed and manufactured. The B&W Company's major nuclear contracts, in addition to manufacture of a substantial percentage of components for the nuclear Navy, have included Indian Point No. 1; NS Savannah; Advanced Test Reactor; Oconee Nuclear Station Units 1, 2 and 3; Three Mile Island Nuclear Station; Crystal River Plant Unit 3; and four other units in various stages of licensing in addition to the Russellville Nuclear Unit.

8. <u>COMMON DEFENSE AND SECURITY</u>

There is no indication that construction and operation of the Russellville Nuclear Unit will in any way be inimical to the common defense and security of the United States.

As stated in the Application, AP&L is a private utility with statutory authority for the production, transmission and sale of electric energy. All of the directors and principal officers are citizens of the United States, and AP&L is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

The Application contains no restricted or other defense information and Applicant has agreed that it will not permit any individual to have access to Restricted Data until the Civil Service Commission shall have made an investigation and report to the Atomic Energy Commission on the character, associations and loyalty of such individual, and the Atomic Energy Commission shall have determined that permitting such persons to have access to Restricted Data will not endanger the common defense and security.

As a licensee, Applicant will be subject to regulations of the Atomic Energy Commission relating to the transfer of and accountability for special nuclear material in its possession. Recent amendments to the AEC Rules and Regulations (10 CFR 50.60) under which the AEC will discontinue allocating quantities of special nuclear material to reactor licensees evidence that such material is no longer scarce. Moreover, in the event of a state of war or national emergency, the AEC may order the recapture of special nuclear material, as well as the operation of any licensed facility. (10 CFR 50.103)

9. CONCLUSION

On the basis of the foregoing and the Application, the Applicant respectfully submits that:

a. Arkansas Power & Light Company's Application, as supplemented, describes the proposed design of the Russellville Nuclear Unit, including the principal architectural and engineering criteria for the design, and identifies the major features or components incorporated in the plant for the protection of the health and safety of the public.

b. The Application, as amended, identifies the technical and design information necessary to complete the final safety analysis. Such information can reasonably be left for later consideration and will be supplied in the final safety analysis report.

c. Safety features which require further research and development, and the research and development programs to be carried out, are identified in Section 1.5 of the PSAR. The research and development program is reasonably designed to resolve any questions associated with such features at or before the latest date stated in the Application for completion of construction of the facility.

d. Taking into consideration the characteristics of the site and environs and the proposed design of the Russellville Nuclear Unit, such facility can be constructed and operated within the limitations established by 10 CFR 20, within the site criteria set forth in 10 CFR 100, and without undue risk to the health and safety of the public.

e. The Applicant is technically qualified to design and construct the proposed facility.

f. The issuance of a construction permit for the Russellville Nuclear Unit will not be inimical to the common defense and security of the United States or to the health and safety of the public.

APPENDIX A

LIST OF REFERENCES

APPENDIX

LIST OF REFERENCES

1. PSAR, Volume I, Section 2.2.2
2. PSAR, Supplement 3, Question 2.3
3. PSAR, Volume I, Section 2.2.5
4. PSAR, Volume II, Appendix 2A
5. PSAR, Volume I, Section 2.4
6. PSAR, Volume I, Section 2.5
7. PSAR, Volume I, Section 2.6
8. PSAR, Volume I, Section 2.7
9. PSAR, Volume I, Section 2.8
10. PSAR, Supplement 3, Question 2.9
11. PSAR, Volume I, Section 1.2.2
12. PSAR, Volume I, Table 1-2
13. PSAR, Volume I, Table 3-2
14. PSAR, Volume I, Table 3-1
15. PSAR, Volume I, Section 3.2.3
16. PSAR, Volume I, Section 3.2.2.1.2 and Section 7.2.2.1
17. PSAR, Volume I, Table 3-6 and Figure 3-1
18. PSAR, Volume I, Section 3.2.2.1.3

19. PSAR, Volume I, Section 3.2.4.3.2

20. PSAR, Volume I, Section 3.2.2.1.3

21. PSAR, Volume I, Section 4.2.2.1

22. PSAR, Volume I, Section 4.1.4.4

23. PSAR, Volume I, Section 4.4.3

24. PSAR, Volume I, Section 4.2.5

25. PSAR, Volume I, Section 4.2.2.4

26. PSAR, Volume I, Section 4.2.2.3

27. PSAR, Volume I, Section 4.2.2.2

28. PSAR, Volume I, Section 5.9.1.2

29. PSAR, Volume I, Section 6

30. PSAR, Volume I, Section 6.1

31. PSAR, Volume I, Section 6.2

32. PSAR, Volume II, Section 14.2.2

33. PSAR, Volume I, Section 7.1

34. PSAR, Volume I, Section 3.2.4.3

35. PSAR, Volume I, Section 7.4.3

36. PSAR, Volume I, Section 7.2.3.2

37. PSAR, Volume I, Section 7.3.1

38. PSAR, Volume I, Section 7.2.2.2

39. PSAR, Volume I, Section 7.1.2.2 and Section 7.1.2.3

40. PSAR, Volume I, Section 9.1

41. PSAR, Volume I, Section 9.5

42. PSAR, Volume I, Section 9.2

43. PSAR, Volume I, Section 9.3

44. PSAR, Volume I, Section 9.7

45. PSAR, Volume I, Section 9.6

46. PSAR, Volume I, Section 9.4

47. PSAR, Volume II, Section 14

48. PSAR, Volume II, Section 14.1

49. PSAR, Volume II, Section 14.2

50. PSAR, Volume I, Section 6.1

51. PSAR, Volume I, Section 6.2

52. PSAR, Volume II, Section 14.2.2.3

53. PSAR, Volume I, Section 8.2.3

54. PSAR, Volume I, Section 14.2.2.4

55. PSAR, Volume I, Section 1.2.7 and Section 1.5.7 and Supplement 3 Question 1.3a

56. PSAR, Volume I, Section 1.5.6 and Supplement 3, Question 1.3b

57. PSAR, Volume I, Section 1.5.8 and Supplement 3, Question 1.3c

58. PSAR, Volume I, Section 1.5.2 and Supplement 3, Question 1.3d

59. PSAR, Volume I, Section 1.5.5 and Supplement 3, Question 1.3e

60. PSAR, Volume I, Section 1.5.3 and Supplement 3, Question 1.3(h)

61. PSAR, Volume I, Section 1.5.1 and Supplement 3, Question 1.3(i)

62. PSAR, Volume I, Section 1.5.10 and Supplement 3, Question 1.3g

APPENDIX B

FIGURES

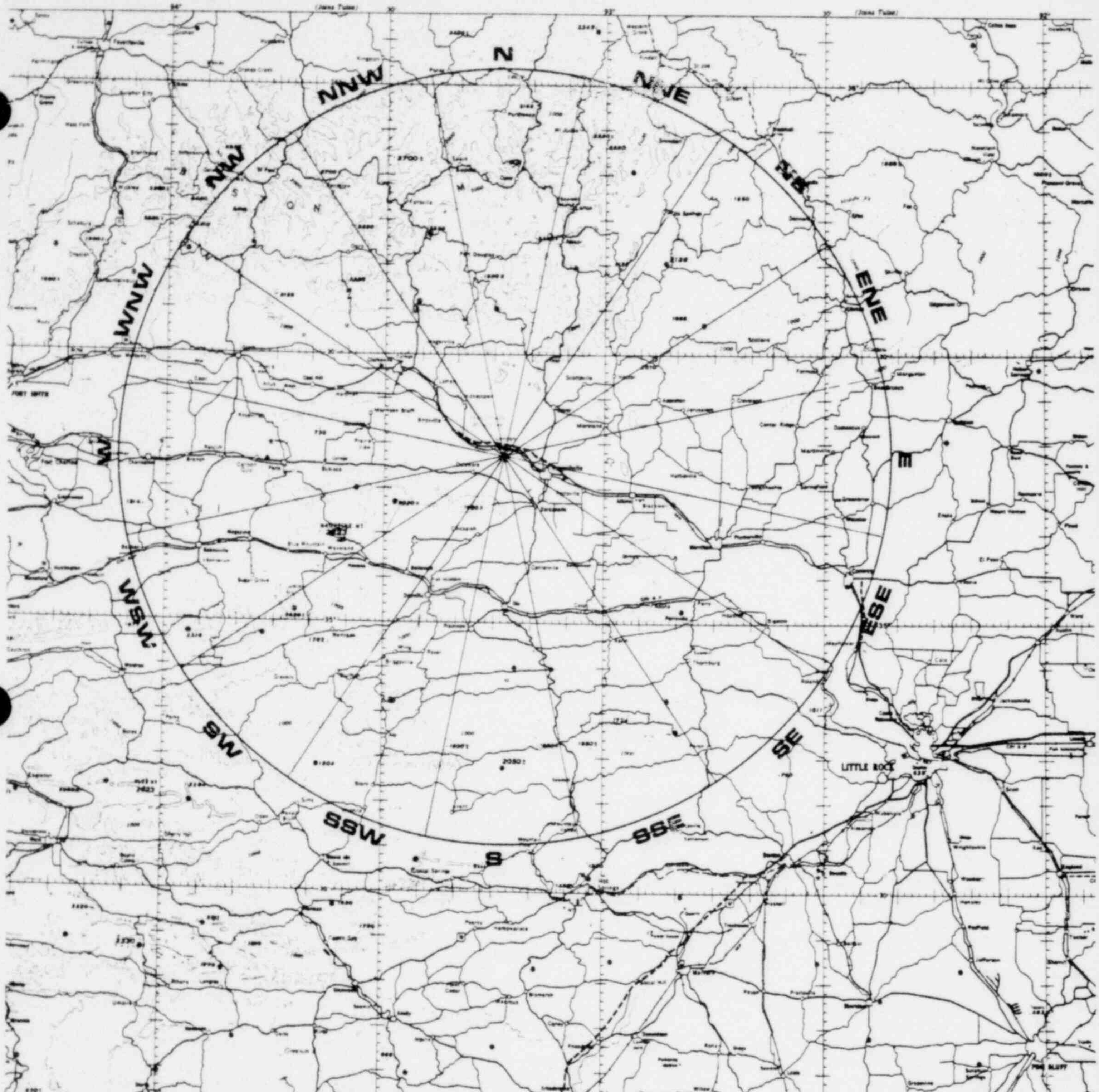
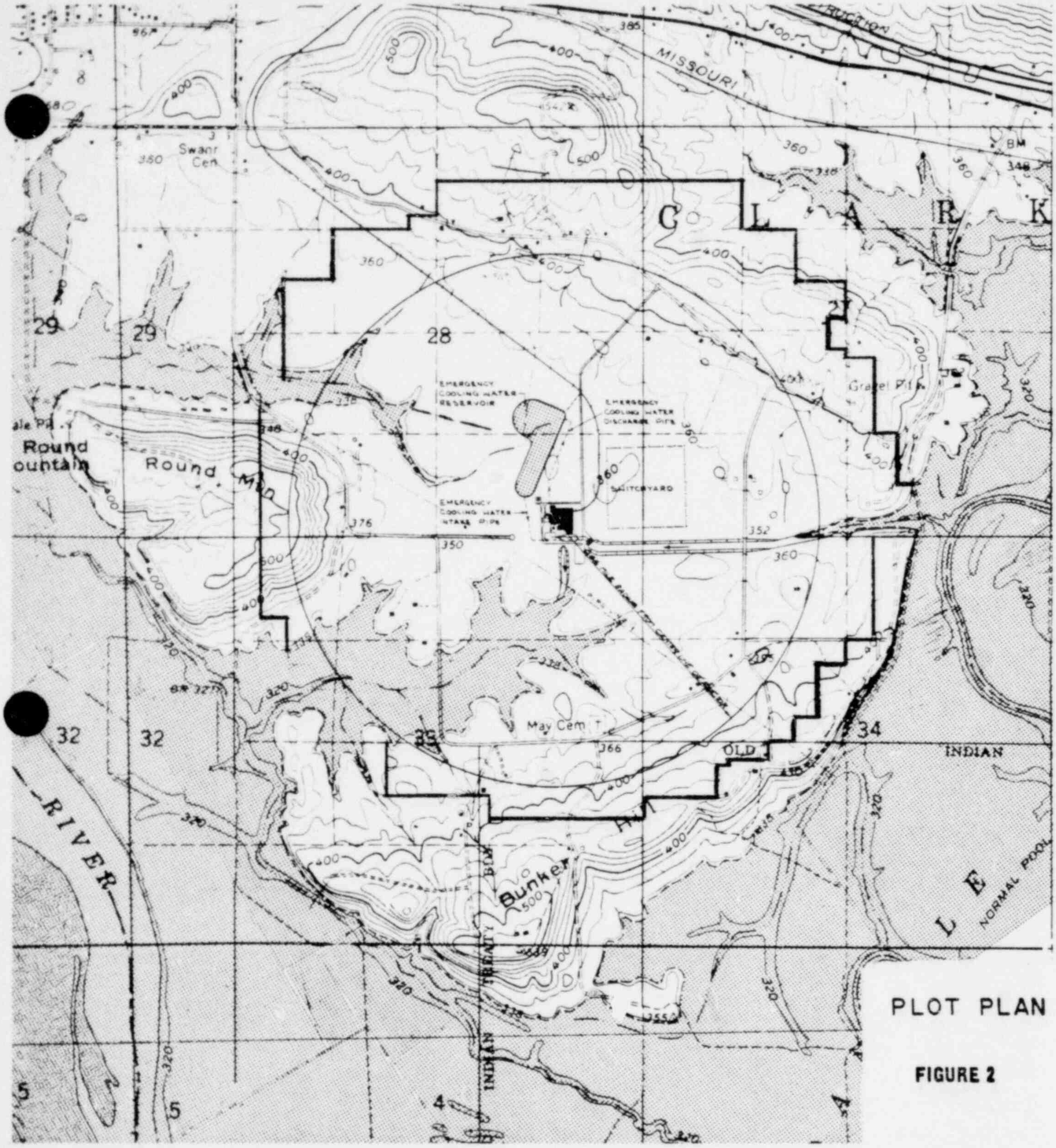


Figure 1

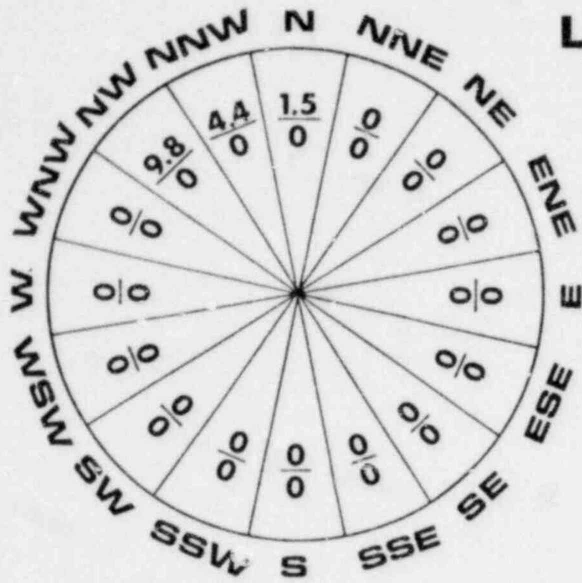
GENERAL AREA MAP



PLOT PLAN

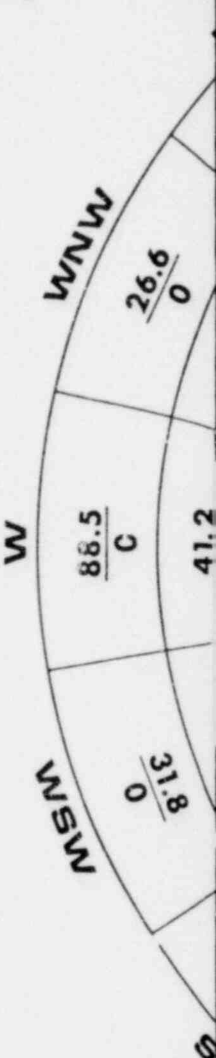
FIGURE 2

0–5 MILES



LAND USE IN SQUARE M
WITHIN A 50 MILE RAD

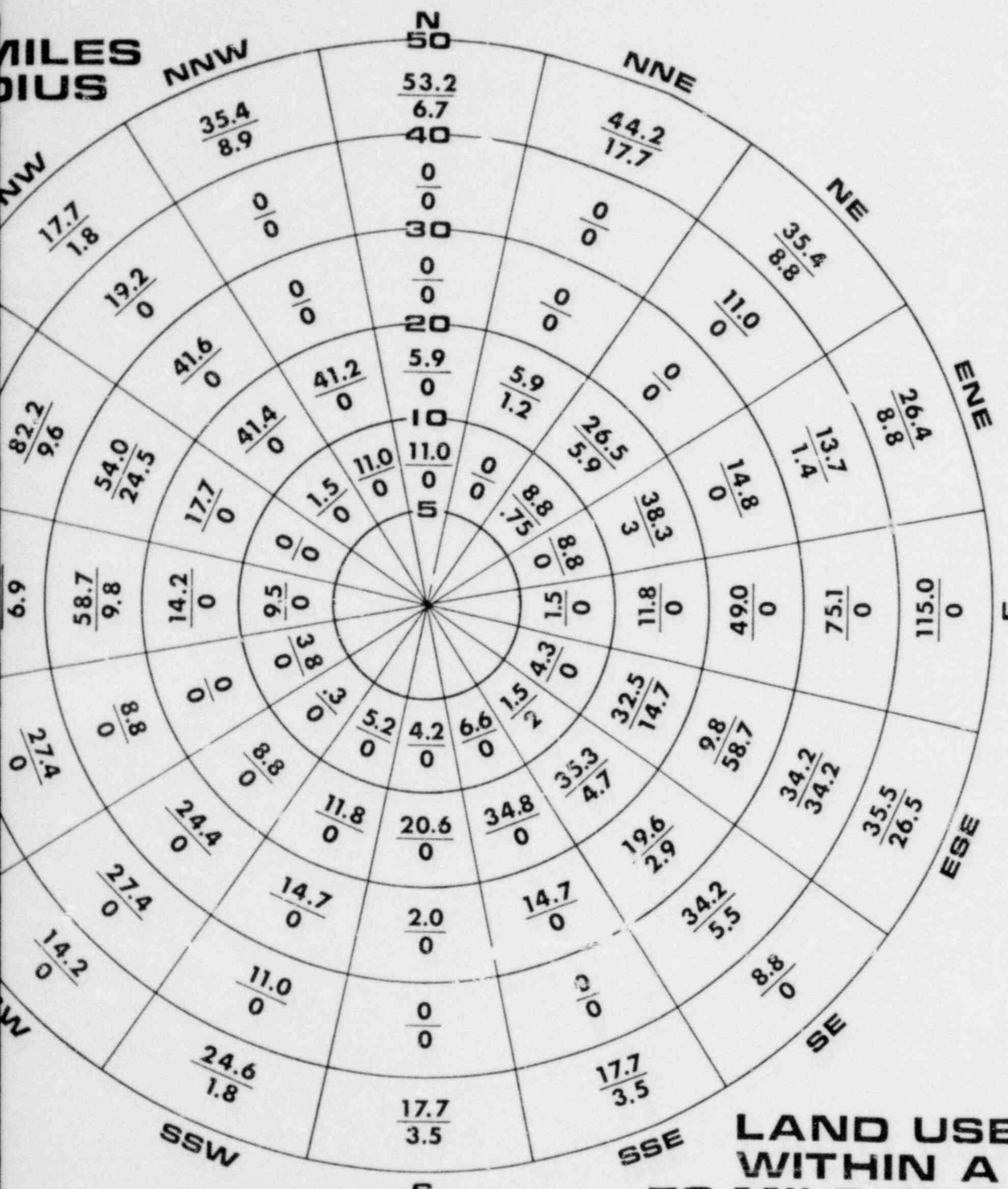
[PASTURED P / CULTIVATED C]



CUMULATIVE TOTALS BY SECTORS

SECTOR	RADIUS IN MILES									
	10		20		30		40		50	
	P	C	P	C	P	C	P	C	P	C
	SQ. MI.		SQ. MI.		SQ. MI.		SQ. MI.		SQ. MI.	
N	12.5	0	18.4	0	18.4	0	18.4	0	71.6	6.7
NNW	15.4	0	56.6	0	55.6	0	56.6	0	92.0	8.9
NW	2.5	0	43.9	0	85.5	0	104.7	0	122.4	1.8
WNW	0	0	17.7	0	71.7	24.5	153.9	34.1	180.5	34.1
W	9.5	0	23.7	0	82.4	9.8	123.6	16.7	212.1	16.7
WSW	3.8	0	3.8	0	12.6	0	40.0	0	71.8	0
SW	0.3	0	9.1	0	33.5	0	60.9	0	75.1	0
SSW	5.2	0	17.0	0	31.7	0	42.7	0	67.3	1.8
S	4.3	0	24.9	0	26.9	0	26.9	0	44.6	3.5
SSE	6.6	0	41.4	0	56.1	0	56.1	0	73.8	3.5
SE	1.5	.2	36.8	4.9	56.4	7.8	90.6	13.3	99.4	13.3
ESE	4.3	0	36.8	14.7	46.6	73.4	80.8	107.6	116.3	142.8
E	1.5	0	13.3	0	62.3	0	137.4	0	252.4	0
ENE	8.8	0	47.1	3	61.9	3	75.6	4.4	102	13.2
NE	8.8	.75	35.3	6.7	35.3	6.7	46.3	6.7	81.7	15.5
NNE	0	0	5.9	1.2	5.9	1.2	5.9	1.2	50.1	18.9

5–50 MILES



LAND USE WITHIN A 50 MILE RADIUS

Figure 3

DAIRY ANIMALS WITHIN A 50-MILE RADIUS

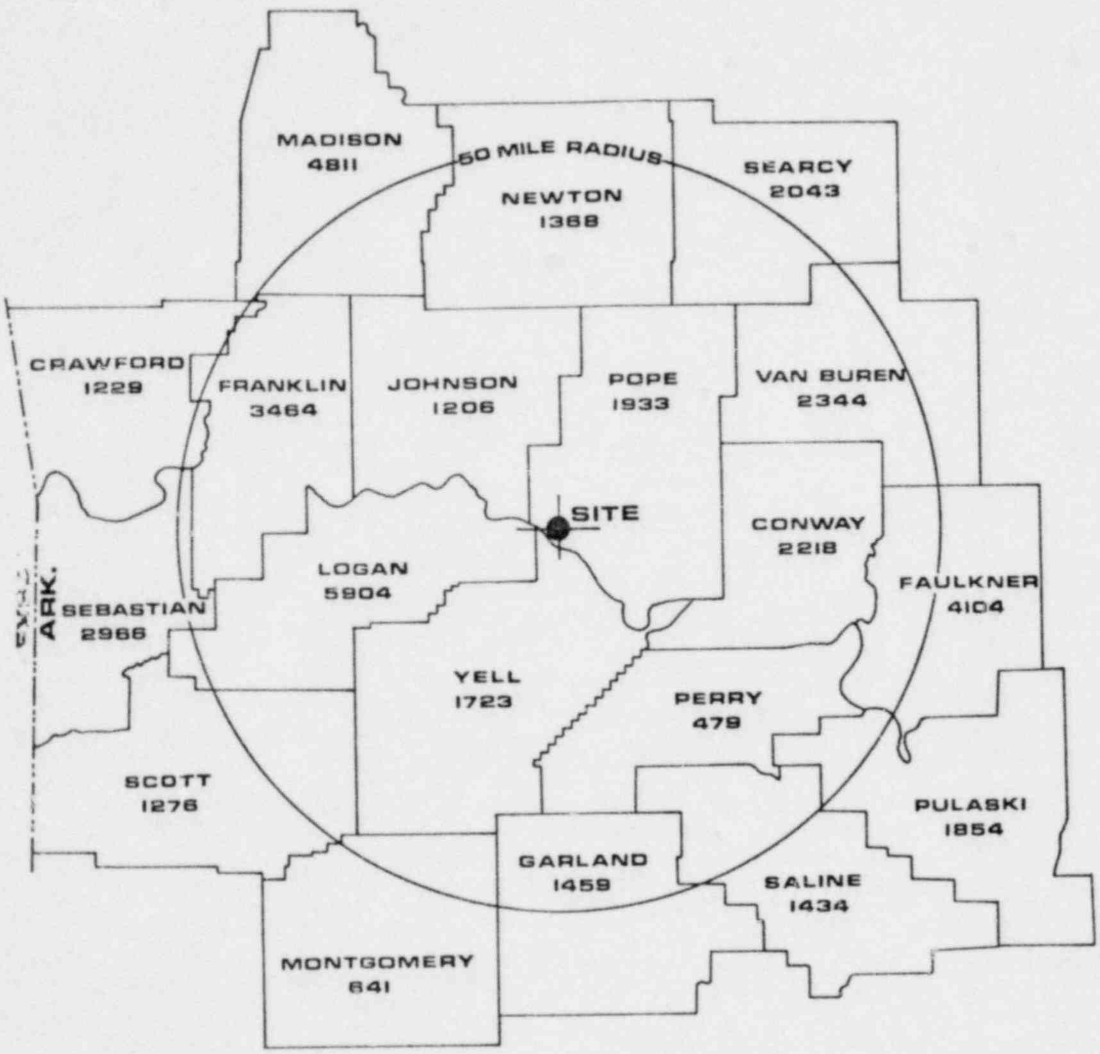
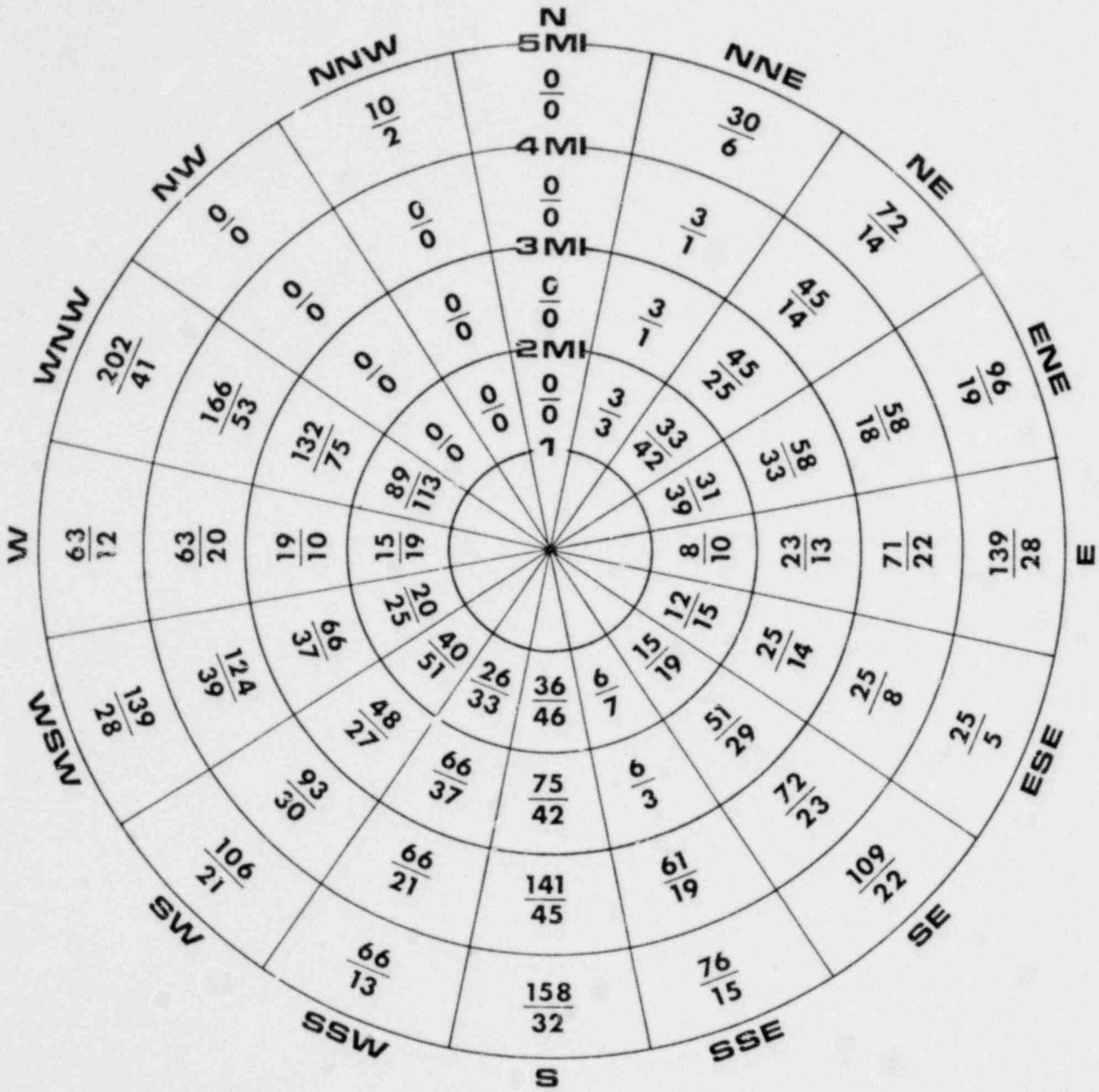


Figure 4

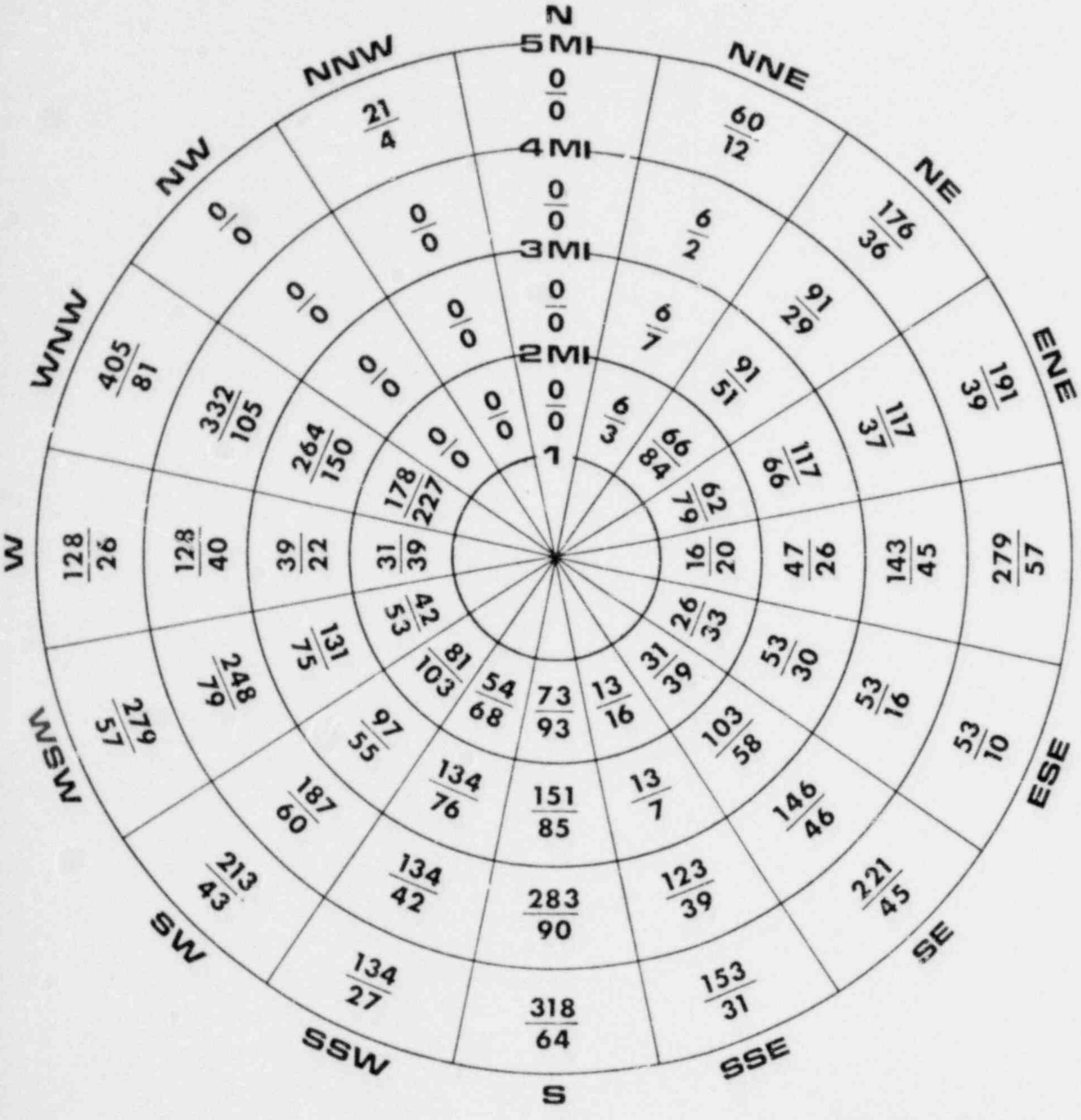
1967
0–5 MILES



TOTAL POPUL.
PERSONS/SQ.

TOTAL POPULATION IS CUMULATIVE FROM THE CENTER.

2012
0–5 MILES



ATION
MILE

ESTIMATED TRANSIENT POPULATION (1967-2012)

Figure 5

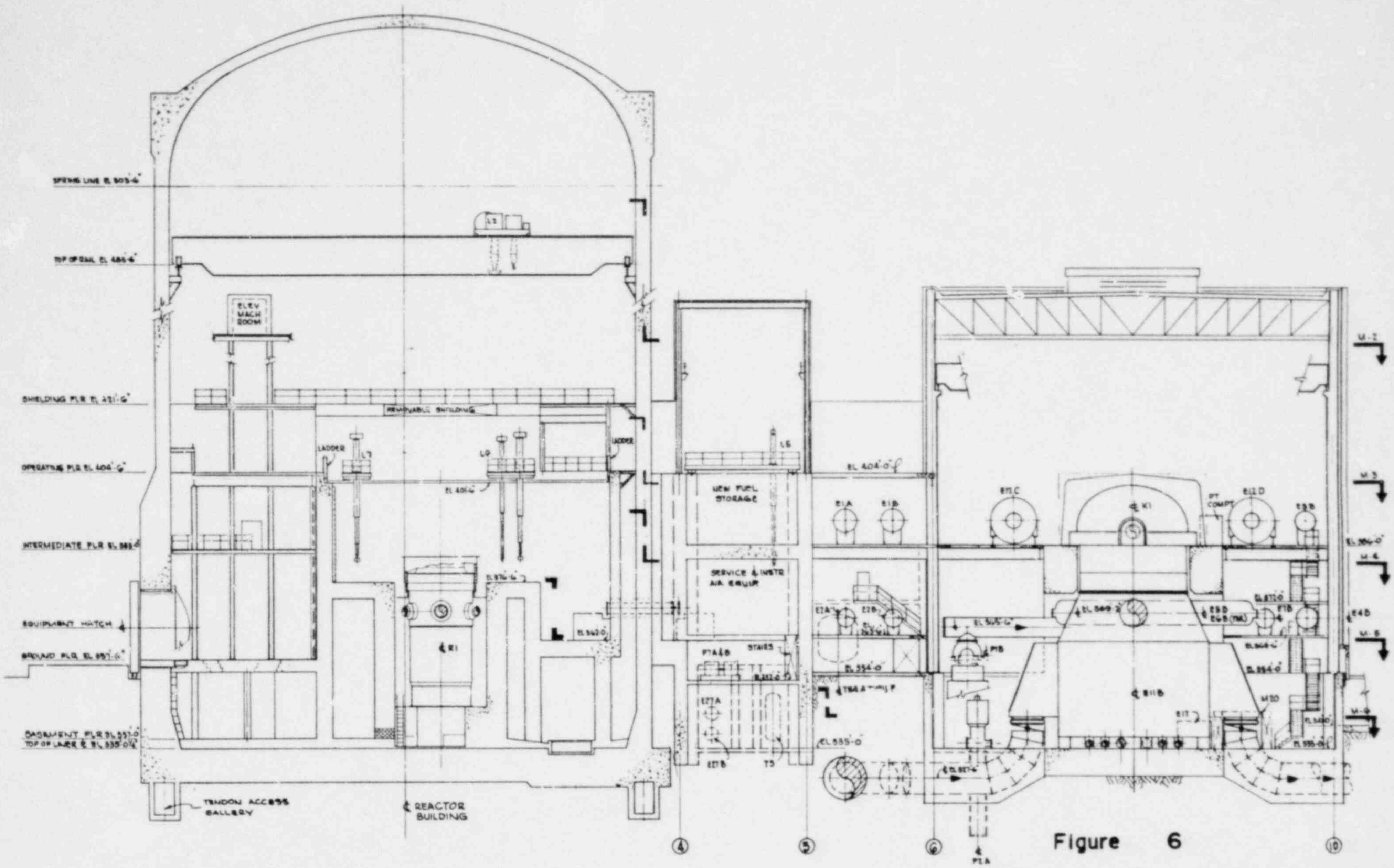


Figure 6

GENERAL ARRANGEMENT

eb 1

MR. JEWELL: Copies of this summary description are on the table in the back of the room, availableto those members of the public who may desire to look at them.

This, Mr. Chairman, completes the direct testimony of the applicant, other than the answers to questions propounded by the Board at the pre-hearing conference. And I would like to suggest for the Board's consideration that it might be appropriate for the staff to now have their witnesses sworn and present their direct testimony before cross-examining Mr. Holmes on the Summary Description, or answering the Board's questions.

CHAIRMAN WELLS: Is that agreeable to the staff?

MR. ENGELHARDT: It is agreeable to the staff.

CHAIRMAN WELLS: Will you proceed then to qualify your witnesses, Mr. Engelhardt?

MR. ENGELHARDT: Yes.

The first matter to be considered is the testimony of Charles A. Lovejoy, whose testimony is directed to the issue of the financial qualification of the applicant to design and construct this facility.

At the pre-hearing conference the availability of Mr. Lovejoy at this hearing was discussed and with the applicant's agreement, the Board agreed to consider the possibility that Mr. Lovejoy would not necessarily be required to present himself at this hearing to sponsor his testimony. The Board

eb2

informed the staff that in the event that the members of the Board would have any questions of Mr. Lovejoy whose testimony was distributed at the pre-hearing conference, that staff Counsel would be so informed. I have not received such information from the Board and from this fact I would assume that the Board has no questions to raise of Mr. Lovejoy as to his testimony.

And thus I would offer the testimony of Charles A. Lovejoy, of the Office of the Controller of the AEC, together with an affadavit of Charles A. Lovejoy who was duly sworn and deposed and stated that he had prepared the document which I have just identified, and that the contents of this testimony are true and correct to the best of his knowledge. And I would like to offer this testimony of Mr. Lovejoy, together with this affidavit as the staff's testimony with regard to the matter of the financial qualifications of this applicant, and request that these two documents, the testimony and the affidavit, be incorporated into the record of this proceeding as if read.

CHAIRMAN WELLS: Your assumption is correct, Mr. Engelhardt. The Board examined the written testimony of Mr. Lovejoy and decided it would not have questions to ask him, and the Board will be glad to receive the two documents presented by the staff, and it is ordered that they be included in the transcript accordingly.

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MR. ENGELHARDT: Copies have just been distributed to the Board, and to the applicant, and to the Reporter.

xzxzz

(Testimony of Mr. Lovejoy follows:)

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of ARKANSAS POWER AND LIGHT COMPANY (Russellville Nuclear Unit)	)))))	Docket No. 50-313

TESTIMONY OF CHARLES A. LOVEJOY
OFFICE OF THE CONTROLLER, AEC

My name is Charles A. Lovejoy. I am employed as a staff accountant in the Office of the Controller, U. S. Atomic Energy Commission, Washington, D.C.

I am a graduate of Benjamin Franklin University with an MCS Degree in Accounting.

From 1933 to 1941 I filled a variety of accounting and auditing positions with the U. S. Government. I was Controller for the United Services Life Insurance Company for two years before entering the military service in 1943. After discharge from the Army, I served for seven years in various staff and supervisory positions in the audit and accounting divisions of the Reconstruction Finance Corporation where my duties included financial analyses and review of the operations of borrowing institutions.

Since 1953, I have been a staff accountant in the Office of the Controller of the Atomic Energy Commission. My duties include the preparation of financial analyses of firms applying for facility licenses, and construction permits. I have appeared as the financial witness for the AEC staff in hearings on the applications for Class 104 licenses held over the past years.

I have reviewed the financial information in the application and amendments thereto of the Arkansas Power & Light Company for a permit to construct a nuclear power facility with an initial net electrical output of about 850 megawatts (2,452 Mwt) to be known as the Russellville Nuclear Unit. Based on this information, it is my opinion that Arkansas Power & Light Company (APL) is financially qualified to construct the proposed nuclear facility.

My opinion is based upon the following facts and considerations:

1. APL estimates that the costs of construction of the plant, including transmission facilities and other associated costs, and for the initial reactor core total about $169 million, made up as follows:

Total nuclear production plant costs	$ 138.0 million
Transmission facilities and other associated costs	7.0
Inventory costs of initial core	24.0
Total	$ 169.0 million

The estimates, as they pertain to the costs of the nuclear production plant, have been reviewed by the Division of Construction and the Division advises me that the estimates appear reasonable.

The Division of Reactor Development and Technology has reviewed the special nuclear material requirements for the first core of the subject reactor and has advised me that the quantity of 91,610 kilograms of UO_2 as stated in the Preliminary Safety Analysis Report is a reasonable amount for a reactor of this size and power level.

2. The applicant states that construction of the nuclear plant will be financed as an integral part of its total construction program in the same general manner as other additions to the APL system are financed (i.e., internally generated funds, temporary bank loans and proceeds from the sale of securities).

Based on APL's record of earnings and provisions for depreciation over the past five years and in view of APL's resources, the strength of its financial position and the regard held for its bond issues, it is a reasonable assumption that a substantial portion of the estimated costs to construct the nuclear facility (including the necessary funding for the initial core) can be financed from funds generated internally (e.g., retained earnings, provisions for depreciation) and the remainder from the sale of securities, including temporary short-term loans, when and as needed. In this connection, none of the applicant's outstanding bonds mature prior to 1974 and accordingly no refinancing of existing debt will be required during the construction period.

3. APL is soundly financed and has adequate resources at its command. As of December 31, 1967, cash and net receivables totaled about $14 million. Gross revenues for the year were $108.4 million. The long-term debt represents 56.7% of total capitalization and the company is not over-capitalized on a book value basis, as evidenced by the ratio of net plant

to capitalization of 1.1. The applicant's Dun and Bradstreet credit rating is AaA1 and Moody's Investors Service rates the company's first mortgage bonds (97.5% of long-term debt) as high-medium grade (A) and its debentures as lower medium grade (Baa).

4. Operating revenues for 1967 were $107.9 million, up 35% over 1963 and net earnings after taxes were $18.3 million, up 59% over 1963. The volume of energy sales over the same five years has increased about 50% to 8,661 million kilowatt hours in 1967. The pertinent financial ratios indicate a sound financial position and are in line with those of the electric utilities industry as a whole. A copy of my financial analysis reflecting these ratios and other pertinent data is attached as Appendix A. In brief, the ratio of the long-term debt of $224.3 million to net utility plant of $440.5 million is .51; the ratio of net plant to capitalization is 1.11; the proprietary ratio is .37; the ratio of operating expenses, including taxes, of $83.4 million to operating revenues of $107.9 million is .77; the rate of earnings on the total investment in APL is 5.4% and on the stockholders' investment is 10.7%; the interest on the bonded debt was earned 2.74 times and earnings retained in the business as at December 31, 1967 were $27.1 million.

Appendix "A"

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1967	1966	1963
Long-term debt	$ 224.3	$ 194.5	$ 145.2
Utility plant (net)	440.5	401.2	292.9
Ratio - debt to fixed plant	.51	.48	.50
Utility plant (net)	440.5	401.2	292.9
Capitalization	395.6	359.5	266.6
Ratio of net plant to capitalization	1.11	1.12	1.10
Stockholders' equity	171.3	165.0	
Total assets	460.9	420.7	
Proprietary ratio	.37	.39	
Earnings available to common equity	16.3	14.2	
Common equity	129.8	123.5	
Rate of return on common equity	12.6%	11.5%	
Net income	18.3	15.7	11.5
Stockholders' equity	171.3	165.0	121.4
Rate of earnings on stockholders' investment	10.7%	9.5%	9.5%
Net income before interest	24.7	22.1	
Liabilities and capital	460.9	420.7	
Rate of earnings on total investment	5.4%	5.3%	
Net income before interest	24.7	22.1	17.1
Interest on long-term debt	9.0	7.3	5.1
No. of times fixed charges earned	2.74	3.03	3.35
Net income	18.3	15.7	11.5
Total revenues	108.4	99.8	80.1
Net income ratio	.17	.16	.14
Operating expenses (incl. taxes)	83.4	77.7	62.7
Operating revenues	107.9	99.8	79.8
Operating ratio	.77	.78	.79
Retained earnings	27.1	23.8	18.6

	1967		1966	
Capitalization as of 12/31:	Amount	% of Total	Amount	% of Total
Long-term debt	$224.3	56.7%	$194.5	54.1%
Preferred stock	41.5	10.5	41.5	11.5
Common stock	129.8	32.8	123.5	34.4
Total	$395.6	100.0%	$359.5	100.0%

Moody's Bond Ratings:	
First Mortgage (97.5%)	A
Debentures (2.5%)	Baa
Dun and Bradstreet Credit Rating	AaA1

ebl

MR. ENGELHARDT: I would now like, with the Board's permission, to call the staff's technical witnesses to be sworn. I would like to call Mr. Charles Long, Mr. Paul Check, and Mr. Albert Schwencer.

CHAIRMAN WELLS: Will you gentlemen stand, please, and take the oath.

Whereupon,

CHARLES LONG,

PAUL CHECK, and

ALBERT SCHWENCER

were called as witnesses on behalf of the staff and, having been first duly sworn, were examined and testified as follows:

DIRECT EXAMINATION

MR. ENGELHARDT: I would now like each of you to state your full name.

MR. LONG: Charles G. Long.

MR. CHECK: Paul S. Check.

MR. SCHWENCER: Albert Schwencer. I have no middle name.

MR. ENGELHARDT: Mr. Long, by whom are you employed?

MR. LONG: U. S. Atomic Energy Commission.

MR. ENGELHARDT: In what capacity?

MR. LONG: I am Branch Chief, Reactor Project Branch No. 3 of the Division of Reactor Licensing.

MR. ENGELHARDT: Mr. Check, by whom are you employed?

eb2

MR. CHECK: I am similarly employed by the Atomic Energy Commission, the Division of Reactor Licensing.

MR. ENGELHARDT: In what capacity?

MR. CHECK: As a project leader under Mr. Long's supervision.

MR. ENGELHARDT: Mr. Schwencer, would you state by whom you are employed?

MR. SCHWENCER: I am also employed by the Atomic Energy Commission in the Reactor Projects Branch No. 3, the Division of Reactor Licensing.

MR. ENGELHARDT: In what capacity?

MR. SCHWENCER: As a project leader.

MR. ENGELHARDT: Did each of you prepare a statement of your professional qualifications?

(Chorus of "yes".)

MR. ENGELHARDT: Let the record show that each witness has responded affirmatively.

Are these the document which I now show you?

(Documents handed to the witnesses.)

(Chorus of "yes".)

MR. ENGELHARDT: These documents have previously been transmitted to the members of the Board and to the applicant. Copies of these professional qualifications are available at the table to the rear of the room.

Do you have any corrections or additions to make to

eb3

these statements?

(Chorus of "no".)

MR. ENGELHARDT: Let the record show that the witnesses responded negatively.

Are the statements in these professional qualifications true and correct to the best of your knowledge?

(Chorus of "yes".)

MR. ENGELHARDT: Do you adopt this statement as your testimony in this proceeding?

(Chorus of "yes".)

MR. ENGELHARDT: I would now offer the statements of Mr. Long, Mr. Check, and Mr. Schwencer, and request that they be physically incorporated into the record into the record of this transcript as if read.

CHAIRMAN WELLS: It is so ordered.

CHARLES G. LONG

PROFESSIONAL QUALIFICATIONS

REACTOR PROJECT BRANCH NO. 3

DIVISION OF REACTOR LICENSING

I am the chief of one of five Reactor Project Branches in the Division of Reactor Licensing. In this position, I am responsible for the analysis and evaluation of the public health and safety aspects of nuclear reactor facilities assigned to the branch.

I attended Alliance College in Cambridge Springs, Pennsylvania from 1952 to 1954, as a preengineering student. I attended the University of Pittsburgh, Pittsburgh, Pennsylvania from 1954 to 1956, and received a BS degree in mechanical engineering. I have completed several graduate courses in engineering methematics at the Pennsylvania State University.

In June 1956, I took a position with the Curtiss-Wright Corporation in Quehanna, Pennsylvania as a design engineer. I was responsible for design of various reactor components for the Projects Pluto and Zebra. As a lead design engineer, I was responsible for design of several research reactors and the preliminary design of the Army's Military Compact Reactor. As a project engineer, I was responsible for the design, fabrication and construction of the Thailand Research Reactor.

In September 1960, I resigned my position at Curtiss-Wright Corporation. I then accepted an appointment as a mechanical engineer with the Division of Reactor Licensing, U.S. Atomic Energy Commission. In this position, I participated in the safety reviews of the NS Savannah; the Humboldt Bay Reactor at Eureka, California; the Big Rock Point Reactor at Big Rock Point, Michigan; the

Elk River Reactor at Elk River, Minnesota; and several research and test reactors. In April 1962, I left the Commission to accept a position as process engineer with the Inter-nuclear Company in Clayton, Missouri. I was responsible for the reactor design and process systems for the University of Missouri Research Reactor, Columbia, Missouri.

In November 1962, I returned to the Division of Reactor Licensing as a reactor engineer. I have had the primary responsibility for safety review of the Consolidated Edison Indian Point Unit No. 1 at Indian Point, New York; the LaCrosse Boiling Water Reactor at Genoa Station, Monroe County, Wisconsin; the Florida Power & Light Company Turkey Point Units 3 and 4 at Turkey Point, Dade County, Florida; the MIT Research Reactor at Cambridge, Massachusetts; the Georgia Tech Research Reactor at Atlanta, Georgia; and the Western New York Nuclear Research Center Reactor at Buffalo, New York.

In my current position I have been the Regulatory Staff's Principal witness at public hearings for several reactor projects including Duke Power Company Oconee Station Units 1, 2 and 3, Oconee County, South Carolina; Vermont Yankee Nuclear Power Station, Vernon, Vermont; Metropolitan Edison Company Three Mile Island Unit, Dauphin County, Pennsylvania; Northern States Power Company Prairie Island Units 1 and 2, Red Wing, Minnesota; Florida Power Corporation Crystal River Unit 3, Crystal River, Florida; and Commonwealth Edison Company Zion Station Units 1 and 2, Zion, Illinois.

PAUL S. CHECK

PROFESSIONAL QUALIFICATIONS

REACTOR PROJECT BRANCH NO. 3

DIVISION OF REACTOR LICENSING

I am a project leader in Reactor Project Branch No. 3, Division of Reactor Licensing. It is my duty to perform safety evaluations of central station nuclear power plants. I have had primary responsibility for the review of Commonwealth Edison's Zion Station and, concurrently, Arkansas Power & Light's Russellville Nuclear Unit. Previously, I lead the safety review of Vermont Yankee and participated in the evaluations of TVA's Browns Ferry application and Northern States Power's Prairie Island Plant.

Before joining the Atomic Energy Commission in 1966, I spent 6 years with Pratt & Whitney Aircraft - CANEL in Middletown, Connecticut holding various positions associated with physics analysis of high-performance fast reactors. When I left Pratt & Whitney, I had been supervisor of the Analytical Physics Group for approximately 3 years.

From 1957 to 1959, I was employed by the General Electric Company's Aircraft Nuclear Propulsion Department in Evendale, Ohio as an Honors Program Engineer. My principal activity was conducting pulsed-neutron experiments to determine diffusion properties of various materials.

I have a Bachelor of Science degree in physics from Fairfield University and a Master of Science degree in nuclear engineering from the University of Cincinnati.

I am a member of the American Nuclear Society.

ALBERT SCHWENCER

PROFESSIONAL QUALIFICATIONS

REACTOR PROJECTS BRANCH NO. 3

DIVISION OF REACTOR LICENSING

I am a project leader in Reactor Projects Branch No. 3, Division of Reactor Licensing.

As a project leader in RPB No. 3, it is my responsibility to coordinate health and safety evaluations covering all aspects of nuclear power plant applications assigned to me and to assist other project leaders in evaluating other applications.

I have assisted in the safety review of the Prairie Island Nuclear Generating plant and currently have the primary responsibility for the review of the Arkansas Power & Light Company's Russellville Nuclear Unit application.

Before joining the AEC regulatory staff in October, 1967, I was associated for 5 years with the Engineering Department of the U. S. Army Nuclear Power Program. My last position was that of Chief, Power Systems Branch. In that position I directed a technical staff responsible for providing reactor facility engineering services for mechanical, electrical, protection and control instrumentation, chemical, and radiochemical control systems. Services were to specify, procure, design, develop, construct, test, inspect, maintain and provide safety reviews for the Army's SM-1, SM-1A, MH-1A; for the Airforce's PM-1 and NETF; for the Navy's PM-3A; and for several conceptual military reactor designs. While Chief, Power Systems Branch, I also held collateral duty as Project Officer responsible for the in-place annealing the SM-1A reactor pressure vessel (accomplished in August 1967).

Prior to assuming the above position, I was Chief, Electrical Engineering Branch, from 1964 through 1966, where I was responsible for evaluating all instrumentation, control and electrical systems for both the MH-1A floating nuclear plant and a "standard plant" design. During this period, I also held collateral duty as Project Engineer for establishing design feasibility of a floating 40-50 Mwe power/desalination plant. Also, I performed a safety inspection of PM-3A Nuclear Power Plant instrumentation and electrical systems in Antartica in 1965 for the Navy.

From May of 1962 until assuming the above supervisory positions, I served as an electronic engineer in the Electrical Engineering Branch and was responsible for specifying and technically evaluating (through installation and operation) a reactor protection instrumentation system and radiation monitoring system specifically developed for use in remote military locations involving extreme environments. Performed instrumentation and electrical safety inspections of Army reactors at Fort Belvoir (SM-1) and on the Greenland ice cap (PM-2A).

Prior to joining the Army Nuclear Program I was associated for 11 years with the engineering staff of the U.S. Navy's Bureau of Ships as an electrical engineer where I was responsible for several research and development projects relating to shipboard power and control systems (thermoelectric generators, cathodic protection and high power SCR's). In addition I performed engineering evaluation of magnetic amplifier and solid state instrumentation and control systems designs used in the nuclear submarines.

I received a BS degree in Electrical Engineering from the Michigan Technological University in 1951. In 1962, I completed an intensive 8-week Nuclear Power Engineering Course, conducted by the U.S. Army Nuclear Power Program's Training Department. And, based on qualification board examination, was certified as a Nuclear Plant Engineer by the Program Director.

In the area of supervisory and management training, I completed a 14-day supervisory leadership course at the Bureau of Ships in 1959 and a 14-day Management Development Program conducted by the Department of Agriculture Graduate School at Williamsburgh, Virginia and in Washington, D. C. in 1964.

ebl

MR. ENGELHARDT: I will now show each of the witnesses a copy of a document entitled "Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission in the matter of Arkansas Power and Light Company, Russellville Nuclear Unit, Docket No. 50-313," dated October 1st, 1968.

Did each of you participate in the preparation of this document?

(Chorus of "I did".)

MR. ENGELHARDT: Are there any corrections or additions to be made in the Safety Evaluation which I have just identified?

MR. LONG: Yes, there are.

MR. ENGELHARDT: Mr. Long, would you identify the corrections?

MR. LONG: On page 9, the seventh line from the top of the page, the last word of that line should read "containment" instead of "core".

On page 15, the fourth line under Section 3.2 should read "at 2452 megawatts thermal with a maximum design fuel burnup of 55,000 megawatt days." The change is the addition of the word "design" after "maximum".

That's all the corrections I have.

MR. ENGELHARDT: Is this identified Safety Evaluation as corrected by you now true and correct to the best of your knowledge?

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(Chorus of "yes".)

MR. ENGELHARDT: May the record show that all three witnesses have so indicated.

Do you adopt this Safety Evaluation as the testimony of the AEC Regulatory Staff in this proceeding?

(Chorus of "yes".)

MR. ENGELHARDT: May the record show that the witnesses have responded affirmatively.

I would now offer this Safety Evaluation which I have identified as the staff's testimony in this proceeding, and request that it be incorporated into the transcript of this proceeding as if read.

CHAIRMAN WELLS: It is so ordered.

(The Safety Evaluation follows:)

October 1, 1968

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ARKANSAS POWER AND LIGHT COMPANY

RUSSELLVILLE NUCLEAR UNIT

DOCKET NO. 50-313

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1.0 INTRODUCTION

On November 29, 1967, the Arkansas Power & Light Company (applicant) submitted an application to construct and operate a single-unit nuclear power plant, to be known as the Russellville Nuclear Unit. Ten supplements to that application have since been filed with the Atomic Energy Commission. The reactor site is located about 6 miles from Russellville on a peninsula in the Dardanelle Reservoir on the Arkansas River, Pope County, Arkansas.

The facility architect-engineer and construction manager will be the Bechtel Corporation, the nuclear steam supply system will be furnished by the Babcock & Wilcox Company (B&W), and the turbine generator will be supplied by the Westinghouse Corporation.

The plant will use a B&W pressurized water reactor designed to operate at 2452 megawatts thermal (Mwt) and produce 850 megawatts of electrical power (Mwe). The expected ultimate capacity of this plant is 2568 Mwt. The applicant has designed the major plant components including the containment and other engineered safety features for a power level of 2568 Mwt, and has used this power level in analyzing postulated accidents in conformance with the siting guidelines of Title 10 - Chapter I, Part 100 of the Code of Federal Regulations (10 CFR 100). We evaluated the containment and other engineered safety features for 2568 Mwt; however we evaluated the thermal and hydraulic characteristics at 2452 Mwt. Before operation above a power level of 2452 Mwt is authorized, the Commission's regulatory staff must perform a safety evaluation to assure that the facility can be operated safely at that power level.

The application, including the Preliminary Safety Analysis Report (PSAR) and Supplements 1-10 (hereinafter collectively referred to as the "application") was the basis on which the Division of Reactor Licensing conducted the technical evaluation of the preliminary design of the proposed plant. The staff used the following approach in its review of this application.

a. Performed an in-depth evaluation of site-related features.

b. Identified and compared all of the design and safety features of the Russellville Nuclear Unit for similarity to those previously reviewed. Where justified, we relied upon previous in-depth evaluations of like systems, components, and structures without performing separate, duplicate evaluations.

c. Determined that the design features and the treatment of safety matters were consistent with current regulatory criteria and policy, and that the applicant adequately addressed concerns which have been identified by the Advisory Committee on Reactor Safeguards (ACRS) in previous reviews.

d. Identified and evaluated those design features and related safety matters that are new or unique, or which, although reviewed in the past for other applications, continue to require review.

Within the Division of Reactor Licensing, the Reactor Projects group was responsible for the review, and for coordinating parts of the review involving personnel within the Division representing various special technical disciplines from the Reactor Technology and Reactor Operations groups, as well as consultants and other governmental agencies outside of the Division of Reactor Licensing. The reports of our consultants are attached as Appendices C th.ough G.

During the review a number of meetings were held with representatives of the applicant to discuss the proposed plant. As a consequence, additional information was received from the applicant.

The Advisory Committee on Reactor Safeguards has considered the application, has visited the site, and has met with both the applicant and the staff. A copy of the ACRS report to the Commission on the Russellville Nuclear Unit is included as Appendix A.

A chronology of the principal actions relating to the processing of the application is attached as Appendix B to this report.

The review and evaluation of the proposed design and construction plans of the applicant prior to construction constitute the first stage of a continuing AEC review of the proposed facility. Prior to issuance of an operating license, the Commission's regulatory staff will review the final, as-built, design and operating features to determine that all of the Commission's safety requirements have been met. The unit would then be operated only in accordance with the terms of th operating license and the Commission's regulations, and under the continued surveillance of the Commission's regulatory staff.

The issues to be considered, and on which findings must be made by an Atomic Safety and Licensing Board before the requested construction permit may be issued, are set forth in the Notice of Hearing published in Federal Register on September 20, 1968, 33 FR 14243.

2.0 SITE AND PLANT DESCRIPTION

2.1 Site Description

The Russellville Nuclear Unit will be constructed on an 1100 acre site located on a peninsula in the Dardanelle Reservoir on the Arkansas River in Pope County, Arkansas approximately 6 miles from the town of Russellville (1967 population, 11,154) and 2 miles from the village of London (1967 population, 495).

An exclusion area with a radius of 0.65 miles (3430 feet) from the reactor has been established for this plant. All land within this radius, except for the bed and banks of the Dardanelle Reservoir is owned by the applicant. The bed and banks of the reservoir are controlled by the U.S. Army Corps of Engineers. The applicant has obtained an easement from the Corps of Engineers for the area which will permit it to exclude all persons from this area in the event conditions at the plant warrant such action. The applicant has specified a low population zone (LPZ), as defined in 10 CFR 100, of 4 miles.

The area around the site is largely undeveloped. In 1964 practically no land was under cultivation out to 4 miles; out to 10 miles less than 0.4 percent was under cultivation. In 1964 approximately 20% of the land out to 5 miles and 27% of the land out to 10 miles from the site was classed as pasture land. The nearest population center with over 25,000 people is Hot Springs, Arkansas, 55 miles south of the site. The applicant has estimated a 1967 population of 3146 within 4 miles (LPZ) and 22,993 within 10 miles of the site. Projections of the total population within these distances have been made by the applicant for the year 2012 and are given as 5700 and 34,827, respectively.

The meteorology of the site is typical of continental locations, with lighter wind speeds and slower diffusion conditions at night than during the day. The site is in an area with appreciable tornado activity with 41 tornadoes reported per 1 degree square[1/] over a 45-year period (1916-1961).

With respect to hydrology, the maximum probable flood, as computed by the Corps of Engineers, combined with failure of the upstream dam will flood the reactor site to 361 feet or 8 feet above plant grade level. An onsite pond which will provide the source of emergency cooling water will be available in the unlikely event that there is a loss of such cooling water from the Dardanelle Reservoir.

In terms of geology, the site is near the axis of the Scranton syncline, one of several westward-trending gentle folds that characterize the Arkansas Basin--a major structural and topographic feature of Arkansas and eastern Oklahoma. The site is underlain by a thick sequence of gently-dipping shales and sandstones of Pennsylvanian age. Overburden consists of alluvial clay and silty clay that ranges in thickness from 13 to 23 feet.

No identifiable active faults or other recent geologic structures exist that would localize earthquakes in the immediate vicinity of the site. Although several ancient faults are associated with the folded structures in the area, none appear to have been tectonically active since latest Paleozoic time (about 230 million years ago).

1/ A 1 degree square as used here is that earth surface area bounded by 1 degree of lattitude and 1 degree of longitude. At the Russellville site a 1 degree square contains approximately 3000 square miles.

A somewhat unique site feature is the buried natural gas transmission pipe line which crosses the site approximately 600 feet from the containment structure. The line, which does not supply this facility, will cross 4 feet beneath the bed of the plant's discharge water canal.

A discussion of the acceptability of the site is given in Section 3.1.

2.2 Plant Description

The Russellville plant will have a closed-cycle, pressurized-water nuclear steam system housed in a prestressed concrete containment building, a steam and power conversion system housed in an auxiliary building and an outside electrical switchyard. It will also have those auxiliary systems and structures required to safely operate and maintain the plant under normal and emergency conditions. These auxiliaries include a radioactive waste disposal system, fuel storage and handling facilities, emergency power systems, and other engineered safety features.

The principal features and design bases for the Russellville Nuclear steam supply system are essentially identical to those of the Metropolitan Edison Company's Three Mile Island Nuclear Station, for which a construction permit has been issued by the Commission. The nuclear steam supply system consists of a pressurized water reactor, a reactor coolant system, and associated auxiliaries. The reactor coolant system consists of two parallel recirculation circuits, each sending reactor coolant through a steam generator (reactor coolant side) where it splits and flows through two pumps and associated piping, back to the reactor vessel.

An electrically-heated, spray-cooled pressurizer is connected to one of the two flow circuits. The reactor core uses fuel rods of uranium dioxide pellets clad in Zircaloy-4 tubes. The fuel rods are supported in assemblies by spacing grids and fittings, and a perforated can, all made of 304 stainless steel. Reactivity is controlled by movement of control rods (Ag-In-Cd), clad with 304 stainless steel, and by varying the boric acid concentration in the reactor coolant.

The control rods are positioned axially in the core by the use of electro-mechanical, rack-and-pinion rod drive mechanisms and tripped (gravity insertion for least reactivity) by deenergizing a magnetic clutch. The clutch design permits the drive motor to apply down-drive force should a rod not fall freely.

A control system monitors reactor system temperatures, pressure, flows, neutron flux and load demand, and adjusts reactor power, steam generator feedwater flow, and turbine throttle within prescribed operating limits.

A reactor protection system monitors reactor coolant system temperatures, flows, and pressure, core neutron flux startup rate, and neutron flux level. If an operating limit is reached, this system shuts down the reactor by releasing rod drive clutches and allowing the control rods to drop into the core.

The principal engineered safety features are the emergency core cooling system (ECCS), the containment ventilation system, and the containment spray system (with chemical additives). A protection system monitors primary coolant and reactor building pressures and will automatically initiate operation of the engineered safety feature systems if preestablished safety limits are reached.

The containment structure will be a steel-lined, prestressed, post-tensioned concrete, vertical cylinder with flat bottom and shallow domed roof. The containment is of the same basic design as that used by Bechtel for the Commission-licensed Duke Power Company Oconee Units 1, 2, and 3, the Florida Power & Light Company Turkey Point Units 3 and 4, and the Consumers Power Company Palisades Plant. The design details of the Russellville containment differ from this basic design in that the design details provide for a modified prestressing system using three vertical buttresses and 240°-span horizontal tendons rather than for a prestressing system using six vertical buttresses and 120°-span horizontal tendons.

All penetrations will be pressure-resistant, leak-tight, welded assemblies. Personnel hatch openings will have interlocked double doors, the equipment hatch will have a double-gasketed, bolted door and an isolation system will be provided to close all fluid lines that penetrate containment and are not required for operation of engineered safety features.

The emergency core cooling system (ECCS) will be designed to provide core cooling for any location and size primary coolant pipe break, up to and including the double-ended rupture of the largest pipe--the 36-inch reactor outlet pipe between the reactor pressure vessel and the steam generator. The ECCS will consist of two operating and one spare high pressure injection pumps, two core flooding tanks (accumulators) and two low pressure (decay heat) pumps. A recirculation system using the two low pressure pumps, returns water from the containment sump to the ECCS. The Russellville ECCS does not differ in concept or capacity from the ECCS reviewed and approved for Metropolitan Edison's Three Mile Station Unit 1.

An emergency containment spray system will provide borated water containing dissolved sodium thiosulfate and sodium hydroxide to limit containment accident pressure (by heat removal) and to remove iodine (by chemical action) in the event of an accidental energy release from the primary system. A containment ventilation system, consisting of three fin-fan air coolers, is used to maintain containment temperatures at normal values during normal plant operations. During accident conditions either the coolers alone or the core spray system alone will be capable of keeping the accident pressure within the design limit.

The major plant auxiliary systems are the chemical and volume control system, the waste disposal system and the fuel handling system. The chemical and volume control system is used to adjust the concentration of the chemical neutron absorber (boric acid) in the reactor coolant and to maintain the proper amount of water in the primary system. The waste disposal system is used to accumulate radioactive gases, liquids and solids from plant operation, process the radioactive wastes, and control and monitor the release of radioactive gases and liquids from the plant to the air and to the reservoir respectively. The fuel handling system includes equipment and facilities designed to transport spent fuel under water from the reactor to the water-filled spent-fuel storage pool from where the spent fuel will be shipped to an offsite processing plant.

The closed steam-feedwater cycle of the steam and power conversion system removes heat energy from the reactor coolant in the two once-through steam generators in the form of steam, converts steam energy into electrical energy in passing through the turbine generator, condenses the steam into feedwater which is purified, chemically controlled for optimum pH and minimum oxygen content, preheated, and recycled to the steam generators.

The condenser circulating water system condenses the steam leaving the turbine generator unit in the main condenser. The pumps for this system will withdraw water from the Dardanelle Reservoir by way of an intake canal and pump it through submerged conduits to the main condenser, and thence back to the reservoir through submerged conduits and a discharge canal. Cooling water for vital plant functions, which must remain operable in the event of an accident, will be supplied by the service water system. This system will draw water from an intake structure which is normally supplied through the intake canal from the Dardanelle Reservoir. The service water portion of the intake structure can be isolated from the intake canal and be gravity-fed by submerged piping from an elevated emergency cooling water pond to be constructed on the site.

Onsite emergency power to operate post-accident emergency core cooling systems, the containment cooling systems, and other vital systems will be supplied by two 2750 kW diesel generators. Two separate 125 volt d.c. systems, complete with charged storage batteries, will also be provided to supply vital instrumentation and provide emergency lighting and switching power.

There are two independent offsite sources of power. Offsite power can be provided automatically upon loss of the main generator, through one of two transformers from a 161 kV transmission system which will be supplied power over separate lines from different sources. Offsite power can also be provided automatically in a similar fashion from a 500 kV transmission system which can also bring power into the plant over separate lines from two different sources.

3.0 IMPORTANT SAFETY CONSIDERATIONS

In our evaluation of this application, we have given special consideration to a number of site and design features which are new, unique, require continuing evaluation, or have important safety implications. The more important of these safety considerations are discussed in the following sections.

3.1 Suitability of the Site

In evaluating this reactor site, we have considered the following aspects: the characteristics of the proposed reactor; the containment capability; the nature and amount of radioactive waste products generated; the site characteristics relating to meteorology, hydrology, geology, and seismology; abnormal weather conditions, such as tornadoes and floods; the population distribution in the surrounding area; and the potential radiation exposures at the site boundary and offsite as a consequence of any of the postulated design basis accidents.

The area around the site is sparsely populated; however, the site does present one potential problem related to evacuation of the few persons on the Bunker Hill section of the peninsula which extends into the Dardanelle Reservoir. Since the land evacuation route for these people would be across the applicant's property inside the exclusion area, the applicant will provide boats to evacuate these persons by water if the land route is unsafe.

The applicant states that no water is removed for either industrial or potable purposes downstream between the plant and the Mississippi River.

To establish background radiation levels, the applicant has outlined an environmental program which will be initiated 12 to 18 months prior to operation of the Russellville plant. This program will include onsite monitoring of radiation exposure levels and radionuclide concentrations in soil, vegetation, lake bottom, water, fish, and air. Offsite monitoring will include analyses of milk, pasture forage, truck crops, and public water supplies. The applicant has consulted with various state and federal agencies in establishing this program.

The applicant's program has been reviewed by the Fish & Wildlife Service (Appendices C1 and C2). The Fish & Wildlife Service has recommended that the applicant's program include pre- and post-operational survey studies regarding specific radionuclides and their effect on selected organisms indigenous to the area. On the basis of our review of supplementary information submitted in response to Question 2.9 in Supplement No. 3 of the PSAR, we conclude that the applicant intends to comply with these recommendations of the Fish & Wildlife Service.

We conclude that with the incorporation of these recommendations, the applicant's proposed program is acceptable.

On the basis of available data, we conclude that the site meteorology does not present any unusual problems. However, to supplement and verify the existing data, the applicant has indicated that an onsite meteorological measurement program will be conducted. We find the scope of this program to be acceptable. The applicant's meteorological assumptions relating to site diffusion factors are considered to be adequately conservative. This finding is based on independent analyses performed by the staff and by the Environmental Science Services Administration, whose comments are attached as Appendix D.

To meet our safety criteria, certain aspects of the site required further definition and/or changes to the material originally presented in the application. These matters are discussed in the following paragraphs.

The applicant added an emergency cooling water pond on the site to ensure, in the unlikely event of failure of the Dardanelle Lock and Dam, a continued source of emergency cooling water for vital plant functions.

The onsite gas transmission line, described in Section 2.1, has been evaluated for effects on the Russellville plant. The buried line, at its nearest point, is 600 feet from the reactor containment building and is 4 feet below the bed of the discharge canal. The applicant has calculated the energy potential for this line due to an explosive rupture and that due to ignition of gas discharged from the open line. Neither the explosive rupture nor the radiant energy from gas ignition at the break are considered capable of damaging this facility. The applicant has further indicated that, should such events occur, the gas line owner will close control valves on both sides of any break in the plant vicinity within 2 hours of notification. The applicant has stated that prior to plant operation, the existing pipe will be replaced with pipe constructed to Type C specification of ASA Code B 31.8 for a distance of 600 feet on each side of the crossing. On the basis of our review of this information and analysis, we do not consider this pipe line to be a significant hazard to the safe operation of this plant.

The geology of the site was found to be generally favorable by us and the U.S. Geological Survey, whose report is attached as Appendix E. In summary, our review shows that the site is underlain by shales and sandstones of Pennsylvanian age. Overburden consists of alluvial clay and silty clay that ranges in thickness from 13 to 23 feet. No identifiable active faults or other recent geologic structures exist that would localize earthquake in the immediate vicinity of the site. The limited subsurface data available indicate that the major units of the nuclear facility will be founded on a hard, dense shale which should provide an adequate foundation.

Considering the site geology, soil conditions and earthquake history, the U.S. Coast & Geodetic Survey (USC&GS) and we concluded that an acceleration of 0.1 g would adequately represent earthquake disturbances likely to occur within the lifetime of the facility and that an acceleration of 0.2 g would adequately represent the ground motion from the maximum earthquake likely to affect the site. The applicant will use these parameters in the seismic design of all Class I structures and systems. The USC&GS report is attached as Appendix F.

The applicant's original design criteria considered tornadoes having a tangential velocity of 300 mph, translational wind velocity of 40 mph, and a barometric pressure drop of 3 psi in 5 seconds. Following discussion with the regulatory staff, the applicant agreed to change these criteria to design for a tornado having a translational wind velocity of 60 mph and a barometric pressure drop of 3 psi in 3 seconds. Design basis missiles equivalent to a 4-inch by 12-foot plank traveling end-on with a velocity of 300 mph at any height and a 4000-lb auto traveling through the air with a velocity of 50 mph at a height of 25 feet or less are proposed. These values are consistent with values used by other nuclear plants recently approved for construction in areas having a significant history of tornado activity and, in our judgment, are reasonable design criteria. We conclude that the tornado design bases including the effects of tornado-generated missiles are acceptable.

In the unlikely occurrence of the maximum probable flood concurrent with the failure of the upstream Ozark Dam, the site would be flooded to a level of 361 feet which is 8 feet above plant site grade level. The applicant has considered this in the facility design and has stated that all vital equipment including service water cooling pumps either will be located above maximum probable flood level or will be protected by waterproof Class I structures. We therefore conclude that the applicant will provide adequate flood protection for this facility.

We conclude that the applicant has adequately considered the important characteristics of the proposed site. We find the proposed site to be acceptable.

3.2 Acceptability of the Nuclear Steam Supply System Design

The reactor design characteristics for the Russellville Plant are essentially the same as those for the Commission-approved Three Mile Island, Crystal River, and Rancho Seco plants. As in those plants, operation will be at 2452 Mw thermal with a maximum fuel burnup of 55,000 megawatt-days per metric ton of uranium (Mwd/MTU).

During part of the first fuel cycle the core is predicted to have a slightly positive moderator temperature coefficient of reactivity. Present calculations indicate that, with this coefficient, the core could withstand a loss-of-coolant accident and not exceed 2000° F peak fuel clad temperature. An acceptable final design value of the positive moderator temperature coefficient will be set at the operating license stage. The applicant has agreed to reduce or eliminate this positive coefficient, if necessary, to bring the consequences of the applicable accident within acceptable limits.

B&W has provided for the evaluation of xenon oscillations and in-core neutron detectors in its research and development program. To date, calculations have been performed which indicate that xenon oscillations are not expected in the azimuthal or radial direction, and are not likely in the axial direction at any time during the initial fuel cycle. Further analyses will be made using final values of core properties. Calculations have also been made to show feasibility of controlling a divergent xenon oscillation using part-length control rods. Since xenon oscillations are relatively slow flux variations which could be detected by the proposed in-core flux instrumentation, we believe that such a control technique is feasible and could be provided.

The above-mentioned in-core flux instrumentation consists of 52 fixed-position self-powered flux detectors distributed throughout the core. Normal readout is provided by the plant computer. Data obtained from this system will provide a history of fuel burnup, power distribution, and power disturbances during operation. In the event that the plant computer fails, there is an alternate readout system for selected in-core detectors.

With respect to the thermal-hydraulic parameters and design features, our review revealed nothing new or different from recently authorized pressurized water reactors. However, as noted in Section 4.0, additional analytical and experimental verification to support the choice of the fuel damage limit, the use of stainless steel shims and the use of part-length rods will be obtained before the Russellville plant receives an operating license.

We have reviewed the applicant's seismic design bases pertaining to the reactor vessel, reactor internals, and other Class I (seismic) mechanical systems and components. These systems will be designed to withstand normal design loads of mechanical, hydraulic, and thermal origin, plus applicable earthquake loads, as well as concurrent accident-induced blowdown loads. Our evaluation of the proposed design criteria for reactor internals and Class I mechanical systems and components indicates that they will provide an adequate margin of safety.

One aspect which we are reviewing in detail is that of thermally-induced stresses in the pressure vessel during actuation of the emergency core cooling system. The initial results of the applicant's analysis of this accident indicate that no loss of vessel integrity would be experienced even if large flaws were presumed to exist in the vessel wall at the beginning of the quenching. However, in view of the uncertainties associated with the analytical methods used to arrive at these results, the applicant plans to continue his work on this problem. While there remain uncertainties in the analyses being pursued, it is important to note that there is a significant time available (about 5 years) until material properties will be affected by irradiation to an extent that will be of concern. Further, it appears that there are means that can be employed, if necessary, to reduce the potential for vessel failure resulting from thermal shock and to mitigate the consequences of such a failure should it occur.

As recommended by the ACRS (Section 5.0), we will continue to review information subsequently developed concerning thermal shock on the pressure vessel to ensure that the calculational models used are not in conflict with experimental data.

Provided that the development program substantiates the reactor design characteristics discussed above, we conclude that the design of the nuclear steam supply system is acceptable.

3.3 Engineered Safety Features Adequacy

Engineered safety features for this plant include the emergency core cooling system (with reactor vessel internal vent valves), the containment ventilation systems, and the containment spray systems, and associated iodine removal system.

The emergency core cooling system (ECCS) is described in Section 2.2 of this report. The applicant's design basis is the same as that of Crystal River and other Babcock & Wilcox-designed systems recently reviewed. That basis is to prevent fuel clad melting for the entire spectrum of reactor coolant system failures from the smallest leak to complete severance of the largest reactor coolant pipe. To provide assurance that this criterion is met and to prevent any mechanical damage that might interfere with core cooling, the applicant has sized the emergency core cooling systems to limit the clad temperature transient to 2300° F or less. The calculated peak clad temperature, about 1950° F, occurs transiently during the postulated hot-leg break (a 14.1 ft^2 break).

We have reviewed the applicant's failure mode analysis of the ECCS and have concluded that adequate short-term cooling can be provided at high and low vessel pressures even in the event of failure of any single active component. In addition, adequate redundancy is provided to accommodate failure of a single active or passive component without jeopardizing the ability for long-term core cooling with the ECCS in the recirculation mode. To achieve this, the applicant revised his originally proposed ECCS design to provide two systems with no sharing of active components and minimum practical sharing of passive components. This applicant's ECCS as revised is the same as those systems previously reviewed. The result is that there are two separable core cooling systems which share only the passive borated water storage tank, core flooding tanks, and containment building sump. Sharing of the tanks is acceptable since they are in use for only a short period of time. Sharing of the reactor building sump is acceptable since the recirculation lines for the two systems take suction from different locations of the sump, the sump is covered with a grating and heavy duty strainers are provided.

As was done in the B&W nuclear steam supply system design provided for the Three Mile Island, Crystal River, and Rancho Seco plants, the Russellville design incorporates one-way internal vent valves in the reactor core barrel to prevent steam binding above the core. In the event of a loss-of-coolant accident initiated by a break in a cold leg of a reactor loop, the valves will open to permit steam generated in the core to flow directly to the leak and thus not prevent the emergency core coolant system from keeping the core adequately covered. These valves have been previously authorized for use in the Three Mile Island plant.

B&W has made a preliminary sensitivity analysis using worst case parameters to show how loss of core flow, by shunting reactor coolant through a failed (open) valve, affects the DNB ratio (design limit is 1.3 or greater). The preliminary analysis shows the reduced-flow DNB ratio is 1.68 at 100% power, 1.30 at 112% power and 1.24 at 114% power (the highest thermal power calculated in any operational transient). An analysis based on the final design of the core is expected to meet the 1.3 DNB ratio design requirement at 114% of rated power as well. We also considered the ability to detect, by change in measured reactor coolant loop flow, the failure of more than one vent valve. Based on the preliminary design data supplied by B&W, the total system flow is increased by 1.1% by failure of one valve. The applicant has stated that flow distribution studies will be made using a model of the re ctor to simulate failure of the vent valves. Completion of valve testing, including vibration tests, is expected by January, 1969. At the operating license review on this plant, or earlier, we will evaluate the results of these tests and verify the ability to identify failure of the vent valves by detection of changes in reactor coolant flow. We conclude, at this stage of our review, that the vent valve design is satisfactory subject to completion of the final design, design analyses, testing, and verification of ability to use flow change to detect failure.

Two diverse methods are provided for containment heat removal under accident conditions: (1) two 120 x 10^6 Btu/hr capacity containment spray systems, each of which takes relatively cool water (initially from the borated water storage tank and later from the containment sump) and delivers it to the containment atmosphere through a spray header and (2) three 80 x 10^6 Btu/hr capacity containment cooling systems, each consisting of a fan and tube cooler, which removes heat from the containment air and transfers it to the low-pressure service water system.

The containment cooling requirement is that the post-blowdown reactor building pressure be maintained below the containment design pressure. This requires an initial heat removal capacity of about 240 x 10^6 Btu/hr. This requirement can be satisfied if either all sprays or all containment cooling systems are assumed to be inoperative. It can also be satisfied if one spray and one cooler are inoperative. On the basis of our review of these systems, we conclude that adequate capacity has been provided to initially limit and subsequently reduce the containment pressure (and thereby reduce leakage) after the design basis accident, in the event such an improbable accident should occur.

A chemical additive (sodium thiosulfate with sodium hydroxide) will be mixed with the spray water to remove iodine from the containment atmosphere following a loss-of-coolant accident. Two spray systems are provided as discussed above. Each spray system has the design capability to deliver an adequate amount of the chemically treated spray to the containment atmosphere to prevent exceeding 10 CFR 100 guidelines for potential radiological doses at the site boundary and at the low population zone boundary. Section 3.7 gives the calculated doses using a single spray system and also states that, in the event additional chemical iodine spray tests now underway indicate that the spray system is not as effective as anticipated, iodine reducing charcoal adsorber units can be added to remove iodine.

The service water system shown on Figure 9-4 of the PSAR provides all water required for emergency cooling of the engineered safety feature equipment including the containment building coolers and the emergency diesel generators. Redundant pumps and piping and an emergency reservoir are provided such that no single failure can cause loss of required cooling.

3.4 Foundation and Structural Design Adequacy

In evaluating the foundation and structural design of the plant structures, we and our consultant,[1/] considered the following general aspects: the geology and nature of the subsoils, the seismic design parameters, site flooding, tornado wind loadings, and the effects of missiles generated from tornadoes and internal plant sources. We considered the following specific aspects in our evaluation of the containment and other Class I structures: design criteria, specifications and inspection for concrete reinforcing, selection of loads, load combinations and allowable stresses for the structure, liner and liner anchorage criteria, tendon and tendon anchorage criteria, design of penetrations, and containment strength and leak testing.

All structures and equipment required for plant safety and to maintain the integrity of engineered safety feature systems have been designated as Class I. All other structures and equipment are Class II. All Class I structures will be designed to behave elastically under normal and accident loads, except that limited yielding will be permitted under a combination of dead load, piping thermal shock or rupture, and design-basis earthquake (0.2 g). Class II structures, which do not perform vital safety functions, will be designed to Zone 1 requirements of the Uniform Building Code. Class II equipment will be designed for an equivalent horizontal loading of 0.05 g.

1/ Nathan M. Newmark Consulting Engineering Services. See report attached as Appendix G.

The containment building, as noted earlier, is similar to other Bechtel designs including the design for the Rancho Seco plant, except that heavier tendons and three instead of six buttresses are used. The cylinder has staggered 240°-span instead of 120°-span horizontal tendons. The vertical and the dome tendon systems are similar to those used in previous designs, except for anchorage designs and tendon sizes.

In response to our questions on several aspects of structural design, the applicant provided additional supporting details on methods of analysis, and construction details.

We and our consultants have reviewed the proposed tendon systems tentatively selected by the applicant. We conclude that use of the tendon systems proposed, with up to 184 wires per tendon, would be acceptable.

The liner anchorage design is similar to that proposed for the Rancho Seco plant. The liner anchorages are designed to fail before the liner itself can fail. We have expressed concern that, with the liner in compression and tending to buckle locally, anchors may fail rapidly and sequentially. On the basis of our review, we do not believe the analyses presented in the PSAR are conclusive. We have discussed this with the applicant and (as noted in Section 4) prior to construction we will obtain confirmatory test specimen data that deal with gross liner failure considerations.

In the tendon anchor zone, we are concerned that sufficient reinforcing be included in the design to cover all possible tension stresses that may exist in this zone. The usual design methods neglect two potentially significant tensile stresses, those generated by temperature gradients and by concrete shrinkage. As noted in Section 4, prior to construction of these anchorages, we will obtain from the applicant analyses and qualification test data to confirm design adequacy.

The problem of tornado-induced loss of water from the fuel pool leading to fuel melting and fission product release is of continuing concern. We have examined the analysis provided by the applicant in this regard and find that it contains no new information or arguments that have not been presented in previous applications.

We are continuing to examine the requirements for spent fuel pool design and we conclude that the design of the fuel storage pool should be such that protection of the pool from water removal effects could be added if this is found necessary. The applicant has agreed to provide this capability in the design of the Russellville fuel storage pool.

The applicant has proposed a 2% statistical sampling program for strength testing of the Cadweld reinforcing bar splices to be made in the structures. Since this may result in a small number of welds being tested, we are examining the area further. In the event that a modified testing program is considered necessary requiring a larger number of welds being tested or placing more emphasis on selected weld locations, we conclude that these relatively minor changes can be agreed upon with the applicant prior to the actual placing of these welded splices in the structures.

From our in-depth review, we and our seismic design consultant conclude that the containment, foundation and general structural designs proposed for the Russellville plant are acceptable except for the submission of confirmatory data on liner and tendon anchorages and Cadweld splice tests. These items have been left for later consideration as discussed in Section 4.0.

3.5 Adequacy of Instrumentation, Control and Emergency Power Systems

The instrumentation and control systems were evaluated and found to comply with the Commission's General Design Criteria (see Section 3.8) and IEEE 279, Proposed Criteria for Nuclear Power Plant Protection Systems. A comparison was also made with the systems proposed for the Three Mile Island Nuclear Station, and Crystal River Unit No. 3. The applicant has verified, and we concur, that the proposed design of the instrumentation and control systems for the Russellville plant and the above mentioned plants are substantially identical in concept except that the Russellville plant (a) uses not one, but all four of the redundant reactor power level channels in an averaging system as inputs to reactivity control; (b) initiates reactor trip upon loss of any two pumps while the other plants utilize systems which permit continued operation with the loss of one pump in each loop provided power is below a predetermined safe limit; (c) supplements reactor coolant systems code safety valves with a pilot actuated relief valve which is not provided in the other plants, and (d) varies boiler feed pump speed as the major means of controlling feedwater flow as opposed to reliance in the other plants solely upon feedwater valve control. The differences noted in (b), (c) and (d) above are considered to be minor and to have no significant effect on reactor safety. In evaluating item (a) we examined the proposed design and found it to be in compliance with IEEE 279. In particular, the protection system has four redundant power level channels. The random failure of any one channel leaves three for protection, only two of which are required. While these channels are also connected to the plant's reactivity control system, a single random failure in any one channel is prevented from causing a control failure by isolation devices and by the manner in which they are combined. Further the applicant reports that tests have been successfully performed simulating open circuits, short circuits, grounds, and faults to high voltages with no failures propagating beyond the channel in which the simulated failure was imposed.

As a result of our evaluation of item (a) we conclude that the design provides satisfactory protection against random failures. We will continue to work with the applicant to ensure that it takes into account, in completing the design of protection and control instrumentation, the possibilities of common failure modes such that by the suitable use of redundant devices with functional and equipment diversity, the proposed interconnections of protection and control instrumentation will not adversely affect plant safety.

The control room contains instrumentation and controls necessary for safe operation of the nuclear facility. Safe occupancy of the control room during abnormal conditions is provided for in the design. In the event the control room becomes uninhabitable, sufficient instrumentation and controls are provided at local stations which permit the operator to maintain the reactor in a hot standby condition. Further, the applicant has stated that the capability to perform an orderly cold shutdown from outside the control room, should this room become inaccessible for a long period of time, will be provided. We conclude that the control room design bases meet the intent of Criterion 11 of the General Design Criteria.

The applicant has established criteria for the selection, protection, and routing of all control, power and instrumentation cables. We conclude that adequate measures will be taken to prevent and minimize the possibility of fire or other damage in electrical cabling.

We have evaluated the proposed offsite and onsite electric power systems and have concluded that they comply with Criterion 39 of the General Design Criteria.

In its letter on the Three Mile Island Nuclear Station, the ACRS recommended that consideration be given to the development and utilization of instrumentation for prompt detection of gross failure of a fuel element. The applicant has indicated that it will provide continuous radiation monitors in the reactor coolant makeup and letdown line and in the containment atmosphere sample line with sufficient sensitivity to promptly detect a gross fuel element failure. Information on the response time as a function of fuel failure severity will be made available during the detailed design of the plant. We will review this matter on other plants scheduled for operation before the Russellville plant, and at the operating license stage review of the Russellville plant.

On the basis of the foregoing, we have concluded that the reactor instrumentation, control, and emergency power systems are acceptable for this construction permit stage of review.

3.6 Radioactive Waste Disposal Adequacy

The radioactive liquid wastes generated in normal plant operations will be collected, stored, treated, measured for activity, and discharged on a batch basis with continuous monitoring during discharge through a line to the plant's circulating water discharge canal. Gaseous wastes will be collected, monitored, diluted and released to the atmosphere. If the activity levels exceed precribed limits, the gases will be compressed and stored in waste gas decay tanks. Following decay, the stored gases will again be monitored prior to release to assure that release is within prescribed limits. Solid radioactive wastes accumulated from plant operation will be temporarily stored onsite. Shipment from the site will be in containers approved for that purpose.

We reviewed the possibility of activity release due to system failures. The solid and liquid disposal equipment is located in shielded, controlled-access areas of a Class I structure with provision for contamination control in the event of spills or leakage. Calculations by us and the applicant indicate that failure of a waste gas tank containing maximum activity would result in whole body doses of less than 2 rem at the site boundary which is well below 10 CFR 100 limits.

On the basis of our review, we conclude that the proposed radioactive waste disposal system will adequately control the radioactive wastes generated from plant operations.

3.7 Analysis of Radiological Consequences from Potential Accidents

Potential accidents which could result in radioactive releases to the environment have been analyzed by the applicant. We have evaluated these accidents and the engineered safety features provided to mitigate or limit the potential offsite exposures. Accidents which have been considered are: the loss-of-coolant accident, the rod-ejection accident, rupture of a steam pipe, rupture of a steam generator tube with loss of offsite power, fuel-handling accident, accidental release of radioactive liquid and gaseous waste, and rupture of a recirculation line in the emergency core cooling system. Of those accidents considered to have a potential for significant releases of radioactivity to the environment, the loss-of-coolant accident would result in the highest potential offsite doses.

For accidents involving loss of coolant from the primary system, the emergency core cooling systems are design to limit fuel cladding temperatures to well below the melting temperature, to prevent shatter of the fuel cladding, and to limit fission product release from the fuel. However, for conservatism we assume that the containment and its associated engineering safety features must be capable of limiting potential doses in conformance with 10 CFR Part 100 guidelines assuming releases of fission products from the fuel based on TID-14844[1/] release fractions. Using these fission product release fractions available for leakage from the containment, and assuming ground release, conservative meteorological diffusion parameters and design data on the containment sprays, we calculated potential doses at the exclusion boundary and the low population zone radius. Utilizing conservative values for drop size spectrum and deposition velocity and the specific characteristics (e.g., droplet size, flow rate, fall distance, terminal velocity of drop) of the Russellville plant's iodine removal system, we have calculated that iodine removal factors of 4.1 for the 2-hour dose and 10 for the 30-day dose are achievable by the sprays. These dose reduction factors assume as much as 10% of the iodine in

1/ TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, DiNunno, J. J., et al, March 23, 1962.

the containment is in "nonremovable" (organic) form. Allowing these dose reduction factors for iodine removal, the potential 2-hour doses at the exclusion area boundary (0.65 miles) are 4 rem whole body and 210 rem to the thyroid and the 30-day doses at the low population zone radius (4 miles) are about 2 rem body and 81 rem to the thyroid. The applicant has stated that analytical and experimental work on the efficiency of chemical additive sprays is being conducted by B&W, Oak Ridge National Laboratory and others. In addition to sodium thiosulfate, other chemical solutions are also being evaluated. In the event that the results of these development programs indicate that the spray systems might not be as effective as anticipated, the applicant has stated that space will be reserved in the plant so that charcoal adsorber units can be added to further reduce the iodine concentration in the containment.

.8 Design Conformance to AEC General Design Criteria

The applicant has assessed the Russellville Nuclear Unit design with respect to conformance with the Commission's General Design Criteria published in the Federal Register on July 11, 1967. We have evaluated the application for conformance with the revised criteria and have concluded that the preliminary design of the proposed unit conforms to the intent of these criteria. Recognizing that the proposed criteria, as revised, may be further modified as a result of comment by interested parties, and that the final design may differ somewhat from the preliminary design, we intend to review the proposed unit for conformance to the General Design Criteria again at the operating license stage.

3.9 Emergency Plans

The scope of emergency planning by the applicant, including proposed preparation of written procedures covering reasonably foreseeable emergency operating conditions, is acceptable. Detailed emergency plans for the low population zone will be developed by the applicant in cooperation with state and local authorities. We will evaluate these plans at the operating license review stage.

4.0 RESEARCH AND DEVELOPMENT

There are a number of areas related to pressurized water reactors for which additional research and development will be required. These areas are summarized in this section. We will follow the programs listed below by meeting with the applicant and his contractors and by evaluating reports submitted on these programs. (Expected completion dates are parenthetically noted).

(1) B&W Development of the Emergency Core Cooling System Design

The core cooling research and development being conducted by B&W, must specifically include (a) the completion of the analysis of the spectrum of small break sizes in the loss-of-coolant accident, (b) the development of the analytical techniques for determining blowdown forces on reactor internals, and (c) demonstration that the injection coolant will cool the core including consideration of core bypass or formation of a vapor lock. Experimental vibration tests will also be performed to show that induced-vibration will not unseat the core barrel vent valves. (July 1969).

(2) B&W Development of Final Reactor Thermal-Hydraulic, Nuclear and Mechanical Design Parameters

Development work to be performed includes the following:

a. Thermal and Hydraulic Programs

The applicant has proposed scaled flow distribution tests on the vessel and internals and rod bundle tests to determine local mixing and flow effects. This further experimental and analytical work must be done to determine the limiting heat fluxes at various positions within the fuel bundle if the design is to be based on the B&W heat transfer data. (prior to 1969)

b. Fuel rod failure mechanisms during loss-of-coolant-accident (LOCA). Various failure modes of the fuel rods during the LOCA, such as clad melting, eutectic formation, bulging, splitting, or brittle failure, will be examined in an experimental program to assure the continue core cooling capability during a LOCA. (late 1969).

c. High burnup fuel tests
Fuel specimens will be tested at heat rates ranging up to 21.5 kw/foot, burnup ranging up to 75,000 MWD/MTU, and with cladding surface temperature of 650°F. (June 1970).

d. Xenon oscillations
The applicant will further develop analytical techniques to determine the stability margins with respect to xenon oscillations (late 1969). If the stability margins are found to be insufficient, a system for stabilizing and controlling the oscillations will also have to be developed. Results from physics tests on Duke Power Company's Oconee Unit 1 will be used to confirm the analytical results. (2nd quarter 1971).

(3) B&W Control Rod Drive Unit Tests

The prototype tests are being conducted on the B&W control rod drive units under operating temperature, pressure, flow and water chemistry and should provide design adequacy information on the operability and reliability of the system. (prior to 1969).

b. Fuel rod failure mechanisms during loss-of-coolant-accident (LOCA).

Various failure modes of the fuel rods during the LOCA, such as clad melting, eutectic formation, bulging, splitting, or brittle failure, will be examined in an experimental program to assure the continue core cooling capability during a LOCA. (late 1969).

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(4) B&W In-Core Neutron Detectors Tests

The self-powered in-core neutron detectors, which have been developed by B&W, are currently under life testing at B&W's Lynchburgh facility and at the Big Rock Point Nuclear Power Plant. The status of the tests to date are acceptable.

(5) B&W Once-through Steam Generator Development and Tests

Investigations of steady-state conditions and operational transients have been completed. Vibrational tests, including vessel response to primary system blowdown, have also been investigated and the thermal response to both primary and secondary blowdown determined. The remaining work involves the development and verification of analytical models for steam system blowdown analyses. (1st quarter 1969).

(6) B&W Development of the Design Details of Iodine Removal System (Chemical Additive to Containment Sprays)

The Russellville plant iodine removal system is being developed by B&W. Chemical characteristics, iodine removal characteristics, compatibility, and radiolysis of spray materials are being evaluated. Experimental investigation of the relationship of absorption rate of containment atmospheric conditions, the effects of process variables on spray nozzle performance and the extent of radiolysis are being conducted by B&W, Oak Ridge National Laboratory, and Battelle Memorial Institute. (early 1969)

4.1 Other Matters to be Further Evaluated During Construction

(1) Instrumentation

There are two areas of instrumentation which will require further information and review.

a. Design of the prompt fuel failure detectors

The applicant has not yet completed the design of these detectors. Upon completion of these detectors, which are to be of two types, one to sample reactor coolant (in the letdown line) and the other to sample containment air, we will review their design capability for adequacy and speed of response as a function of percent of fuel failed.

b. Interaction of control and protection systems

As discussed in Section 3.5 we and the applicant will continue evaluation of the protection and control instrumentation systems with regard to interaction. In particular, we are reviewing the proposed design as it is finalized, for common failure modes, taking into account the possibility of systematic, nonrandom, concurrent failures of redundant devices, not considered in the single-failure criterion.

(2) Containment Design Details

Three containment items have been selected for further evaluation prior to construction of the affected subsystems. This information, which will be developed in the normal course of design, includes the design details and associated analyses for the tendon anchor system and for the liner anchorages.

For tendon anchorages, the applicant has agreed to submit a report giving both predictions and results of the tendon anchorage qualification test. This report, will identify analytical methods and material properties used in the predictions, results of actual tests and comparision of predictions

a. Design of the prompt fuel failure detectors

The applicant has not yet completed the design of these detectors. Upon completion of these detectors, which are to be of two types, one to sample reactor coolant (in the letdown line) and the other to sample containment air, we will review their design capability for adequacy and speed of response as a function of percent of fuel failed.

b. Interaction of control and protection systems

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(2) Containment Design Details

Three containment items have been selected for further evaluation prior to construction of the affected subsystems. This information, which will be developed in the normal course of design, includes the design details and associated analyses for the tendon anchor system and for the liner anchorages.

For tendon anchorages, the applicant has agreed to submit a report giving both predictions and results of the tendon anchorage qualification test. This report, will identify analytical methods and material properties used in the predictions, results of actual tests and comparision of predictions with test results. We plan to review this data as it becomes available as well as additional design information prior to construction of the tendon anchorages.

For liner anchorages, the applicant has agreed to perform tests demonstrating his design will not result in sequential anchorage failures. We plan to review these tests as well as additional design information prior to construction of the liner anchorages.

For Cadweld splices, we and the applicant will agree on the relatively minor changes, if any, required in the statistical sampling strength testing program prior to use of such plices in the plant structures.

(3) Quality Assurance Information

After the constructor has been selected and prior to starting any major construction at the site, we will review the additional quality assurance information, indicated in Section 6.2, which the applicant has agreed to submit.

(4) Reactor Vessel Thermal Shock

As discussed in Section 3.2 we are continuing our review of the problem of thermal shock as a potential consequence of actuation of the core cooling systems.

4.2 Conclusion

We have examined each of the above areas and conclude that they can reasonably be left for later consideration. Moreover, on the basis of the descriptions supplied by the applicant, we conclude that the proposed research and development programs are reasonably designed to resolve the identified safety questions.

5.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards, by letter to Chairman Seaborg, dated September 12, 1968, reported on the Russellville Nuclear Unit. A copy of this letter is attached as Appendix A. The letter contains comments and recommendations which we are implementing, as noted in appropriate sections of this safety evaluation.

The Committee has reiterated its belief that additional consideration be given to common mode instrumentation failures not considered in the single-failure criterion. This is discussed in Section 3.5. The Committee also emphasizes the importance of quality assurance and quality control programs, discussed in Section 6.2; and early training of a sufficient number of personnel for the operating staff, discussed in Section 6.1. Modification of the containment prestressing system design is also mentioned. This is discussed in Section 3.4.

The Committee further calls attention to other matters that warrant careful consideration by the manufacturers of all large, water-cooled, power reactors. These matters, applicable to the Russellville plant involve the following: effects of blowdown forces on primary system components, effects of fuel clad perforation on emergency core cooling performance, and fuel element performance under operational transients, all of which are addressed in Sections 3.2, 3.3 and 4.0 of this report. Additional matters about which the Committee expressed concern include pressure vessel shock from cold water injection, discussed in Section 3.2; prompt detection of gross failure of a fuel element, discussed in Section 3.5; and primary system quality assurance, discussed in Section 6.2. These items will be resolved to our satisfaction as the design work progresses and will be reviewed by the ACRS prior to issuance of an operating license.

The report of the ACRS concluded, ... "The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing items, the proposed reactor can be constructed at the Russellville site with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

6.0 TECHNICAL QUALIFICATIONS OF THE APPLICANT

6.1 Technical Qualifications

We have reviewed the application with respect to the technical qualifications of the Arkansas Power and Light Company (AP&L) and its contractors to design and construct the proposed facility. AP&L has over 45 years experience covering design, construction, and operation of conventional steam, hydro, and diesel electric generating plants which, at the end of 1967, had a total capacity of 1,734 megawatts.

Officers and engineering personnel of AP&L have had previous nuclear experience through AP&L's participation, as a member of the Southwest Atomic Energy Associates, in the Southwest Experimental Fast Oxide Reactor Facility, SEFOR, and through AP&L's participation in the Peach Bottom Atomic Power Station project.

AP&L will rely upon its architect-engineer, contractors, and consultants for technical support during the design and construction of the plant. The Bechtel Corporation has been retained as the architect-engineer and will be responsible for procurement and management of construction of the plant.

Bechtel has wide experience as architect-engineer and engineer-constructor for several pressurized water reactor power plants as well as other types of nuclear and conventional power plants. Babcock and Wilcox will supply the nuclear steam supply system and two fuel cores. B&W has extensive background in supplying nuclear steam supply systems. The turbine generator and its auxiliaries will be supplied by the Westinghouse Electric Corporation.

The number of people proposed for operation of the plant totals 61. Personnel assigned to the plant will have extensive experience in conventional power plants and all supervisory and operating personnel will be given special nuclear training including operator training at a comparable nuclear power plant. The applicant has planned for four-man operating shifts consisting of a shift supervisor with a Senior Operator's License, a plant operator and an assistant plant operator, each with an Operator's License and an auxiliary operator who may have an Operator's License.

On the basis of our review, we conclude that the applicant and its principal contractors have the technical competence to design and build the Russellville Nuclear Unit. We believe, however, that 4-man operating shifts may prove inadequate. We will pursue this matter further with the applicant as it develops its emergency and normal operating procedures and will satisfy ourselves that its training program will assure timely availability of adequate operating manpower.

6.2 Quality Assurance and Quality Control

We have reviewed the quality assurance and control program proposed for the Russellville facility. At our request, the applicant has supplemented its PSAR with additional information which is provided in Supplement 3 (answers to Questions 8.1 through 8.11, 9.5 and 9.7) and in Supplement No. 9.

The applicant's Safety Review Committee reviews all plant designs, specifications, and procedures to ensure compliance with all plant design criteria, codes and standards as set forth in the PSAR with responsibility and authority to reject those which are not in compliance. The AP&L Manager of Safety, who reports directly to the Executive Vice President, is a member of this committee.

AP&L also has established a Quality Assurance Committee (QAC) for the Russellville plant. A key member of the QAC is the Chief Quality Control Coordinator who will be in residence full time at the plant site during construction. He will work closely with the Bechtel Quality Assurance Engineer, who will also be onsite during construction. The Chief QC Coordinator will review all inspection and test procedures prior to inspection or test, monitor tests and inspections at the site and at vendor facilities on a frequent "spot-check" basis and review the results of all quality control programs. The QC Coordinator will be assisted in his duties by AP&L Engineering or Production Department personnel experienced in plant design and construction. In areas where AP&L does not now have experienced personnel, they will either hire or obtain the services of such personnel through a consultant firm.

In addition to the applicant's organization, Bechtel will have a Quality Assurance Engineer (QAE) under the Project Engineer and a separate field inspection force under a Job Engineer. The QAE will have access to and will review, for compliance with established requirements, all Bechtel and vendor quality control procedures and reports of all tests and inspections performed by others in vendors' plants and at the job site. The Bechtel field inspection force reports through the Job Engineer and Project Superintendent to the San Francisco Office Construction Manager while the QAE reports through the Project Manager and Nuclear Power Engineering Manager to the San Francisco Engineering Manager. Bechtel will also have independent checks on quality assurance during the design and pre-fabrication phase by having design bases, designs, and procurement documents, which are prepared by the Project Engineer's staff, reviewed by the staffs of Chief Engineers in each engineering specialty. These Chief Engineers independently report directly to the San Francisco Office Manager Engineering.

B&W, the nuclear steam supply system vendor, has recently established in July of 1968 a quality assurance organization which will be responsible for quality assurance of B&W's nuclear product line from bid proposal to final customer acceptance. This organization, which is independent from the previously existing B&W design, production, and quality control groups, reports directly to the Vice President of the B&W Nuclear Power Generation Department and is responsible for assuring that the Russellville nuclear steam supply system furnished by B&W conforms to all established requirements.

Upon selection of the general contractor for construction of the Russellville facility, the applicant has agreed to submit the following information: (a) a list of all organizations involved in the design and construction of this plant, (b) description of the various responsibilities of all organization including quality assurance and control, (c) a schedule of major construction activities, (d) a listing of responsible persons (plant site and vendor shops) as contacts for Division of Compliance inspectors, (e) location of complete specifications and quality assurance and control documents, and (f) a list of all major vendor shop locations.

Subject to our review of this additional information, we conclude that the applicant, together with its contractors, will have an adequate quality assurance program and that independent checks on quality assurance and quality control can be provided at all stages, from establishing adequate design bases initially, through design, fabrication, testing and final inspection.

7.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. We find nothing in the application to suggest that the applicant is owned, controlled or dominated by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the regulations. The applicant will obtain fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

8.0 CONCLUSIONS

On the basis of the proposed design of the Arkansas Power and Light Company's Russellville Nuclear Unit; the criteria, principles, and design arrangements for systems and components thus far described, which include all of the important safety items; the calculated potential consequences of routine and accidental release of radioactive materials to the environs; the scope of the development program which will be conducted; and the technical competence of the applicant and the principal contractors; we have concluded that the appropriate findings as set forth in the notice of hearing of this proceeding, September 20, 1968, can be made by the Director of Regulation.

In summary, we conclude that the proposed plant can be built and operated at the proposed location without undue risk to the health and safety of the public.

APPENDIX A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SEP 1 2 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON RUSSELLVILLE NUCLEAR UNIT

Dear Dr. Seaborg:

At its one-hundred-first meeting, September 5-7, 1968, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Arkansas Power and Light Company to construct the Russellville Nuclear Unit. This project had been considered previously during Subcommittee meetings on August 23, 1968, at the site, and on September 4, 1968, in Washington, D. C. In the course of its review, the Committee had the benefit of discussions with representatives and consultants of the Arkansas Power and Light Company, the Bechtel Corporation, the Babcock and Wilcox Company, and the AEC Regulatory Staff. The Committee also had available the documents listed.

The plant will be located about six miles from Russellville, Arkansas, on a peninsula formed by the Dardanelle reservoir. The normal elevation of the reservoir is controlled downstream by the Dardanelle Lock and Dam No. 10 on the Arkansas River. An emergency reservoir on the site will provide adequate storage of water in the unlikely event of failure of Lock and Dam No. 10. The consequences of the maximum probable flood have been studied, and adequate protection has been provided for the critical equipment of the nuclear unit.

The proposed nuclear unit is a pressurized water reactor, 2452 MWt and 850 MWe, and is similar to previously approved units (e.g., Rancho Seco, Crystal River, and Three Mile Island, ACRS Reports of July 19, 1968, May 15, 1968, and January 17, 1968, respectively). The Committee continues to call attention to matters that warrant careful consideration by the manufacturers of all large, water-cooled, power reactors.

The Committee reiterates its belief that the instrumentation design should be reviewed for common failure modes, taking into account the possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single-failure criterion. The applicant should show that the proposed interconnection of control and safety instrumentation will not adversely affect plant safety in a significant manner, considering the possibility of systematic component failure. The Committee believes this matter can be resolved with the Regulatory Staff.

The containment for the reactor is a prestressed concrete vessel similar to previously approved designs (e.g., Rancho Seco), but with modification of the prestressing system design.

The Committee emphasizes the importance of the implementation and management of the quality assurance and quality control programs necessary to achieve the design, construction, and operation objectives.

Inasmuch as a long lead time is required in the training of the operating staff, the Committee emphasizes the need for early training of sufficient personnel to assure adequate operating manpower.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing items, the proposed reactor can be constructed at the Russellville site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Original signed by
Carroll W. Zabel
Carroll W. Zabel
Chairman

References Attached.

References - Russellville Nuclear Unit

1. Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated November 24, 1967.
2. Volume I - Preliminary Safety Analysis Report, Arkansas Power and Light Company Russellville Nuclear Unit, dated November 24, 1967.
3. Volume II - Preliminary Safety Analysis Report, Arkansas Power and Light Company Russellville Nuclear Unit, dated November 24, 1967.
4. Supplement No. 1 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated January 22, 1968.
5. Supplement No. 2 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated February 14, 1968.
6. Supplement No. 3 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated May 3, 1968.
7. Supplement No. 4 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated June 5, 1968.
8. Supplement No. 5 to the Arkansas Power and Light Company Preliminary Safety Analysis Report, dated July 3, 1968.
9. Corrections to Supplement No. 5 to the Arkansas Power and Light Company Preliminary Safety Analysis Report, dated July 10, 1968.
10. Supplement No. 6 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated July 11, 1968.
11. Correction to Supplement No. 6 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated July 15, 1968.
12. Supplement No. 7 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated August 15, 1968.
13. Supplement No. 8 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated August 26, 1968.
14. Supplement No. 9 to Application for Licenses, Arkansas Power and Light Company Russellville Nuclear Unit, dated August 30, 1968.

APPENDIX B

CHRONOLOGY

REGULATORY REVIEW OF THE ARKANSAS POWER AND LIGHT COMPANY

RUSSELLVILLE NUCLEAR UNIT

1.	November 29, 1967	Submittal of Preliminary Safety Analysis Report and License Application.
2.	January 22, 1968	Submittal of Supplemental No. 1, response to AEC General Design Criteria.
3.	January 24, 1968	Meeting with applicant to discuss plans and scheduling of regulatory review.
4.	February 14, 1968	Submittal of Supplement No. 2, design changes in electrical systems and emergency core cooling systems, and data on Dardenell Lock and Dam.
5.	February 28, 1968	Meeting with applicant to discuss areas of the Preliminary Safety Analysis Report that require additional information.
6.	April 3, 1968	Request to applicant for additional information on site, safety analysis, reactor, instrumentation and control, emergency power, engineered safety features quality assurance, training schedules, emergency plants, and initial tests and operations.

7.	May 6, 1968	Request to applicant for additional information on Foundation and Structural Design and miscellaneous other items.
8.	May 3, 1968	Submittal of Supplemental No. 3 in response to April 13, 1968 request for additional information.
9.	May 17, 1968	Meeting with applicant to discuss training schedules and operating staff.
10.	June 5, 1968	Submittal of Supplement No. 4 in response to May 6, 1968 request for additional information.
11.	June 20, 1968	Meeting with the applicant to discuss modified containment design proposed by applicant, site matters and other areas.
12.	July 3, 1968	Submittal of Supplement No. 5, changes in containment design.
13.	July 11, 1968	Submittal of Supplement No. 6, supplemental information in clarification of areas discussed at June 20, 1968 meeting.
14.	August 6, 1968	Meeting with applicant to discuss containment design matters.

15. August 15, 1968 — Submittal of Supplement No. 7, supplemental information in clarification of areas discussed at August 16, 1968 meeting.

16. August 23, 1968 — ACRS Subcommittee meeting and Russellville site visit.

17. August 26, 1968 — Submittal of Supplement No. 8, additional supplementary information in clarification of containment design.

18. August 27, 1968 — Meeting with applicant to discuss quality assurance and quality control plans and organizations.

19. August 30, 1968 — Submittal of Supplement No. 9, documenting information and oral commitments given at August 27 meeting and miscellaneous other items.

20. September 3, 1968 — Meeting with applicant to discuss containment liner anchorage design.

21 September 4, 1968 — ACRS Subcommittee meeting.

22. September 5, 1968 — ACRS meeting.

23. September 6, 1968 — Submittal of Supplement No. 10, updating financial and personnel information and correcting minor errors.

24. September 12, 1968 — ACRS Report issued.

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D.C. 20240

MAY 2 9 1968

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

This is in reply to Mr. Boyd's letter of December 11, 1967, requesting our comments on the application by the Arkansas Power and Light Company for construction permit for the proposed Russellville Nuclear Unit, Pope County, Arkansas, AEC Docket No. 50-313.

The project would be located on a 1,100-acre site on a peninsula at Dardanelle Reservoir, Pope County, Arkansas. A pressurized water reactor would be used as a power source and the plant is designed for an ultimate output of 2,568 thermal (880 gross electrical) Mwt. Cooling and dilution water will be withdrawn from a small inlet embayment west of the plant at a rate of approximately 1,700 c.f.s. and be discharged into the large Illinois Bayou embayment east of the plant, after receiving radioactive and heat wastes. As currently designed, the temperature of the cooling water would be raised approximately 15° at the condenser when the plant is operating at full capacity. The applicant is cooperating with the Fish and Wildlife Service and the Arkansas Game and Fish Commission in the development of an environmental surveillance program.

Dardanelle Reservoir, especially the Illinois Bayou embayment, supports valuable fish and wildlife resources. The large embayment is a productive nursery and harvest area for fish. Waterfowl make extensive use of the reservoir for resting during the migration period. Public and private use facilities on Federal and private land around the embayment are highly developed. Indications are that future development around the embayment will probably result in higher recreational use there than any comparable area of the reservoir. Sport fishing is presently, and will continue to be, one of the chief recreational use attractions in the embayment. Commercial fishing is limited but moderately valuable.

The application indicates that the release of radioactive wastes would not exceed maximum permissible limits prescribed under the Code of Federal Regulations. Although these limits refer to maximum levels of radioactivity that can occur in drinking water for man without resulting in any known harmful effects, operations within these limits may not always guarantee that fish and wildlife will be protected from adverse effects.

If concentrations in receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge. However, radioisotopes of many elements are concentrated and stored by organisms that require these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required, but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of the food chain just as are the nonradioactive elements. These transfers may result in further concentration of radionuclides.

In view of the above, we believe that the environmental monitoring program planned by the applicant should include pre- and post-operational radiological monitoring of selected organisms which require the waste elements or similar elements for their metabolic activities. These surveys should be planned in cooperation with the Fish and Wildlife Service and the appropriate Federal and State agencies.

In view of the extensive sport fishery and the potential value of the commercial fishery in the project area, it is imperative that every possible effort is to be made to protect the valuable resources from radioactive contamination. Therefore, it is recommended that the Arkansas Power and Light Company be required to:

1. Include in their pre-operational environmental surveillance program radiological monitoring of water and sediment samples and of organisms indigenous to the project area that concentrate and store radioactive isotopes. Water and sediment samples should be collected within 500 feet of the reactor effluent outfall site and be measured for gamma radioactivity. Aquatic plants, mollusks, crustaceans and fish should be collected as near as possible to the reactor effluent outfall site and be analyzed for both beta and gamma radioactivity.

2. Prepare a report of pre-operational radiological monitoring and provide five copies to the Secretary of the Interior for evaluation prior to project operation.

3. Continue a radiological monitoring program similar to that specified in recommendation 1 above, analyze the data, and prepare and submit reports every six months during reactor operation or until it is conclusively demonstrated that no significant adverse conditions exist. Five copies of these reports should be submitted to the Fish and Wildlife Service for distribution to the appropriate State and Federal agencies for evaluation.

4. Make modifications in project structures and operations to reduce the discharge of radioactive wastes to acceptable levels if it is determined by the monitoring program that the release of radioactive effluent might result in harmful concentrations of radioactivity in fish and wildlife.

We understand that the Commission's regulatory authority over nuclear power plants involves only those hazards associated with radioactive materials. However, we recommend and urge that before a construction permit is issued, the possibility of thermal and other detrimental effects on fish and wildlife which may result from plant construction and operation be called to the applicant's attention.

We are concerned particularly with the possibility of damages to aquatic life from the heated effluent. Large volumes of heated water discharged into an aquatic environment may not only be detrimental to fish directly, but may also affect these resources indirectly through changes in the environment. The proposed heat load may adversely affect fish habitat and productivity in the Illinois Bayou embayment during the periods (spring and summer) when fish reproduce and have a maximum growth rate. It is likely that the use of the area for spawning will be greatly reduced. It is likely that fish will disperse and avoid the heat-affected area during the maximum temperature months of June through September. Conversely, it is expected that fish will be attracted to the discharge channel and heat-affected area during winter months, resulting in high fisherman-use there.

A General Plan for use of project lands and waters for wildlife conservation and management has been approved for Dardanelle Reservoir by the Secretary of the Army, the Secretary of the Interior, and the Director of the Arkansas Game and Fish Commission. The Russellville Nuclear Unit would occupy land and water covered, in part, by the General Plan. The General Plan provides for a subsequent management agreement between the Department of the Army and the Arkansas Game and Fish Commission. It further provides that the subsequent agreement may make adjustments in the boundaries of the areas shown in the General Plan by the addition or deletion of tracts mutually agreed upon by the parties making the agreement. We understand that the Department of the Army and the Arkansas Game and Fish Commission are now negotiating an agreement pursuant to the General Plan. The Company should be made aware of these documents and plan its operations so that they are in accordance with the Arkansas Game and Fish Commission's fish and wildlife management plan for the reservoir.

The applicant has given assurance that additional studies will be carried out, and has to date cooperated fully with the Fish and Wildlife Service and the Arkansas Game and Fish Commission in discussing and developing plans for the protection of fish and wildlife in the area. This study

4. Make modifications in project structures and operations to reduce the discharge of radioactive wastes to acceptable levels if it is determined by the monitoring program that the release of radioactive effluent might result in harmful concentrations of radioactivity in fish and wildlife.

We understand that the Commission's regulatory authority over nuclear power plants involves only those hazards associated with radioactive materials. However, we recommend and urge that before a construction permit is issued, the possibility of thermal and other detrimental effects on fish and wildlife which may result from plant construction and operation be called to the applicant's attention.

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The applicant has given assurance that additional studies will be carried out, and has to date cooperated fully with the Fish and Wildlife Service and the Arkansas Game and Fish Commission in discussing and developing plans for the protection of fish and wildlife in the area. This study program should complement the radiological monitoring program recommended above, should be designed to measure habitat changes in the affected area of Dardanelle Reservoir, and should be carried out prior to and during plant operation, so that comparative data will be available for analysis.

In view of the above, we recommend that the Atomic Energy Commission urge the Arkansas Power and Light Company to:

1. Continue to cooperate with the Fish and Wildlife Service, Arkansas Game and Fish Commission, and other interested Federal and State agencies in developing plans for ecological surveys, initiate these studies at least two years before reactor operation, and continue them during project operation on a regular basis or until it has been conclusively demonstrated that no significant adverse conditions exist.

2. Meet with the above-mentioned Federal and State agencies at frequent intervals to discuss new plans and to evaluate results of the ecological surveys.

3. Make such modifications in plant structures and operations, including but not limited to facilities for cooling discharge waters, as may be determined necessary to protect the fish and wildlife resources of the area.

The opportunity to present our views is appreciated.

Sincerely yours,

Clarence F. Pautzke
Commissioner

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D.C. 20240

AUG 29 1968

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

This is in response to Mr. Boyd's letter of July 16 transmitting Amendment No. 6, dated July 11, 1968, to the application by Arkansas Power and Light Company for a construction permit for the proposed Russellville Nuclear Unit, Pope County, Arkansas, Docket No. 50-313.

Modification of project plans to reverse the direction of cooling water flow through the project would not alter overall effects of the project on fish and wildlife significantly. The recommendations contained in our letter of May 29 are still applicable.

Thank you for the opportunity for comments on Amendment No. 6.

Sincerely yours,

Clarence F. Pautzke
Commissioner

UNITED STATES GOVERNMENT

Memorandum

TO : Peter A. Morris, Director
Division of Reactor Licensing

DATE: JAN 22 1968

FROM : Milton Shaw, Director
Division of Reactor Development & Technology

SUBJECT: SAFETY ANALYSIS REPORTS

RDT:NS:S349

Reference is made to the letters of November 22, 1967, December 11, 1967, and December 26, 1967, from the Division of Reactor Licensing, to the Environmental Science Services Administration requesting comments on the following safety analysis reports respectively:

Rancho Seco Nuclear Generating Station Unit No. 1
Sacramento Municipal Utility District
Preliminary Safety Analysis Report
Volumes I, II, III and IV dated November 1967

Russellville Nuclear Unit
Arkansas Power and Light
Preliminary Safety Analysis Report
Volumes I and II dated November 29, 1967

Donald C. Cook Nuclear Plant
Indiana and Michigan Electric Company
Preliminary Safety Analysis Report
Volumes I, II and III dated December 18, 1967

Review by the Environmental Meteorology Branch, Air Resources Laboratory, ESSA, has now been completed and their comments are attached.

Attachments:
Three Sets of Comments (Orig. & 1 copy)

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA FPMR (41 CFR) 101-11.6

UNITED STATES GOVERNMENT

Memorandum

TO : Peter A. Morris, Director
Division of Reactor Licensing

DATE: JAN 28 1968

FROM : Milton Shaw, Director
Division of Reactor Development & Technology

SUBJECT: SAFETY ANALYSIS REPORTS

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Donald C. Cook Nuclear Plant
Indiana and Michigan Electric Company
Preliminary Safety Analysis Report
Volumes I, II and III dated December 18, 1967

Review by the Environmental Meteorology Branch, Air Resources Laboratory, ESSA, has now been completed and their comments are attached.

Attachments:
Three Sets of Comments (Orig. & 1 copy)

APPENDIX D

Comments on

Russellville Nuclear Unit
Arkansas Power and Light
Preliminary Safety Analysis Report
Volumes I and II dated November 29, 1967

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
January 10, 1968

The analysis of the Fort Smith and Little Rock meteorological data indicates that a continental diffusion climate can be expected at the Russellville site. This means a pronounced difference between daytime and nighttime atmospheric diffusion rates, with the lower wind speeds and slower diffusion occurring at night. The predominant daytime wind direction for the general area would be from the southwest as shown by the Little Rock wind rose. Nighttime wind directions with inversion conditions will most likely be towards the Dardanelle Reservoir of the Arkansas River.

The analysis of the Little Rock hourly weather reports with regard to diffusion types shows an average frequency of about 35% for Pasquill F condition during the four months considered (see Table 2A.15). The annual nighttime wind speeds were less than 3 knots about 20% of the time at Little Rock (see Table 2A.6). On this basis, it would seem appropriately conservative to use inversion diffusion conditions (Type F) and a 1 m/sec wind speed to compute the initial two-hour average concentration. This would result in a concentration of 6.4×10^{-4} sec m^{-3} at the site boundary assuming a ground source with no credit for building-induced dilution. Taking credit for the building effect as determined empirically in tests at the National Reactor Testing Station would result in a concentration value of about 2×10^{-4}, which agrees with the applicant's value.

The analysis of the persistence of a diffusion condition in a unidirectional flow (Tables 2A.17 and 18) shows that no cases persisted longer than 10 hours. Consequently, for the 24-hour average concentration it would be conservative to assume inversion conditions, a 2 m/sec wind with concentrations averaged over a 22 1/2 degree arc. At the site boundary this would result in an average concentration of 7×10^{-5} sec m^{-3}, which is in reasonable agreement with the applicant's computation.

In summary, a reasonable, conservative analysis has been made of the atmospheric diffusion conditions of the Russellville site which provides a sound basis for a preliminary safety evaluation of the proposed nuclear plant.

APPENDIX E

DOCKET NO. 50-313

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

AUG 16 1968

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by Mr. Roger S. Boyd is a review of geologic and hydrologic aspects of the site for the Russelville Nuclear Station proposed by the Arkansas Power and Light Company.

The review was prepared by H. H. Waldron and E.L. Meyer and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

Arthur A. Baker

Acting Director

Enclosure

Russelville Nuclear Unit
Pope County, Arkansas
AEC Docket 50-313

Hydrology

The site is located on the left bank of the Arkansas River 6 miles upstream from Dardanelle Lock and Dam No. 10. The plant site grade at 353 feet msl (above mean sea level) is 15 feet above the normal operating level of Dardanelle Reservoir.

Flood stages in the pool of Dardanelle Reservoir for a computed maximum probable flood of 1,500,000 cfs (cubic feet per second) have been given by the Corps of Engineers as 353 feet msl at Dardanelle Dam and 389.5 feet msl at the upstream end of the reservoir. The applicant's estimate of 358 feet msl for the stage of such a flood at the site appears reasonable. The failure of Ozark Dam about 46 miles upstream from the site during such a flood could cause an additional rise in stage. The head differential across Ozark Dam during a maximum probable flood as computed by the Corps of Engineers would be 11.5 feet, and on that basis the applicant has estimated an additional 3 feet rise at the site resulting in a stage of 361 feet. This appears to be reasonable.

At a stage of 361 feet the site grade would be overtopped by 8 feet and the reactor structures would be surrounded by water. A certain amount of wave action may then be expected and should be reflected in the level of flood protection chosen for essential equipment.

The cooling water requirements of the reactor are given as 1,700 cfs (cubic feet per second). Flow of the Arkansas River has been measured at a gage at Dardanelle 6 miles downstream from the site. Average flow during 1937-66 was 34,920 cfs; minimum flow was 416 cfs, and the lowest mean monthly flow was 592 cfs in October 1956. Low flow occurs generally in late summer and fall.

Geology

The analysis of the geology of the Russellville Nuclear Generating Plant in Arkansas, as presented in AEC Docket No. 50-313 and supplements, was reviewed and compared with the available literature. The analysis appears to be carefully derived and to present an adequate appraisal of those aspects of the geology that would be pertinent to an engineering evaluation of the safety of the site.

There are no identifiable active faults or other recent geologic structure that could be expected to localize earthquakes in the immediate vicinity of the site.

Tectonically the site is located near the axis of the Scranton syncline, one of several westward-trending, gentle folds that characterize the

(59)

Arkoma Basin--a major structural and topographic feature of Arkansas and eastern Oklahoma that developed in late Paleozoic time. Although several ancient faults are associated with the Arkoma Basin folded structures in the area, none of these appears to have been tectonically active since latest Paleozoic time.

The limited subsurface data available indicate that the major units of the nuclear facility will be founded on a hard, dense shale (the McAlester Formation), which should provide an adequate foundation for the proposed structures.

APPENDIX F

U.S. DEPARTMENT OF COMMERCE
ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
COAST AND GEODETIC SURVEY
ROCKVILLE, MD. 20852

AUG 15 1968

IN REPLY REFER TO: C23

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of Russellville, Arkansas, and vicinity. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismic activity of the area as presented by the Arkansas Power and Light Company in the "Preliminary Safety Analysis Report," and we are now submitting our conclusions on the seismicity factors.

If we may be of further assistance to you, please do not hesitate to contact us.

Sincerely yours,

James C. Tison, Jr.
Rear Admiral, USESSA
Director

Enclosure

2960

REPORT ON THE SITE SEISMICITY FOR THE RUSSELLVILLE NUCLEAR UNIT, ARKANSAS

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has examined the seismicity of the area around the proposed site near Russellville, Arkansas, and has examined a similar analysis made by the applicant, the Arkansas Power and Light Company in the "Preliminary Safety Analysis Report." The applicant's report is satisfactory for an evaluation of the seismic factor of the site.

Based upon the review of the seismic history of the site and the surrounding area and the related geologic conditions, the Coast and Geodetic Survey agrees with the applicant that an acceleration of 0.10 g on good foundation would be adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. In addition, the Survey agrees with the applicant that the acceleration of 0.20 g would represent the ground motion from the maximum earthquake likely to affect this site. We believe this value would provide an adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

August 14, 1968

APPENDIX G

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

19 August 1968

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
The Russellville Nuclear Unit, Arkansas Power and Light Company
(AEC Docket No. 50-313)

Dear Dr. Morris:

We are transmitting herewith two copies of our report entitled "Adequacy of the Structural Criteria for the Russellville Nuclear Unit," prepared by Drs. W. J. Hall, W. H. Walker and myself.

Sincerely yours,

N M Newmark

N. M. Newmark

mlw
cc: W. J. Hall
W. H. Walker
Enclosure

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO AEC REGULATORY STAFF

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE RUSSELLVILLE NUCLEAR UNIT

ARKANSAS POWER AND LIGHT COMPANY

(AEC Docket No. 50-313)

by

N. M. Newmark, W. J. Hall and W. H. Walker

August 1968

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE RUSSELLVILLE NUCLEAR UNIT

Arkansas Power and Light Company

by

N. M. Newmark, W. J. Hall and W. H. Walker

INTRODUCTION

This report is concerned with the adequacy of the containment structures and components for the Russellville Nuclear Unit for which application for a construction permit has been made to the U. S. Atomic Energy Commission by the Arkansas Power and Light Company. The facility is located on a peninsula in the Dardanelle Reservoir, Arkansas River, Pope County, Arkansas, about 6 miles WNW of Russellville, and 2 miles SE of London, Arkansas.

Specifically this report is concerned with the design criteria that determine the ability of the containment system and Class I equipment and piping as well as Class II structures and equipment, to withstand an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration simultaneously with the other loads forming the basis of the design. The facility also is to be designed to withstand a Design Basis Earthquake of 0.20g maximum horizontal ground acceleration to the extent of ensuring safe shutdown and containment.

This report is based on information and criteria set forth in the Preliminary Safety Analysis Reports (PSAR) and supplements thereto listed at the end of this report. Also, we have participated in discussions with the applicant and the AEC Regulatory Staff concerning the design of this unit.

DESCRIPTION OF FACILITY

The Russellville Nuclear Unit is described in the PSAR as consisting of a pressurized-water type reactor employing two closed cooling loops connected in parallel to the reactor vessel. The system is arranged as two heat transport loops, each with two circulating pumps and one steam generator; one of the loops contains an electrically heated pressurizer. The nuclear steam supply system will be furnished by the Babcock and Wilcox Company, and the turbine generator is to be supplied by the Westinghouse Electric Corporation. The plant is to be designed for a power level of 2452 MWt (850 MWe).

The reactor containment structure is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system of horizontal and vertical tendons. The dome is post-tensioned using a 3-way system. The hoop tendons are to be placed in three 240° systems using three buttresses as anchorages, with the tendons staggered so that half of the tendons at each buttress terminate at that buttress. The foundation slab is conventionally reinforced with high-strength reinforcing steel.

The cylinder has an internal diameter of 116 ft. and an inside height of 206 ft. The distance from the top of the foundation slab to the springline of the domed roof is approximately 166 ft. The vertical wall thickness is noted to be 3 ft. - 9 in. and the dome thickness, 3 ft. - 3 in. The foundation slab thickness is about 9 ft.

For prestressing, the applicant proposes to use 90 to 184 wire tendons, unbonded. The discussion presented in the PSAR suggests that the BBRV type anchorage system will be employed, although the PSAR notes that other prestressing systems will continue to be studied. The prestressing tendons will be protected against corrosion by a pressure-injected casing filler. The liner plate will conform to specification ASTM-A442, Grade 60, and will be 1/4 in. in thickness. The reinforcing steel in the base slab of the containment structure will conform to ASTM designation A432-65; this steel possesses a minimum yield strength of 60,000 psi. Splices in bars larger than No. 11 will be made by the Cadweld method.

The design of the containment structure for this facility is essentially similar to that employed for the Rancho Seco Nuclear Generating Station Unit No. 1.

The geological description of the site indicates a stiff clay and silty clay of 13 to 23 foot thickness overlying hard and dense horizontally bedded shale of the Pennsylvanian McAlester formation. All major structures of the facility will be founded on the underlying McAlester formation shale bedrock. No active or recent faulting has been mapped in the area of the proposed site. The closest known faults are the London and Perry View faults located 5 or 6 miles from the site.

SOURCES OF STRESSES IN CONTAINMENT STRUCTURES IN CLASS I COMPONENTS

The reactor containment structure is to be designed for the following loadings and conditions: dead load; live load (including snow and equipment loads); prestressed loadings; design accident temperature of about 285°F and pressure of 59 psig; an air test pressure of 115 percent of the design pressure; an external pressure loading with a differential of approximately 2½ psi from outside to inside; wind loading corresponding to 80 mph basic wind at 30 ft. above grade; buoyancy loadings; tornado loading associated with a 300 mph tangential wind velocity and a 40 mph forward progression velocity, including a differential pressure of 3 psi from inside to outside with associated missiles; and earthquake loading as described next.

The seismic design is to be made for an Operating Basis Earthquake based upon a maximum horizontal ground acceleration of 0.10g and a Design Basis Earthquake based upon a maximum horizontal ground acceleration of 0.20g.

The containment walls and liner are shielded by various types of barriers from impact from missiles which possibly could have enough energy to strike or penetrate them. The high-pressure reactor cooling system equipment which could be the source of missiles is screened either by the containment shield wall enclosing the reactor cooling loops, by the concrete operating floor, or by a special missile shield to block any passage of missile to the containment walls.

The general criteria controlling the design of piping and reactor internals f.r seismic loadings are presented in various places in the PSAR.

COMMENTS ON ADEQUACY OF DESIGN

Foundations and Dams

The major facility structures are to be founded directly on competent bedrock, and on the basis of the information presented in the PSAR and amendments, the foundation conditions appear acceptable to us.

The Dardanelle Reservoir from which the plant will draw its cooling waters is discussed in several places in the PSAR and particularly in Appendix 2F and in the answers to Questions 2.7 and 2.8 of Supplement No. 3. The analysis of the Dardanelle Lock and Dam as reported in Appendix 2F suggests that some damage to the L.ck and Dam facility might be expected. Thus, the applicant notes in the answer to Question 2.7 that emergency shutdown cooling water will be supplied from an emergency reservoir to be located northwest of the plant site. The emergency reservoir will be excavated in impervious cl.y and will have an effective storage capacity of about 35 acre feet. We concur in this approach for an assured source of cooling water in view of the possible effects of an earthquake on the Dardanelle Lock and Dam.

The effect of a flood on the structure is discussed in the answer to Question 2.8 of Supplement No. 3. It is noted there that the plant grade level is elevation 353 ft. and the maximum elevation of a flood is estimated to be 361 ft. The applicant indicates that the early forecast of a severe flood of this type would provide ample time for precautionary measures in terms of plant shutdown. All Class I equipment is either located above maximum probable flood level or protected by waterproof Class I structures which are designed for buoyancy effects.

Gas Pipeline

In the answer to Question 2.11 of Supplement No. 3, there appears a discussion of the natural gas transmission pipeline which crosses the discharge water channel. It is indicated in the answer to that question that the existing pipeline crossing will be re-layed beneath the water channel with 4 ft. of earth cover. We understand that it will be possible to valve off this section of line in the event of difficulty. It is noted that the pipeline will be at its closest about 400 ft. from the intake structure and 600 ft. from the containment structure. These distances are sufficient, we believe, to preclude any serious consequences with regard to plant safety in the event of a pipe rupture.

Seismic Design and Criteria

We are in agreement with the earthquake loading criteria selected for the seismic design, namely that associated with an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration and a Design Basis Earthquake of 0.20g maximum horizontal ground acceleration. These earthquake design criteria are in agreement with those given by the U. S. Coast and Geodetic Survey (Ref. 2).

The response spectra for the Operating Basis Earthquake and Design Basis Earthquake to be employed in the dynamic analysis are presented as Fig. 5A-1 and 5A-2 of Appendix 5A of the PSAR. These spectra are scaled after those presented in publications by Dr. G. W. Housner, and we concur in their use.

The earthquake analysis will include the effects of vertical earthquake excitation which will be taken as 2/3 of the horizontal component as noted on page 5-3 of the PSAR. It is noted in the answer to Question 12.3.6 that the effects of vertical and horizontal earthquake motions will be combined linearly and directly with each other and with the other applicable stresses. We are in agreement with these design criteria.

The percentage of critical damping to be employed in the analysis is listed on page 5-A-5 of the PSAR, and we are in agreement with the values given there.

The method of dynamic analysis is described in Section 5.1.5.6 of the PSAR. The method of analysis is not described in enough detail to evaluate it completely; however, it would be our recommendation that a standard modal analysis procedure be employed to take account of structural rocking, lateral translation, and the shearing and flexural distortion of the structure. With proper attention to damping and coupling of the various modes, it should be possible to arrive at reasonable and consistent values of direct stress, shear, moment, etc.

The loading combinations to be employed for the design of the containment structure are given in Section 5.1.4 of the PSAR. The loading combination expressions given appear acceptable to us, and it is noted that for these load factor combinations, the resistance will be less than the yield strength of the structure. We concur in this approach.

eb1

MR. ENGELHARDT: Mr. Chairman, that completes the direct testimony of the AEC regulatory staff, with the exception of the responses that have been prepared to the questions of the Board which were raised in the pre-hearing conference.

CHAIRMAN WELLS: Yes. I would like to have a brief moment to confer with the other members of the Board.

(The Board conferred.)

CHAIRMAN WELLS: Mr. Engelhardt, I do not remember. At the time you introduced the Joint Exhibit did you call attention to the fact that you have the qualifications of the members of the ACRS?

MR. ENGELHARDT: No, sir, I neglected to offer two exhibits that the staff intends to offer and I think this would be an appropriate time to do this now.

CHAIRMAN WELLS: If it is convenient, will you please proceed?

MR. ENGELHARDT: Yes. I would like to offer for identification the statement of educational and professional qualifications of the 15 members of the Advisory Committee on Reactor Safeguards, the group which we have identified previously in our opening statement.

These 15 members participated in the review of the application submitted by Arkansas Power and Light Company for a construction permit.

This offer of Staff Exhibit No. 1, Statement of

eb2

Professional Qualifications of the ACRS, is now being distributed

(Documents being distributed.)

MR. ENGELHARDT: I would now like to request that Staff Exhibit No. 1 be made a part of the record of this proceeding.

CHAIRMAN WELLS: It is so ordered.

(The document was marked
Staff Exhibit No. 1
for identification and
received in evidence.)

ebl

MR. ENGELHARDT: I would now like to offer for identification Staff Exhibit No. 2, which consists of a statement of the educational and professional qualifications of the 15 members of the AEC regulatory staff, other than the witnesses who are present today, who participated in the review of the application submitted by Arkansas Power and Light Company.

(Documents being distributed.)

MR. ENGELHARDT: I would request that this Staff Exhibit 2 be incorporated into the record of this proceeding.

CHAIRMAN WELLS: It is agreed.

(The document was marked Staff Exhibit No. 2 for identification and received in evidence.)

ebl

MR. ENGELHARDT: I think that that now completes the direct testimony and presentation of exhibits at this time for the staff.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

Now let me inquire if the staff desires to cross-examine any of the witnesses presented by the applicant?

MR. ENGELHARDT: With regard to the testimony presented up to now by the applicant, the staff has no cross-examination questions.

I want to indicate that, as I stated previously, the staff has had somewhat over approximately over one year to review the application which has been submitted by the applicant. The testimony presented here today is essentially a summary of that application. Consequently, any questions that the staff had of the applicant have long since been raised and responded to and are reflected -- at least the answers to our questions are now appropriately reflected in the application as it now stands before this Board.

And thus at this time we have no further questions to raise of the applicant.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

I suppose theoretically it is conceivable that the applicant might cross-examine the staff. Mr. Jewell, may I assume that you do not?

MR. JEWELL: We do not even theoretically.

eb2 (Laughter.)

CHAIRMAN WELLS: Well, then, gentlemen, I believe that we have covered Item 16 of the agenda as well as Items 14 and 15. We have covered Item 17 and now we reach questioning of the witnesses by the Board members, if I am following my agenda correctly.

In connection with this item, let me note that the Board received last night answers from the applicant and from the staff to two questions -- well, answers from the applicant and the staff on one question, and answer by the applicant on another question which the Board raised in the pre-hearing conference.

Do I assume that you are going to introduce that at this time for the record?

MR. JEWELL: Mr. Chairman, those were answers to, in our case, only two of the questions. We are prepared to introduce those and also answer the other questions that were

10 asked by the Board at the pre-hearing conference.

BLOOM 11
[illegible] 1

CHAIRMAN WELLS: Let me suggest that you introduce those documents now if you will and then we will proceed to the other questions.

You may wish to later on summarize them orally, and the Board would be glad to hear your oral summary, but at this point let us just have them introduced.

MR. JEWELL: Mr. Holmes, the Board has stated its interest in the general subject of containment spray system for removing iodine including your evaluation of the degree of conservatism in the staff's calculation of reduction factor, the details of additional R&D work that is required, who will do the work, the schedule for accomplishment of work and problems that might cause the spray system to prove inadequate.

Have you prepared a statement in response to this general question by the Board?

MR. HOLMES: Yes I have.

MR. JEWELL: Is your answer to the Board's question contained in a document which is entitled, "Applicant's Response to the Board's Question on Iodine Removal?"

MR. HOLMES: Yes, sir, it is.

MR. JEWELL: Mr. Holmes, was this document prepared under your supervision?

MR. HOLMES: Yes, it was.

MR. JEWELL: Are all of the facts set forth in the document true and correct?

MR. HOLMES: Yes, they are.

MR. JEWELL: Do you adopt this document as part of your testimony in this case?

MR. HOLMES: Yes.

MR. JEWELL: Mr. Chairman, I would like to request that the document which Mr. Holmes has described, a copy of which was given to the members of the Board last evening and a copy of which was given to the staff last evening -- I would like to request that this document be incorporated into the transcript of the testimony and to be treated as the testimony of Mr. Holmes to the same extent as if it were here read.

CHAIRMAN WELLS: It is so ordered with the understanding that Mr. Holmes may wish to make oral statements and the Board may wish to ask questions about it.

Thank you.

(The document follows.)

In the matter of	)	
	)	
ARKANSAS POWER AND LIGHT COMPANY	)	Docket No. 50-313
	)	
(Russellville Nuclear Unit)	)	

Applicants response to the Board's questions on Iodine Removal.

"alkaline sodium thiosulfate" (or sometimes just thiosulfate) and "sodium hydroxide" (or sometimes just hydroxide).

Some experiments referred to in past cases have been performed on solutions containing only boric acid and sodium thiosulfate but no sodium hydroxide. This solution has an acid pH, is unstable in several respects, and is generally unacceptable. The "acid sodium thiosulfate" solution should not be confused with the "alkaline sodium thiosulfate" solution since they have different chemical characteristics. Only "alkaline sodium thiosulfate" has been considered for use in the Russellville chemical spray system.

Both sodium thiosulfate and sodium hydroxide are known to react rapidly with iodine and, in fact, were suggested by Griffiths[1] in his 1963 study on the use of sprays for removing iodine from containment atmospheres. Both "alkaline sodium thiosulfate" and "sodium hydroxide" have certain merits and certain sensitivities when their performance is evaluated in the post-accident environment.

Although both solutions contain sodium hydroxide, its function is completely different in the two solutions. The thiosulfate in the "alkaline sodium thiosulfate" solution reacts rapidly and completely with the iodine. The sodium hydroxide in the "alkaline sodium thiosulfate" is not intended to play an active role in the absorption of the iodine. Its function is merely to preserve the long-term stability of the thiosulfate solution by maintaining the desired pH.

On the other hand, the hydroxide in the "sodium hydroxide" solution plays an active role in absorbing the iodine. The reaction between iodine and hydroxide solution is generally classified as a hydrolysis reaction. This reaction can yield several different ionic forms of the absorbed iodine, and under certain conditions it can be considered reversible.

The merit of "alkaline sodium thiosulfate" solution is that it reacts rapidly, completely, and irreversibly with the iodine; whereas with "sodium hydroxide" solution care must be taken to guard against conditions that might tend to reverse the reaction. On the other hand, because of the sulfur compound in the "alkaline sodium thiosulfate," care must be taken to assure satisfactory performance when exposed to thermal and radiation conditions in the accident environment. "Sodium hydroxide" is generally immune to direct damage from radiation and thermal effects.

Data presently available(2 - 7) indicate that either solution is satisfactory when used in a properly engineered system.

II. IODINE REMOVAL EFFECTIVENESS

In the past year, ORNL has conducted a number of spray tests in the Nuclear Safety Pilot Plant (NSPP) facility. These tests, as reported in ORNL-4253,(7) have demonstrated that elemental radioactive iodine is rapidly removed by chemical sprays.

Using an NSPP run made at accident conditions closely approximating those predicted for Russellville, the measured iodine half life was 31 sec; that is, half of the radioactive iodine was removed from the steam-air atmosphere in 31 sec after starting the sprays. These data have been scaled to the Russellville design. They result in an iodine half life of 23 sec with the full spray installed capacity operating and a half life of 46 sec at half capacity. The iodine half life reported in the PSAR is 90 sec at full capacity and 180 sec at half capacity.

On the basis of calculations presented in chapter 14 of the PSAR, the iodine removal half life required to reduce the 2-hour thyroid dose at the exclusion distance to the 300 rem limits of 10 CFR 100 is 1410 sec. Thus, only about 1/15 of the available spray effectiveness reported in the PSAR and only 1/60 of the available effectiveness as indicated by NSPP tests is required to meet the 10 CFR 100 site acceptability requirements.

A large number of confirmatory tests(7 - 12) have been made which demonstrate that chemical sprays are effective for iodine removal. These spray tests have been made using a wide range of variables--spray distributions with droplet sizes ranging from 100 to 1200 microns, fall heights ranging from a few feet to approximately 50 ft, temperature and pressure conditions varying from ambient to maximum accident conditions, iodine concentrations ranging from 1 to 130 mg/cubic meter, single and multiple spray nozzle installations, spray fluxes ranging from 0.007 to 0.2 gpm per square foot vessel cross section, and condensing and non-condensing conditions. With this wide range of confirmatory test conditions, we are confident that the Russellville Nuclear unit chemical spray system will perform as predicted.

III. NON-REACTIVE IODINE

The non-reactive iodine consists primarily of methyl iodide but includes a small, almost insignificant, fraction of other organic iodides and particulate aerosols.

Experimental data obtained under a wide variety of conditions on the amount of methyl iodide released from overheated fuel are reported in numerous publications. Iodine release experiments using irradiated Zircaloy-clad UO_2 fuel in a PWR accident environment show less than the 5% non-removable iodine assumed in the PSAR. Six tests[13] were performed at Battelle-Northwest Laboratories. They found that 1% or less of airborne iodine was in the non-removable form. Thirteen other experiments were performed in England;[14] all but two show less than 0.2% as methyl iodide. The highest result was 3%. There are a number of other experiments reported in the literature which deal with the amount of non-reactive iodine released from over-heated fuel. Some of these experiments have observed greater than 5% non-removable iodine; however, these experiments were conducted under conditions that are not applicable to the PWR accident environment. It is on the basis of all the data above that we concluded that 5% non-removable iodine was a conservative value for use in the accident analysis.

Spray tests at Oak Ridge National Laboratory[7,15] have demonstrated that alkaline sodium thiosulfate spray is effective for removal of methyl iodide. While the removal rate is not as dramatic as that for elemental iodine, the methyl iodine removal rate is sufficient to make a significant reduction in the airborne iodine concentration and thus in the off-site doses. Even though the chemical spray will remove methyl iodine, to be conservative, we have not taken credit for this removal in our accident evaluation.

On page 29 of the Staff Safety Evaluation, they state that their dose reduction factors "assume as much as 10% of the iodine in the containment is in nonremovable (organic) form." Even using this more conservative assumption, there is a large margin between the required and attainable iodine removal half life. With 10% non-reactive iodine, a half life of 1250 sec is necessary to meet the limits of 10 CFR 100. This is considerably greater than the 90 sec half life reported in the PSAR or the 23 sec half life indicated by scaling NSPP data to the Russellville plant.

IV. SPRAY SOLUTION STABILITY

The answer to Question 5.4 in Supplement No. 3 states the requirements for solution stability as:

a. The components of the solution must remain chemically and physically compatible.

b. The solution must retain adequate capacity for iodine removal and retention.

c. The decomposition products must not result in excessive pH changes, excessive amounts of solid precipitates or excessive gas formation, or in any way reduce the concentration of the soluble poison dissolved in the solution.

The specific conditions under which this stability must be exhibited correspond to those for an MHA, which are:

a. Thermal exposure to temperatures of between 250-300°F for 15 minutes, 200-250°F for 15 minutes, 150-205°F for 1 day, and 100-140°F thereafter.

b. Radiation exposure to doses of about 1×10^8 rads in 20 days, 2×10^8 rads in 80 days, and 3×10^8 rads in over 300 days following an MHA.

Test results currently available from ORNL, which are summarized in the following paragraphs, indicate that alkaline sodium thiosulfate will satisfy the stability requirements.

A series of thermal stability experiments was performed at Oak Ridge National Laboratory.[3] In these experiments, the alkaline sodium thiosulfate solutions--the proposed spray solution--was exposed to temperatures of 185, 248, and 285°F for periods up to about 5 days. The results of the tests indicated that the respective solutions approached equilibrium while retaining about 90, 70, and 64 percent of their iodine absorbing capacity. Since the spray solution is only above 248°F for about 15 minutes and above 185°F for less than 6 hours, it is estimated that the spray solution in the reactor building would reach an equilibrium while retaining about 90% of its iodine absorbing capacity due to thermal decomposition.

ORNL has also performed a series of irradiation experiments using alkaline sodium thiosulfate.[2,6] These irradiations were performed at room temperature and indicated that after absorbing about 1×10^8 rads, which corresponds to about 20 days after the MHA, the solution retains about 57 percent of its iodine absorbing capacity.

ORNL has also performed a series of heated irradiations to demonstrate the combined effect of temperature and irradiation.[5] The irradiations were performed at 185, 248, and 285°F and to total doses of 1 x 10^8 rads. At all three temperatures the rate of decomposition decreased as the dose increased. At 1 x 10^8 rads the respective solutions r‑ ‑in 30, 28 and 17 percent of their iodine absorbing capacity.

The solution in the reactor building will be below 185°F after about the first 6 hours, so that most of the irradiation will be performed between 100-140°F. Based on this fact and the data above, we estimate that the alkaline thiosulfate in the reactor building will retain about 40% of its iodine absorbing capacity after absorbing 1 x 10^8 rads in about 20 days. This reduction in the solution's iodine absorption capacity should have no effect on the performance of the spray solution because of the several hundred-fold excess of thiosulfate initially present in the solution.

These results demonstrate that even after 20 days the spray solution retains its ability to absorb iodine. However, any iodine release, following a loss-of-coolant accident would occur shortly after the accident (within minutes). Therefore, the ability to remove iodine from the reactor building atmosphere exists long after the release period.

As long as the solution retains the ability to remove iodine, none of the iodine in the solution can be released. However, the solution's ability to retain iodine is even greater than its ability to absorb iodine.

The 10 to 20 pounds of iodine assumed to be released during an MHA is reduced to the nonvolatile iodide form when absorbed by the alkaline sodium thiosulfate solution. Initial results from experiments being conducted by B&W under simulated accident conditions have indicated that, once reduced to the iodide form, the volatile iodine will not re-evolve from the solution even after all the thiosulfate's capacity for additional iodine has been lost, thus indicating the solution's ability to retain the absorbed iodine indefinitely.

As a result of radioactive decay, it is only necessary to retain this capacity for a limited time. For example, if at the end of 60 days all of the remaining iodine were released from the solution, then the low population zone dose would increase by less than 10% of that predicted in the PSAR.

All of the ORNL experiments cited above contain the proper amount of boric acid, and no compatibility problem is evidenced. In addition, no colloidal sulfur or other solids formed as long as the pH remained sufficiently alkaline. These results demonstrate that although the alkaline sodium thiosulfate solution undergoes some radiolytic decomposition, the products of this decomposition do not adversely affect the ability of the solution to perform as the coolant in either the Emergency Core Cooling Systems or the Reactor Building Spray System.

In order to confirm these conclusions, Babcock & Wilcox is conducting an R&D Program on the radiation and thermal stability of the chemical spray solution under conditions simulating the post-accident period. These experiments will measure the radiolytic decomposition of the spray solution when irradiated at temperatures and pressures that conservatively approximate the accident conditions, and irradiated to doses in excess of those expected to be absorbed by the solution during the accident period. The thermal stability experiments will measure the amount of thermal decomposition and the scaling tendencies of the spray solution when exposed to heated Zircaloy cladding under temperature and heat flux conditions simulating those expected during the accident.

As indicated in response to Question 1.3 of Arkansas Power and Light PSAR, Supplement No. 3, the B&W R&D Program is described in more detail in answer to Question 17.4 of Metropolitan Edison PSAR (Docket 50-289) Supplement 3. The results of this program are scheduled to be available in early 1969.

V. MATERIALS COMPATIBILITY WITH SPRAY SOLUTION

The alkaline sodium thiosulfate spray solution should not react significantly with the materials used in the reactor building, the reactor system, or the engineered safeguards systems. The reactor building surfaces are painted carbon steel and concrete. Reactor system surfaces are Zircaloy, stainless steel, and Inconel. The recirculation system and the spray system are stainless steel. All these materials must be resistant to the alkaline sodium thiosulfate solution.

In order to demonstrate that the materials to be used in construction of the plant are compatible with the spray solution in the accident environment, Babcock & Wilcox is conducting an R&D Program. This program is explained in more detail in answer to Question 17.4 of the Metropolitan Edison PSAR (Docket 50-289) Supplement 3. Specimens of stainless steel, Zircaloy, Inconel, carbon steel, and concrete, and specimens of the paint and the coatings used inside the reactor building are being checked for chemical attack and corrosion after exposure to the alkaline sodium thiosulfate solution. The tests are being conducted under temperature versus time conditions which correspond to those of the post-accident period.

All the materials mentioned above are under test at the present time, and the results to date have indicated acceptable performance. The testing will be complete in early 1969.

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2. Cottrell, Wm. B., ORNL Nuclear Safety Research and Development Program, Bimonthly Report for July - August 1967, ORNL-TM-1986, p 35.

3. Cottrell, Wm. B., ORNL Nuclear Safety Research and Development Program, Bimonthly Report for March - April 1968, ORNL-TM-2230, p 81.

4. Cottrell, Wm. B., ORNL Nuclear Safety Research and Development Program, Bimonthly Report for May - June 1968, ORNL-TM-2283, pp 64 - 73.

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6. Cottrell, Wm. B., ORNL Nuclear Safety Research and Development Program, Annual Report for 1967, ORNL-4228.

7. Parsly, L. F. and Franzreb, J. K., Removal of Iodine Vapor from Air and Steam-Air Atmospheres in the Nuclear Safety Pilot Plant by Use of Sprays, ORNL-4253, June 1968.

8. Nuclear Safety Quarterly Report; November, December, 1967, January 1968; for Nuclear Saf[illegible] Branch of USAEC Division of Reactor Development and Technology, b[illegible]f of Battelle-Northwest, BNWL 816.

9. Nuclear Safety Quarterly Report; February, March, April 1968; for Nuclear Safety Branch of USAEC Division of Reactor Development and Technology, by the Staff of Battelle-Northwest, BNWL 88[illegible]

10. Parker, G. W., Reaction of Molecular Iodine and of Methyl Iodide With Sodium Thiosulfate Sprays, ORNL-4076, p 169.

11. Nishizawa, Y., et al., "A Study on the Removal of Iodine From the Atmosphere by Sprays," Journal of the Japan Society of Nuclear Power, Vol. 8, No. 11, November 1966, pp 598 - 602.

12. Maekawa, T., et al., "A Study on Removal of Iodine from Atmosphere by Spray I - Removal of Iodine by Spray Under Atmospheric Pressure." Journal of the Japan Society of Nuclear Power, Vol. 7, No. 10, pp 563 - 569, 1965.

13. Hilliard, R. K., Coleman, L. F., and McCormack, J. D., Comparisons of the Containment Behavior of a Simulant with Fission Products Released from Irradiated UO_2, BNWL-581, March 1968, Table 4, p A.5.

14. Collins, R. D. and Hillard, J. J., Some Experiments Relating to the Behavior of Gas-Borne Iodine, TRG-R-983(W) U.K.A.E.A., Windscale, England, April 1965.

15. Parker, G. W., Creek, G. E., and Horton, N. R., Dissolution and Hydrolysis of Methyl Iodine in Misting Spray Solutions, Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1967, ORNL-4228, April 1968.

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Now, will you proceed to the other?

MR. JEWELL: Mr. Holmes, the Board also indicated its interest in a complete and concise statement updating the research and development -- excuse me.

Mr. Holmes, the Board has requested that you update all of the research and development programs which have been undertaken in connection with this application. Have you prepared a response to this request of the Board?

MR. HOLMES: Yes, I have.

MR. JEWELL: Is your answer contained in a document which is entitled "Applicant's response to Board's Question on Research and Development?"

MR. HOLMES: Yes, sir, it is.

MR. JEWELL: Mr. Holmes, was this document prepared under your supervision?

MR. HOLMES: Yes, it was.

MR. JEWELL: Are all of the facts set forth in the document true and correct?

MR. HOLMES: Yes, they are.

MR. JEWELL: Do you here adopt that document as a part of your testimony in this case?

MR. HOLMES: Yes, I do.

MR. JEWELL: Mr. Chairman, I would like to request that this document identified by the witness be inserted and incorporated in the transcript of the testimony

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and that it be treated as the testimony of this witness to the same extent as if read in full.

CHAIRMAN WELLS: It is so ordered.

(The document follows.)

In the matter o.'	)	
	)	
ARKANSAS POWER AND LIGHT COMPANY	)	Docket No. 50-313
	)	
(Russellville Nuclear Unit)	)	

APPLICANT'S RESPONSE TO THE BOARD'S QUESTION

ON RESEARCH & DEVELOPMENT

QUESTION:

"One of the things that concerns me is, as a member of the Board, and I think it does my colleagues, is at the construction permit stage much of the design is yet to be completed, there is still research and development to be done. That is normal and we accept that. But that does mean that we have to come to some kind of conclusions as to whether there is reasonable assurance that it will be done. Now again we may have a comparatively easy task here, because it appears it is uncontested. But noting in both the applicant's summary description and the staff evaluation there are identification of additional research and development to be done and also further design to be completed -- at the hearing I think it would be useful, if you could, to update these things to the extent that you can."

ANSWER:

The status of the Research and Development programs discussed in Section 1.5 of the Russellville Nuclear Unit PSAR, and updated in answer to Question 1.3, Supplement No. 3 of the PSAR, is given below.

(1) B&W Once-Through Steam Generator Development and Tests

All active testing portions of this program have been completed. The steady state and transient operation tests have confirmed the analytically predicted performance characteristics of the steam generator and have provided a means for developing a satisfactory control scheme. Feedwater spray nozzle tests have demonstrated that the design arrangement will satisfactorily heat feedwater. Tube leak simulation tests have demonstrated that a leak in one tube will not propagate by causing a failure in adjacent tubes. Mechanical tests have demonstrated that the tubes can withstand, without failure, the mechanical loads they may experience either during normal operation or accident conditions. Vibration testing demonstrated that the unit contained no undesirable resonance characteristics.

Tests to simulate a steam line failure or reactor coolant system failure have demonstrated the integrity of the steam generator under conditions of rapid depressurization and large temperature differentials between the tubes and the shell of the unit.

The results of these tests have been evaluated to the extent necessary to establish final design characteristics for manufacture of the steam generators. Thus, the design of the steam generators, which are being manufactured, is based on data already available.

Work is progressing to apply the results of the blowdown testing performed to the development and verification of analytical models to predict steam system blowdown. The work will be reported when completed.

(2) B&W Control Rod Drive Test Program

The development and testing of the rack and pinion control rod drive is essentially complete. It was conducted under three separate programs:

A. Full-scale prototype testing at reactor operating conditions of temperature, pressure and flow.

B. Full-scale prototype testing under no-flow conditions.

C. Components testing.

A. Full Scale Testing with Flow

A complete life-cycle test of a full-scale control rod driveline prototype was conducted under conditions simulating all reactor operating environmental conditions except radiation. The driveline prototype components used in prototyping the driveline included the fuel assembly, control rod, upper guide tube of the reactor internals, and the drive mechanism. This testing concentrated mainly on the performance characteristics under coolant flows ranging from zero to full flow at reactor conditions of temperature, pressure, and water chemistry. A major objective of these tests was to determine the compatibility of the mechanism trip time with the specified 1.4 seconds for 2/3 insertion. The measured trip time ranged from 1.37 to 1.4 seconds.

The complete life test of full-stroke cycles and trip cycles was conducted simulating maximum tolerance misalignments in the driveline. Examination of components following the test indicated unacceptable wear of a miter gear, although the drive continued to operate satisfactorily throughout the test. The lifetime test of full-stroke cycles and trip cycles was then re-run in the full-scale prototype test without flow (described below) with a miter gear of improved materials. This latter life test showed acceptable performance of all aspects of the prototype driveline.

B. Full-Scale Testing Without Flow

A prototype control rod drive was tested under no-flow conditions in an autoclave in which the reactor conditions of control rod stroke, temperature, pressure, and water chemistry were duplicated. The tests were performed with a dummy weight equivalent to the weight of the control rod assembly attached to the rack. The mechanism was subjected to approximately 100 full-stroke cycles and 100 trip cycles simulating both hot and cold reactor conditions. This testing verified the design concept and provided preliminary verification of the trip insertion time. The time for 2/3 insertion was less than 1.2 seconds; the snubber design worked properly, and the buffer seal did not impair trip capability.

C. Component Testing

Selected component testing was performed prior to and in addition to the life testing programs in order to resolve potential material or design problems. These component test programs:

(i) Provided the basis for the selection of Graphitar bearing material,

(ii) Established the buffer seal injection flow rate,

(iii) Demonstrated acceptable wear from the revised miter gear combination, and

(iv) Demonstrated that corrosion product buildup in the static test of the splines and bearings did not noticeably affect the resistance to rotation of the system.

All active testing portions of the program have been completed, and only analysis of test results and documentation remain to be done. The work done to date has established the adequacy of the design for its intended service.

(3) Self-Powered Detectors

Although the program which led to development of a successful incore detector has not yet been documented, the development program was described in a seminar conducted by B&W for the ACRS and members of the DRL staff on January 31 of this year.

At the present time, incore detectors of the type proposed for use are installed in the Big Rock Point Reactor and in the B&W Test Reactor. The Big Rock Point detectors had accumulated up to 48 months of equivalent full power operation in the Russellville reactor, and the BAWTR detectors had accumulated the equivalent of 29 months of full power operation. These tests have demonstrated the successful development of the detectors for use in power reactors. Only the longevity of the detectors remains to be determined by the testing that is continuing.

(4) Core Thermal and Hydraulic Design

The PSAR contains, in Section 3, an evaluation of the core thermal capability in which the heat transfer limits were predicted based on a correlation of experimental DNB data developed by the Babcock & Wilcox Company. In order to completely substantiate the B&W correlation additional research and development data is necessary.

Core thermal performance has also been evaluated using the W-3 correlation for predicting DNB. This correlation is available in the literature and has been used and found acceptable in establishing thermal design limits for other large pressurized water reactors. The hot channel DNB comparison using the W-3 correlation is also discussed in the PSAR in Section 3.2.3. With the use of this correlation, only the vessel model flow tests described in the references given in Question 1.4, Supplement No. 3 of the PSAR, are necessary to substantiate operation of the plant within acceptable thermal limits. Flow testing which demonstrated acceptable flow distribution for the rated power level without check valves in the model has been completed. The flow testing with check valves installed and with open check valves is under way at the present time and will be completed during 1968.

(5) Emergency Core Cooling and Internals Vent Valves

B&W has completed development of an analog computer program to predict the forces which would be exerted on the reactor internals during a loss-of-coolant accident. The results of this analytical program as applied to the B&W product line were discussed with members of the ACRS and the DRL staff on January 31 of this year. The analog results are being correlated with test data from the quarter-scale LOFT blowdown tests. A report of B&W analytical results to date is now in preparation.

The analytical methods to be used for evaluating the effect of blowdown on the internals, as well as the stress and deformation limits to be allowed, are presented in answer to Question 9.11 in Supplement No. 1 of the Crystal River Nuclear Unit No. 3 application, Docket 50-302, adopted by reference here in answer to Question 10.4, Supplement No. 3.

Internals vent valves testing as described in Supplement No. 3 to the PSAR, Question 1.3(e), has been completed. Test data now under analysis are expected to confirm the design basis.

(6) Fuel Failure

The ACRS has recommended that evidence be obtained to show that fuel rod failure during a loss-of-coolant accident would not affect significantly the ability of the ECCS to prevent clad melting. Consequently, a program to evaluate the applicability of available data and to obtain data where necessary was undertaken by the B&W Company. The preliminary work on this problem was completed in January of 1968 and is discussed in detail in Supplement No. 3, Item 5 of Question 1.5, of the PSAR. The results of this work support our analytical conclusion that the loss-of-coolant accident could lead to some clad deformation which reduces the cross-section flow area in some channels. However, since emergency cooling is accomplished by a flooding mechanism, the deformation will not interfere with the ability of the emergency core coolant to limit the cladding temperature following the accident. Current plans include performance of a three-phase program. In the first two phases, which are experimental, single-rod excursions will be performed to better establish temperature-pressure relationships at the time of clad perforation. The single rod tests of the first phase will also investigate the extent of deformation to be expected for specific conditions associated with in-reactor temperature excursions. The second phase of the program will consist principally of a series of multi-rod tests to explore the effect of the restraining action of spacer grids and adjacent fuel rods and to determine the distribution of the localized deformations in an assembly of fuel rods. In the third phase of the program the data obtained from the two experimental phases will be applied to the analysis of the effects in a loss-of-coolant accident.

This program for the evaluation of the effects of perforation and deformation of fuel rods, including the analysis and application of the data obtained, is scheduled for completion during 1969.

(7) Xenon Oscillations

A program to evaluate the possiblity of xenon oscillations throughout life is underway. This program includes development of one-dimensional, two-dimensional, and three-dimensional computer programs to be used in the evaluation of core stability margin. The one and two-dimensional analyses should be completed during 1968. The current status of these analyses is that a modal analysis to permit evaluation of xenon instability thresholds is nearing completion. The three-dimensional analysis will be completed in 1969.

The design of means to eliminate or control xenon oscillations is being carried out in parallel with the evaluations to determine if they will occur. These means include the use of fixed shims or burnable neutron poison rods to reduce the positive moderator coefficient if necessary, and the use of partial length, movable control rods for controlling axial oscillations if they occur. This latter system will be verified by two-dimensional calculations and the interaction of these rods with other full length rods that possibly could be inserted will be ascertained in some three-dimensional analysis. These parallel programs will provide the means for solving the potential xenon oscillation problems in the event that they occur.

(8) B&W Development of Iodine Removal System (Chemical Additive to Containment Sprays)

The answer to Question 1.3(g) in Supplement No. 3 to the PSAR references the basis of R&D requirements to demonstrate the ability of the chemical sprays to effectively remove and retain iodine and to demonstrate the compatibility of the chemical with plant materials. The references include a list of 17 experiments which were planned by the Oak Ridge National Laboratory to provide information to substantiate the effectiveness of the sprays for removing iodine. These experiments have now been completed and are reported in Reference (1). An evaluation of the results indicates that they substantiate the effectiveness of the chemical spray system. The spray system effectiveness for iodine removal is discussed more fully elsewhere in the testimony of this hearing.

(9) High Burnup Fuel Tests

One additional R&D program is discussed in the Russellville Nuclear Unit PSAR. This program, however, is not required to finalize design details, but rather is associated with product improvement as related to the economics of future cores, and thus with future operational safety aspects of the plant. Some lower burnup samples have been removed from test and are awaiting analysis. Long-term irradiation of high-burnup capsules is continuing.

Reference:

(1) Parlsy, L. F. and Franzreb, J. K.; "Removal of Iodine Vapor from Air and Steam-Air Atmospheres in the Nuclear Safety Pilot Plant by Use of Sprays"; ORNL-4253; June, 1968.

CHAIRMAN WELLS: Thank you, Mr. Jewell.

Does staff counsel desire to introduce its response to questions by the Board?

MR. ENGELHARDT: In connection with the Board's question regarding iodine removal by sprays I would like to offer for identification as Staff Exhibit 3 a 10-page document entitled, "Iodine Removal by Sprays," dated October 28, 1963, which was prepared by the Division of Reactor Licensing.

This document which I have just identified was provided to the members of the Board and to the applicant last evening and is one of those documents which the Chairman previously has identified.

I would like to offer this identified Staff Exhibit 3 for inclusion in the record of this proceeding. I believe that at a suitable time in the proceeding later that this report will be the subject of some discussion by the regulatory staff in its further testimony and possibly will be the subject of some questions by members of the Board.

But at this point I would like to request that Staff Exhibit No. 3 be made a part of the record of this hearing.

CHAIRMAN WELLS: It is so ordered.

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(Staff Exhibit No. 3, marked for identification, was received in evidence.)

CHAIRMAN WELLS: I would like at this time to thank the applicant and the staff for their preparation of these responses. It could have been left to oral testimony. And we very much appreciate the fact that you have prepared it and given us an opportunity to read it before the hearing this morning.

It is now 25 minutes to one. I think we should try to adjourn around one o'clock or shortly thereafter.

I wonder if it might not be useful if the Board now asked or called to the staff and the applicant's attention a third question which was asked at the pre-hearing conference. I think along with the two that we just referred to, it is one of the three most important questions that were asked.

If we might have your responses to that, then perhaps we could have the benefit of reading the transcript before the adjournment of the session this afternoon.

The question I am referring to is a question I think was alluded to by all three members of the Board at different times. And that is the matter of quality assurance.

I don't know how you want to proceed with that or

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whether or not the staff desires to respond to the qustion that I asked at the prehearing conference, namely, what was the basis for the staff concluding that the quality assurance was adequate. Or maybe you would prefer that the applicant respond to the question.

I believe Dr. Quarles asked to what extent did its quality assurance program take into account the possibility of strain on the system because of possible delays in delivery of important components.

Which of you gentlemen would prefer to proceed?

MR. JEWELL: Mr. Chairman, the applicant is ready to proceed. We do not have this prepared in written form that we can hand to you as we did the other questions which were rather lengthy answers, about 18 pages. We do not have copies -- or three copies that we could hand you.

CHAIRMAN WELLS: I want to emphasize that the Board does not wish to imply that we think you should have had it written. We are grateful that you wrote the other two out, but we believe that it would be in the interest of all concerned for us to have a chance to read the transcript of this.

Now, if it is 18 pages, Mr. Jewell, I think that would take a long time for us to read.

I wonder could one of your witnesses summarize it for us?

MR. JEWELL: Mr. Chairman, we con: der that to be an important question and we would like o have it in the transcript in full.

CHAIRMAN WELLS: Very well. Do understand that the applicant is prepared to introduce thi document now?

MR. JEWELL: We are prepared to ve Mr. Holmes read it, yes.

CHAIRMAN WELLS: I see. You are repared to read it.

Let me inquire of the regulatory taff, would it be preferable to have the applicant rea the statement before you make your response?

MR. ENGELHARDT: I think that th whole subject here is to intertwined that it probably ma s little difference who leads off. We are prepare to respond to that portion of the quality assurance ques on that was raised by the Board. We know the applican is well prepared to do so. And I think that whatever greement is reached as to how best to proceed, the matt of quality assurance relates to several different que ions from the Board, and it should all be kept at least the same point in the transcript, so that we may have the nefit of all of these thoughts on this subject matter.

But if the Board would prefer th the regulatory staff proceed with certain direct questions that it is responsible for responding to at this point, we can do so.

CHAIRMAN WELLS: This is what occurs to me. We will have the benefit of the written document after Mr. Holmes reads his statement, as I understand it. Is that correct, Mr. Jewell, we will be given a copy of the Holmes statement after he has read it.

MR. JEWELL: [illegible] sir, we do not have copies [illegible] this statement prepared. [illegible] apologize for not having [illegible] prepared, but it was completed last night and there [illegible] was not time to have [illegible] written form [illegible] insertion in the record.

CHAIRMAN WELLS: Well, let me make another [illegible] suggestion. Since we are [illegible] having an opportunity of reading the actual words of both the applicant's [illegible] statement and the staff's statement, are you agreeable to staying in session, say, until 1:15 and then with the possibility that maybe [illegible] afternoon the reporter would be able to give us a copy [illegible]. Would that be agreeable.

MR. JEWELL: Perfectly agreeable to the applicant.

CHAIRMAN WELLS: Then, will you proceed, Mr. Jewell.

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MR. JEWELL: Mr. Holmes, the Board has expressed an interest in the quality control problem of the Russellville Nuclear Unit. Specific information was requested concerning the qualifications of the key personnel involved in quality control and the effect of possible delays of equipment delivery and quality control and scheduled completion dates for the unit.

Will you please state the position of the applicant regarding these matters?

MR. HOLMES: Let me begin by commenting on Arkansas Power & Light's philosophy concerning the quality and safety of the Russellville Nuclear Unit. Arkansas Power & Light Company expects to have the Russellville Nuclear Unit completed and operating on schedule. We will certainly exert our best efforts toward that end.

However, we are even more concerned with the health and safety of the public and the plant personnel. Our primary objective is to have the Russellville Nuclear Unit operate safely and reliably.

To insure that this goal is achieved, Arkansas Power & Light has provided a multi-level quality control and quality assurance program, including inspections both in vendor shops and at the construction site.

The primary purpose for the quality control programs is to insure that the codes, standards, and overall quality called for in the Preliminary Safety Analysis Report and in our specifications and detailed designs are rigorously adhered to by fabricators and constructors alike.

Now let me address myself to the Board's concern that delays in equipment delivery might affect the emphasis placed on quality control for the Russellville Nuclear Unit.

We realize that unforeseen events might adversely affect our construction schedule. However, we also are aware that pressure to get the unit on-the-line must not and will not be allowed to compromise plant safety. Should we allow such pressure to interfere with our quality control efforts, we would not meet our responsibility for protecting the health and safety of our employees and the general public.

We believe that properly administered quality control and quality assurance programs are necessary to prevent delays due to faulty materials or workmanship and therefore supplements our maintenance of the construction schedule.

Consequently, our Quality Assurance Committee reports directly to the Vice President and Chief Engineer of the company, Mr. J. D. Phillips, who has the authority and will utilize his authority to prevent considerations of schedules and expediting to interfere with our quality assurance program.

Mr. Phillips is directly responsible for the design and construction of our production facilities and transmission and distribution systems and for the operation of our generation and transmission facilities. He graduated from Mississippi State University in 1941 with a degree of Bachelor of Science in Electrical Engineering. Mr. Phillips is a registered professional engineer in both Mississippi and Arakansas.

He is a member of the National Society of Professional Engineers, the Arkansas Society of Professional Engineers and the Institute of Electrical and Electronic Engineers.

Mr. Phillips has been in the Midle South Utilities, Inc., organization in various positions since 1946, having spent 17 years with the Mississippi Power & Light Company before joining Arkansas Power & Light Company as Assistant Chief Engineer in 1963.

Under Mr. Phillips' supervision in the Engineering Department is a staff of 20 graduate engineers, any of whom may be used in specific areas in the construction of the Russellville Nuclear Unit.

Mr. Phillips is a member of the Technical Committee of Southwest Atomic Energy Associates. He participated in the studies leading to the SEFOR project and in the planning of that project. He has kept himself informed of this project through attendance at meetings, review of periodic reports and visits to the site.

He has served as the Company representative at meetings of the Southern Interstate Nuclear Board. He regularly reviews reports from High Temperature Reactor Development Associates, Inc.

He has followed and participated in all of the nuclear projects in which the company has been a participant and to which it has been a contributor.

He directly supervised and participated in the study which led to the decisions of the Company to construct the Russellville Nuclear Unit and has actively participated in the planning of this generating plant.

As you realize, one of the Arkansas Power & Light's chief responsibilities is to provide sufficient generating capacity to meet our customers' increasing demand for electric power.

We have always met this demand and will continue to do so. We expect the Russellville Nuclear Unit to be completed on schedule and to assist in meeting our 1973 peak. However, if delays in manufacturing or construction of the unit are encountered we can, and will, take steps to meet our projected load without the

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Russellville plant.

Arkansas Power & Light Company operates as a part of the interconnected system of Middle South Utilities. In 1973, the Middle South System's projected peak demand will be 8093 megawatts and our system generating capability will be 9703 megawatts, including the Russellville Unit.

If the Russellville plant were not available, our system would still meet the projected load, but, of course, our reserves would be lower than desirable.

If delays arise in the manufacture of the major components for the Russellville, plant, we have several alternate means for meeting our demand and reserve requirements independent of the Russellville Nuclear Unit, although these alternatives will obviously impose economic penalities.

Next, I will describe our quality assurance program giving the names and qualifications of several of the key personnel therein.

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13 The desired quality of the Russellville Nuclear Unit is defined by the preliminary designs and specifications, codes, standards and design criteria given in the Preliminary Safety Analysis Report. To insure that this desired quality is reflected in the detailed designs and specifications which will be developed as the project proceeds, several levels of review are utilized.

For example, if a detailed system design or specification is developed by the Bechtel engineers assigned to this project, it must be first approved by Bechtel's project engineer, Mr. H. P. Marsh. Mr. Marsh hold a Bachelors degree in Mechanical Engineering from the University of California and has 25 years experience in many phases of design and construction work. He has worked for Bechtel Corporation since 1954 on the design, procurement, construction and start-up of steam electric power plants. Mr. Marsh is a registered professional engineer in Carlifornia and Arkansas.

Next, the design or specification must be approved by Bechtel's staff of chief engineers. Then, it is submitted to Arkansas Power and Light's production department management and the Design Review Board for approval. Each of these approvals require review to assure that the proper codes, tests, and criteria have been incorporated.

A detailed design or specification developed by B&W -- Babcock and Wilcox -- receives even more review. A

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B&W specification or design is first reviewed at least twice by the B&W staff and then is submitted to both Bechtel and AP&L for review. Therefore, we believe that the detailed design of the Russellville plant will reflect the desired quality assuring safety and reliability.

Our quality control program then has the responsibility of insuring that the specifications for the plant are rigorously met. B&W has a well-staffed force of quality control inspectors that inspect work both in their own shops and those of their suppliers. Overseeing this entire effort is B&W's Quality Assurance Department which independently verifies that B&W inspectors are on the job and that all necessary records are up to date.

The Manager of B&W's Quality Assurance Department is Mr. H. Dobell who has many years of experience in many aspects of nuclear and conventional power plants. Mr. Dobell has been with B&W 22 years. Since 1954 he has been actively engaged in nuclear work including reactor component design, procurement and specification. For the past eight years, Mr. Dobell has been Manager of the Systems Design Section responsible for plant arrangement, containment, fuel handling, instrumentation and control systems including specification and procurement of the components for these systems.

B&W's Manager of Quality Control for their Mt. Vernon Plant is Mr. W. C. Buskey. Mr. Buskey has 27 years of

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diversified experience with B&W, all of which has been in the Quality Control Department including inspections for compliance with ASME codes, preparation of specifications and procedures, test equipment design and fabrication and review of customers' specifications.

Mr. James C. Quinn is B&W's Manager of Quality Control at Barberton, Ohio. He holds a Bachelor degree in mechanical engineering from Steven Institute and a Master's degree in mechanical engineering from Carnegie-Mellon Univerisyt. He has 31 years of experience with B&W in various aspects of engineering and erection work. He is a registered professional engineer in the State of Ohio.

Mr. J. L. Perkins is B&W's Chief Quality Control inspector at Barberton. He has been involved in inspection work for B&W for 14 years. Mr. Perkins holds a bachelor degree in mechanical engineering from Virginia Polytechnic Institute and is a registered professional engineer in Ohio.

B&W's Senior Radiographer in the Power Generation Division is Mr. G. R. Forrer. Mr. Forrer has 26 years experience with B&W in non-desbructive testing work. He has given lectures on Radiography at Oak Ridge Institute of Nuclear Studies and several colleges and universities. He has authored several papers on radiography and radiation safety and is a member of the Society of Non-Destructive Testing. He is also a member of the American Society for Testing and Materials and is first Vice Chairman of Committee E-7 on Non-Destructive Testing. Mr. Forrer assisted in revising the ASM Handbook and was the recipient of the 1961 "DuPont Award" for outstanding achievement in radiography.

Bechtel's Quality Control efforts may be divided into two categories: (1) inspections in vendors' shops; and (2) on-site quality control. Bechtel has a large staff of resident inspectors stationed throughout the nation who routinely visit all vendors' shops. Bechtel has full-time resident inspectors at B&W's Barberton and Mt. Vernon shops. These inspectors witness inspections of key manufacturing steps and review all quality control records thoroughly.

Some of the key Bechtel personnel associated with this aspect of quality control are:

Mr. T. I. McHugh is Bechtel's Inspection Supervisor responsible for inspecting all nuclear steam supply systems. Mr. McHugh coordinates the field inspections and activities between the Inspection Department and the Engineering staff. Additionally, he verifies that the Quality Assurance program is followed. He is thoroughly familiar with pressurized water reactors and boiling water reactors and has had 21 years experience in fabrication, erection and inspection experience with pressure vessels, pumps, heat exchangers, and so forth, on both conventional and nuclear power plants.

Mr. R. E. Smith has been employed in construction, welding, and inspection work since 1935. For the past 18 years

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Mr. Smith has served as Inspection Manager for Bechtel Corporation involving supervision of approximately 50 technical inspection personnel at the present time. Mr. Smith has a thorough working knowledge of all types of non-destructive testing and components for nuclear and conventional power plants.

Bechtel's on-site quality control inspections are performed by Bechtel's field engineers. These are experienced men in various technical fields at the site who are involved in the plant construction on a day-by-day basis. Additionally, Bechtel will have a Quality Assurance engineer in residence at the site who is independent of the construction force and who reports directly to the Bechtel project engineer, Mr. Marsh.

The Quality Assurance engineer's primary duties are to make certain that Bechtel inspectors are on the job and that all Quality Control records are complete. The Quality Assurance engineer has full authority to stop work at the site if the Quality Control requirements are not being met.

Bechtel's Quality Assurance engineer for the Russellville plant will be Mr. D. Wainwright. Mr. Wainwright was graduated from the City University of New York in February 1962 in the upper 10 percent of his class with a Bachelor of Science Degree in Civil Engineering. He is a registered Civil

13 Engineer in the State of California.

In 1961 he joined Goodkind and O'Dea Consulting Engineers as a structural engineering serving in both design and inspection of reinforced concrete and steel structures. In 1962 he accepted a position with the California Department of Water Resources. He was involved in contract administration and construction inspection for the Delta Pumping Plant Project. He attended a course in concrete inspection conducted by the Department of Water Resources.

In 1964 he joined Bechtel Corporation, Power and Industrial Division. He participated in the design of concrete and steel structures for the Intalco Aluminum Plant in Bellingham, Washington. In 1965 he accepted a position as Senior Engineer with Sverdrup and Parcel. He was responsible for the engineering supervision of the construction of thirty-six miles of 500 KV transmission line.

In 1968 he rejoined Bechtel, Power and Industrial Division, as Quality Assurance Engineer for the Russellville Nuclear Unit. He was assigned to the San Francisco design office to become familiar with the specifications, design drawings, codes and quality assurance requirements. In August of 1968 he was temporarily assigned as Assistant Quality Assurance Engineer to the Point Beach Nuclear Unit at Two Rivers, Wisconsin. He has been serving in the above capacity in order to gain familiarity with the Bechtel Quality Assurance Program.

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In addition to the Bechtel and B&W Quality Control Programs, AP&L has an independent effort designed to assure that our contractors' quality control programs are operating effectively. AP&L's Quality Assurance Committee is composed of the following:

Mr. John P. White, Manager of the Production Department. He is a graduate of the University of Arkansas in 1928 with a degree of Bachelor of Science in Electrical Engineering. Mr. White has been with the Arkansas Power and Light Company since May, 1943, serving as Assistant Power Plant Superintendent, Power Plant Superintendent, and, since 1949, as Manager of the Production Department.

Under Mr. White's direction eleven generating units have been added to the company's system. Mr. White is a registered professional engineer in Arkansas, a member of the American Society of Mechanical Engineers, and a member of the EEI Prime Movers Committee. Through studies, reports and meetings, Mr. White has kept himself advised as to all the nuclear projects in which the company has been a participant or to which it has been a contributor, including the Atomic International Project of SAEA, the SEFOR project, and the Peach Bottom Project sponsored by High Temperature Reactor Development Associates, Inc. He has participated actively in the evaluation studies made by the Company of nuclear power and the planning for the Russellville Nuclear

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Unit.

I am a member of the Quality Assurance Committee, and my qualifications have been previously introduced.

Mr. Roger Bottoms, Engineer in the Mechanical Section of the Production Department of Arkansas Power and Light is a graduate of Vanderbilt University with a degree of Bachelor of Engineering in Mechanical Engineering in 1950. He also received a Master of Science degree in Power and Fuel Engineering from Virginia Polytechnic Institute in 1951, and has also continued in post-graduate studies in nuclear engineering at the University of Arkansas Graduate Institute of Technology.

Mr. Bottoms has been with Arkansas Power and Light Company since 1951 and has worked in generating plants, in managerial positions, since 1953. He has participated in the construction and acceptance testing of three new generating units.

In July, 1967, he was transferred to the general office to devote full time to the Russellville Nuclear Unit. Since then he has been actively engaged in planning for this unit.

Mr. Harvey Miller, Engineer in the Mechanical Section of the Production Department of Arkansas Power and Light is a 1960 graduate of the University of Arkansas with a Bachelor of Science degree in Mechanical Engineering. He wb4 has done post-graduate work in nuclear engineering at the University of Arkansas Graduate Institute of Technology. Mr. Miller has been with Arkansas Power and Light since 1962 with most of his experience being as Results Engineeringin the generating stations, including start-up, acceptance testing, and equipment inspection.

In September of last year he was transferred into the General Office to work on the Russellville Nuclear Unit. He is now devoting full time to training and planning for this unit. For the past several months he has been stationed at the SEFOR project.

Mr. Miller is a member of the American Society of Mechanical Engineers and is an Engineer-in-Training for Professional Engineer in Arkansas.

Mr. R. W. Toler, Jr., Assistant Chief Engineer of the Arkansas Power and Light Company, was graduated fromthe University of Arkansas in 1947 with a Bachelor of Science degree in Electrical Engineering. He is a registered professional engineer in Arkansas and a member of the National and Arkansas Societies of Professional Engineers, and the Institute of Electrical and Electronic Engineers.

Mr. Toler began to work for Arkansas Power and Light Company in February of 1947 as an electrical engineer, and has beenemployed by the company sincee that date as an electrical engineer, manager of research and design, and,

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since May, 1967, as assistant chief engineer.

This committee will continually reviewethe results of the quality control program and will plan the necessary inspections to be made by AP&L in vendors' shops.

Mr. Bottoms is AP&L's chief quality control coordinator. He will be in residence at the site and will have the responsibility for verifying that the necessary qualified Bechtel inspectors are at the site at all times and that quality control records are complete, and that AP&L inspectors are at the plant or at vendors' shops to make the necessary spot checks of critical fabrication and construction steps.

Mr. Bottoms has available to him for inspection duties many experienced AP&L employees who are familiar with civil, structural, welding, electrical and mechanical work. When inspectors with specialized training are required that are not available from within the company, such inspectors will be obtained from independent contractors.

When Mr. Bottoms is scheduled away from the site for formal training he will be replaced at the site by Mr. John Anderson. An overlap at the site will be scheduled.

Mr. Anderson, Engineer in the Mechanical Section of the Production Department, is a 1949 graduate of Ouachita Baptist University with a Bachelor of Science degree in chemistry. He has done post-graduate work in nuclear engineering at the University of Arkansas Graduate Institute of Technology. Mr. Anderson has been with Arkansas Power and Light Company in power plants since 1949.

In 1955 he assumed the position of Results Engineer and Assistant Plant Superintendent in 1960. In August, 1968 Mr. Anderson was transferred from Robert E. Ritchie Steam Electric station, a two-unit plant of 906 megawatts capability, to the General Office to devote full time to the Russellville nuclear unit. Since then he has been actively engaged in planning the nuclear unit. Mr. Anderson is a member ofthe American Society of Mechanical Engineers. He participatedin the Nuclear Power Familiarization Program sponsored by Arkansas Power and Light Company.

Mr. Anderson has served as an inspector in a concrete batch plant, and has been involved in the installation and acceptance testing on one of our major steam electric units. He also has served as assistant superintendent of a steam electric plant during later stages of construction, including duties of quality control and acceptance inspections.

In the areas of quality control -- concrete, weld electric, and so forth -- we plan to select men from our Productim Department or other areas of the Engineering Department who are seasoned veterans of construction work, and provide themspecialized training and famliarization with the specifications for the Russellville plant.

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The applicant realizes and accepts the full responsibility for the Russellville Nuclear Unit, and we believe that the organization I have described to you will insure a safe and reliable plant.

CHAIRMAN WELLS: Thank you.

I wonder if the staff now would like to proceed?

MR. ENGELHARDT: I would like to call Mr. Check.

During the pre-hearing conference the Board requested the staff provide to the Board information as to how the AEC regulatory staff evaluated the quality assurance program in terms of the ability of the contractors to meet their obligations on a timely basis.

Mr. Check, would you respond to that question?

MR. CHECK: Before I respond to that question I would like to call attention to the comments he made about Mr.Buskey's qualifications, Mr. Buskey of B&W.

You refer to that testimony, inadvertently I am sure, and to his responsibility to insure compliance with AEC codes. I am sure he meant ASME codes.

MR. HOLMES: You are right, that is a correction.

MR. CHECK: We assumed the thrust of the Board's inquiry into the staff's method of evaluating quality assurance to be directed toward the timeliness of construction, the effect that we might foresee on quality of the plant in view of delays and things of this nature.

With regard to that, I would like to make a statement. And then if it is not entirely satisfactory, we can answer more specific questions from the Board.

CHAIRMAN WELLS: Please go ahead.

MR. CHECK: Our acceptance of the quality assurance program is based in part on the finding that it is not in any way improperly influenced by component fabrication or delivery delays or by on-site construction delays.

We recognize, in fact, that an effective program may indeed cause delays and therefore we assure ourselves that the quality assurance program administration is sufficiently independent of the line organization to avoid the possibility of quality compromise.

Now, this I submit as our answer to what we interpreted your question to be, namely, some general idea of how we, the staff, evaluated the quality assurance program in terms of the ability of the contractors to meet their obligations on a timely basis.

We have outlined for the Board in our safety evaluation in a general way how we reached the conclusion that the current quality assurance program proposed by Arkansas Power and Light is acceptable.

If there are specific points or if the Board would like to inquire further, we will take those questions.

CHAIRMAN WELLS: Thank you very much. I think that your answer was indeed responsive to the specific one that I asked. I have the transcript in front of me. And I said, I would be grateful for any general exposition at the hearing on the quality assurance program taking into account the strain that thereby might be placed. And that follows the statement I made about possible delays.

I wonder, however, if we might not have a little more amplification not just in terms of the fact that you want to find an independent quality assurance program which then becomes independent of all delays. Let me broaden the question somewhat. And if you wish to respond to it after that, you can do so.

Let me ask you this and I will phrase it this way: What general guides, what components of a quality assurance program have you looked for to come to the conclusion that the applicant in this case has such an independent quality assurance program that would enable it to be completely independent of delays off-site, on-site or whatever?

Do you understand?

MR. CHECK: Your question has in it some of my answer, the very important consideration is the independence of the quality assurance organization, vis-a-vis the line organization.

In addition to that, we wish to make certain that the quality assurance, the responsible quality assurance personnel, are vested with sufficient responsibility to stop or authorize the cessation of work if any variance or non-conformance is detected.

I am afraid I am somewhat hard put to expand on this question in or answer beyond what we have stated in the safety evaluation. We have had available to us for our review much the same information that you have heard from the applicant today.

We have had a series of meetings with them in which we explored with him more precisely the structure of his program and his systems, quality assurance systems, on each of his principal contractors. He had responded to questions from us in amendments three and nine to the application, and in addition in amendment nine had in effect commited himself to providing us with additional information which we believe to be important to allow us to make a final decision that the quality assurance program is indeed acceptable.

These items which are also enumerated in our safety evaluation are considerations or are pieces of information which will allow our compliance division which is responsible for inspection of construction as it progresses, to allow them to do -- to perform their function more adequately.

CHAIRMAN WELLS: Mr. Check, excuse me. May I interrupt. Maybe I wasn't sufficiently clear and it may

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not be possible to respond entirely to my question, but I would like to try again to phrase it because I think it is fairly important that we get a responsive answer, if we can.

Let me phrase it this way: In the testimony which Mr. Holmes read he stated that the Committee, within the management of the applicant, Arkansas Power and Light, whose job it would be to ascertain that the quality assurance programs are properly carried out by the contractor. And you have stated that you have inspected and that the staff has evaluated the information given to you by Arkansas Power and Light.

What I am trying to find out -- and admittedly I ask this questions not as a person with a scientific background. Perhaps after I confer with my colleagues at lunch, I will wish I had not asked the question, but as a non-scientific person, the thing that bothers me, what I am simply seeking to find out, is how, one, does Mr. Holmes decide that the quality assurance program is adequate? And, secondly, how have you decided that the quality assurance program which the applicant proposes to carry out is adeqate? What criteria do you have? What constitutes a good quality assurance program?

I think we certainly identified one essential criterion, that is, that there be independence. But what else is there, if there is anything?

MR. CHECK: Mr. Wells, I will try to be responsive. We do not currently have any specific criteria against which we evaluate quality assurance programs.

We have informed another Board in a recent proceeding that we have such criteria under development.

CHAIRMAN WELLS: Just as a matter of interest -- and don't misunderstand me, I really wantto see if I can't get something useful on the public record. First of all, let's distinguish between any published or formal criteria. I know you don't have those. If you had them, I would be aware of them. I am speaking of what kind of mental processes do you use.

As I said at the prehearing conference, it is not conceivable to me that somebody reaches a judgment about something without having some basis for that judgment, and that is what I am trying to find out.

MR. CHECK: In the absence of specific criteria, the alternative to specific criteria, of course, are the combined judgments of many knowledgeable people.

Now, we had a review of this application including the quality assurance programs proposed by members of our technical staff who are sufficiently experienced in these matters to make a finding based upon their judgment which is based, of course, on their experience, their working experience in these areas, that these programs are indeed adequate at this time. I do want to indicate to the Board that quality assurance is becoming of increasing interest to us. We are evolving methods for evaluating programs for quality assurance.

We are very much concerned with being able to respond clearly and affirmatively to questions such as the Board if raising,but at the present time we will have to rest our case on the combined judgment and experience of knowledgeable peple on the staff of the Commission.

CHAIRMAN WELLS: Thank you, Mr. Check, I think that has been a very useful answer. And as far as I am concerned, it basically satisfies me.

I would like to ask one other question. Again, I believe that those members of the public here are entitled to know the answer to this.

As you indicated, some earlier Board asked the question as to whether there are criteria. I believe that the Board was informed that criteria were under preparation. I don't know whether or not we will every be able to have very specific criteria, but it will be useful to know what progress are you making, what predictions can be made as to the time when some criteria might be available.

MR. CHECK: I can reiterate what our response was to the earlier Board, and that is several months.

I'm afraid I can't be more specific than that. I can tell you that progress is being made.

I would like to add to that our review of this plant and all of its ramifications, its organizational Quality Assurance program as well as finalization of design, are under continuous scrutiny by the staff. We have identified a number of areas which we intend to continue to review with the applicant even following the granting of a construction permit.

CHAIRMAN WELLS: Now, in order to round out this line of questioning and perhaps go to lunch I want to state again, so the members of the public may know, or anyone who may read this transcript, as to what is the reason the Board has asked these particular kinds of questions.

This is an uncontested hearing. We are not required under the regulations to make independent judgments as to whether or not the Quality Assurance program of the applicant is adequate or not. If I were, as I say, I would be greatly impressed with the statement that the applicant has made. We are entitled to rely upon it, and of course, we like to be able to rely upon the evaluation made by the staff.

So when we ask these questions of the staff, again it isn't because we doubt that the staff has carefully examined it; I am sure the staff has. But we want to be able to have in the record something so that we can say that we knew something about how the staff evaluated the Quality Assurance program

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and then be able to conclude that that was an adequate examination.

I wonder if my colleagues have anything on this?

DR. QUARLES: Nothing.

MR. BRIGGS: Nothing.

CHAIRMAN WELLS: If not, I propose that we adjourn, it now being 1:25.

Would 2:30 be agreeable as a time to reconvene?

We will reconvene at 2:30.

(Whereupon, at 1:25 p.m., the hearing in the above-entitled matter was recessed to reconvene at 2:30 p.m. the same day.)

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AFTERNOON SESSION

(2:40 p.m.)

CHAIRMAN WELLS: The hearing will come to order.

Whereupon,

HARLAN T. HOLMES,
KNOX M. BROOM,
JAMES MC. FARLAND,
ROBERT E. WASCHER,
WILLIAM R. SMITH,
HARRY P. MARSH,
R. PAUL SCHMITZ, and
HOWARD W. WAHL

on behalf of the applicant, and

CHARLES LONG,
PAUL CHECK, and
ALBERT SCHWENCER

on behalf of the regulatory staff,

resumed the stand and, having been previously duly sworn, were examined and testified further as follows:

CHAIRMAN WELLS: I am going to try a new microphone this afternoon. Can I be heard in the rear of the room?

Before the recess, the staff had answered the question of the Board concerning the evaluation of quality assurance. If wonder if that completes your answer Mr. Engelhardt, or is there anything that you might like to

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add?

MR. ENGELHARDT: Mr. Chairman, I think that we may have left a rather inconclusive impression as a result of some of our responses before the luncheon break, and I would like to ask Mr. Long to amplify on certain of the testimony regarding the quality assurance program that has been put in direct by the staff earlier.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

Mr. Long?

MR. LONG: I think first, in order to put the thing into perspective, particularly in relation to what Mr. Check said earlier this morning, is that if we go back to the original application and the questions asked by the staff with relation to the quality assurance and quality control program, I think this first starts to set the stage for the type of thing we, the staff, are looking for and are reviewing against.

I would like to list these question numbers as they appear on the application. They are questions 8.3, 8.4, 8.5, 8.9, 9.5, and 9.7.

Now these were questions that were formally requested of the applicant to supply us additional information on in relation to quality assurance, quality control programs. These questions are quite pointed. They indicate our concern and certainly indicate the areas in which we were looking to determine that the program that this applicant was going to

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use would meet at least these minimum situations.

I think, secondly, it should be pointed out that as Mr. Check mentioned earlier, we did have a meeting with the applicant in August, at which the subject was specifically quality assurance and quality control. Again, because of the lack of criteria that the Commission has at this point that are published, the persons attending this meeting were not only myself and Mr. Schwencer but also Dr. Beck, who is our Deputy Director of Regulation, also a Mr. William Morrison, who is a member of the Division of Reactor Standards whose sole purpose is to develop the criteria for quality assurance and quality control programs and whose experience is extensive in the U. S. Navy Quality Assurance Program.

We also had at that same meeting two members of the compliance organization, one of which of course is the inspector who will be involved with the Russellville plant itself and who has had extensive experience in reviewing other plants that are now under construction. And there were also staff members of the Division of Compliance present at the same meeting who themselves are working in conjunction with the Division of Reactor Standards to develop these criteria.

Now this is the type of personnel that the AEC staff had at that meeting.

Now the applicant, of course, came well prepared as well.

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At this meeting we went through and we had some basic guidelines which we had given the applicant in advance of that meeting so it could be fruitful, and this dealt with such things as the organization, including the contractors and subcontractors involved at the site, and also organization charts concerning the interrelationship between the organizations that exist, in this case the Bechtel organization and the B&W organization, and Arkansas Power and Light, the delineation of the division of responsibility, and the relationship between the organization -- of the various organizations and the procedural documents that would be used to follow this through from Arkansas Power and Light down through the various organizations.

In addition, we did discuss at this particular meeting -- and of course Mr. Holmes this morning put on the record the names and titles of persons who would have the major responsibility for quality assurance for all three of the organizations. And in addition to this we went through a rather detailed discussion of what actually happens when one piece of equipment, for example,-- And Bechtel, for example, used an example of an air compressor, if I remember it correctly, that they had followed through on the Monticello project which had already been ordered, built, delivered and so forth, and installed. In this procedure they showed us clearly the forms that were required to be filled out, the

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sign-off procedures required, and where if there were any storage problems, things of this nature, how they were delineated and the paper work, who it was reported back to and things of this nature.

B&W also presented an equal situation of how they within their own internal organization, handle the situation.

Now based on this meeting and the persons who attended that meeting along with the information which is in this application which-- By the way, Amendment 9 did bring up-to-date the information that we discussed at this meeting. It was the opinion of the staff members who did this review that we had done at this point the best we could in terms of what we understood to be and now still understand to be the tentative criteria that are now being generated internally within the regulatory staff although not published.

Although I cannot give you those criteria, the exact guidelines and so forth, I can only say that this applicant has been reviewed against what we believe to be those criteria at this time.

Now, have I helped in this answer or not?

CHAIRMAN WELLS: You helped me tremendously, Mr. Long, and I am grateful to you. Actually, I think this has been a most interesting statement.

My colleagues have not asked questions about this particular point and I don't know whether they have any comment ab6

to make or not. If not, let me thank you very much for this additional information, and we will proceed with the other answers.

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I believe it would be convenient for the Board now, if it would be convenient to the applicant and to the staff to pick up the remainder of the answers in whatever order pleases you.

MR. JEWELL: Very well, sir.

Mr. Holmes, will you please tell us how independent are the sources of off-site power and whether or not any single accident could cause failure of all off-site power?

MR. HOLMES: There will be two 500 kV and two 161 kV transmission lines serving the station switchyard. The two 500 kV lines will come into the plant from the west on one right-of-way. The two 161 kV lines will come into the plant from the north on an entirely separate right-of-way.

The two right-of-ways will not be adjacent to each other at any point outside of the switchyard. Any one of the four transmission lines will be sufficient to supply auxiliary and emergency power to the Russellville plant.

Power circuitbreakers with proper line protective relays will be provided at each end of each of the four transmission lines. Any single electric fault caused by any means whatever will automatically and selectively isolate the affected line from the system and off-site auxiliary and emergency power will still be available to the plant.

Because of the physical separation of the lines and the protective devices a single local accident will not cause the loss of all off-site power to the plant. System stability studies indicate that no single accident can cause a blackout of the system and thereby cause a loss of all off-site power to the plant.

MR. JEWELL: Shall we proceed, Mr. Chairman? Or does the Board desire to ask any further questions?

CHAIRMAN WELLS: Mr. Jewell, let me consult with my colleagues.

Would it be more desirable to ask at the conclusion of each one of these answers any questions that may come to mind?

DR. QUARLES: I think it would be better to do it as we go along, but I have no questions.

CHAIRMAN WELLS: Mr. Briggs, do you have any questions?

MR. BRIGGS: No questions.

CHAIRMAN WELLS: Then we can proceed to the next. But after each one of these then, we will amplify our questions if we need to do so.

MR. JEWELL: The Board has expressed an interest in the automatic selection of off-site power. How reliable

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is this equipment? And what can the operator do, if there is a failure in the automatic selection?

MR. HOLMES: There are four measures which are designed to reduce the likelihood of automatic transfer failures. (A) Careful inspection and quality control of circuit components and wiring during construction.

(B) Pre-operational testing of individual components and systems.

(C) Routine testing of components and systems while in service.

(D) Continuous monitoring of control power supplies.

The two 6.9 kV buses and the two 4.16 kV bus systems are capable of being supplied independently from a unit auxiliary transformer or from each of the two startup transformers. Bus transfer circuits for every supply circuit breaker on any bus are independent from each other and a simultaneous automatic transfer failure affecting both 6.9 kV or both 4.16 kV buses is therefore extremely unlikely.

However, in the unlikely event that the automatic transfer fails, other automatic features or manual actions occur as follows:

If following protective relay action auto-transfer fails to connect one or more buses to one of the two available standby supplies, the motor breakers connected to the affected bus or buses will be tripped by the under-voltage relays. This may or may not result in unit trip depending on what power level the unit was being operated at and whether one or more buses failed to transfer.

Two, if the automatic transfer fails, each normal bus can be connected to a standby supply manually and the motors can be restarted by manual operation.

In each instance the manual control is performed by the use of control switches in the control room.

Three, a failure to transfer and the consequent loss of voltage on one or both 4.16 kV engineered safe-guard buses will result in an immediate automatic start of both diesel generators.

Within 15 seconds the diesel generator circuit breaker will close and power will be available for the engineered safeguard equipment.

The outside supply can be reestablished to the affected bus or buses by manually closing the outside source circuit breaker operators over synchronizing inter-locks and then manually tripping the diesel generator circuit breaker.

CHAIRMAN WELLS: Would you proceed to the next answer, please?

MR. JEWELL: Mr. Holmes, the Board has expressed

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concern about the gas pipeline which goes across the property near the site of the reactor. Will you please describe the several imaginable accidents relating to this pipeline which you and your consultants have analyzed.

MR. HOLMES: The cooling water discharge channel crosses an existing 10 and 3/4 inch OD natural gas transmission line about 800 feet from the proposed plant. The shortest distance between this pipeline and the plant is about 600 feet.

The portion of the pipeline which is crossed by the water channel was constructed in 1928 and has a maximu operating pressure of 500 psig.

We propose to construct a pipeline crossing by installing 1200 feet of new pipeline to the same specifications as were used for the portion of the 10 and 3/4 inch pipeline constructed in 1962 and to which the new crossing section will be connected.

The proposed crossing will be buried beneath the outlet water channel with four feet of earth cover. This construction will meet specifications for class C construction of USASB-31.8-1968.

We have investigated the possible effects of a pipeline failure on the safe operation of the Russellville Nuclear Plant by looking at three conditions:

One, the explosive force if a rupture of the pipeline did occur.

Two, the thermal heating effect that would occur per unit vertical surface of the power plant, if the escaping gas from the open pipeline should ignite.

Three, the possibility of introducing an explosive gas mixture into the buildings of the power plant, if the escaping gas from the open pipeline did not ignite.

In the first two cases the investigation led to the conclusion that neither of these occurrences was detrimental to the safe operation of the power plant.'

In the third case, the investigation led to the conclusion that an explosive mixture could not be introduced into the power plant.

The worst postulated rupture effect would be caused by a brittle facture of the pipeline in which an unknown length would split open and the in-place compressed gas would expand without restraint.

In the event of such an occurence, the TNT equivalent is 7/100ths of a pound of TNT per foot of pipe. For comparison, on the order of 35 times this quantity of TNT equivalent is used to open trenches within 30 feet of large operating high pressure gas pipelines without damage to them.

For the effect of thermal heating resulting from ignition of the escaping gas we have computed the maximum escaping rate of 48 million cubic feet per day. This is the open flow capacity of a 10-inch line with initial pressure of 500 psig and final pressure of zero, and a length of 25 miles.

The maximum connected source volume to the pipeline and the limiting flow rate is 42 million cubic feet per day. If this should occur, the computed heat energy received per square foot of vertical surface at a distance of 600 feet from the base of the flame would be 21 btu per hour.

The radiant energy is under the maximum of 36 btu per hour that is expected to be received from solar radiation.

Arkansas-Louisiana Gas Company has stated that pipeline control valves would be closed within two hours of such an event, in which case prolonged heating of the concrete containment structure would not exist.

In the case of no ignition of escaping gas from theopen pipeline, a hazard would exist if that explosive mixture could be introduced into the air intake structure of the sensitive buildings.

The explosive mixture range for natural gas is from 5 to 25 percent byvolume. The vertical orientation of the jet, the low density of the gas and the diffusion tendency of the gas will act to reduce the gas

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concentration in the air to less than then 5 percent minimum required for a combustible mixture within a short distance from the pipeline. Even a 70 mile per hour wind blowing rectly from the source of gas to the plant is insufficient to deflect a combustible mixture into the plant air intake systm. Natural gas is considered to be a simple asphyxiant gas.

This requires 33 percent concentration before mental alertness is affected. There are no chronic aftereffects after exposure to this limit. To detect leakage, gas is this pipeline is odorized.

We have invetigated the possibility that if a pipeline should rupture the pipeline would come out of the trench and direct a stream of gas in a random direction and particularly a whipping horizontal discharge. Our search of the records for a failure of this kind has revealed no such failures.

But the records do show that the rupture we have assumed as a worst case has always opened only the top of the pipe with the bottom portion of the pipe intact in the bottom of the trench. Any other failure would discharge less gas than the worst case assumed -- upward with the pipe in place in the trench.

The only missiles that can be created are small clods of earth. The 600 feet between the pipeline and rms9

the powerplant is greater than the limit of missile flight for any missile, earth or rock, that would be

end 17

disturbed by failure of the pipeline.

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MR. ENGELHARDT: Mr. Chairman, could the staff amplify on that statement before the Board questions any of the witnesses?

CHAIRMAN WELLS: Yes.

MR. ENGELHARDT: I'd like to call Mr. Schwencer to amplify on the statement.

CHAIRMAN WELLS: Mr. Schwencer?

MR. SCHWENCER: Before I do that I would like to for the record indicate that the question on this arose -- and it is our question 2.11 in Supplement No. 3 of the Application

On the basis of our review the consequences of the pipeline rupture and considering additional steps that the applicant has agreed to take to reduce the likelihood of rupture and to minimize the amount of gas escaping after rupture which have just been identified by the applicant, we concluded, as noted on page 13 of our Safety Evaluation, that this pipeline is not a significant hazard to the safe operation of the Russellville plant.

Our consultant, Nathan M. Newmark Consulting Engineering Services also reviewed the applicant's response to this question. They agree with this conclusion which I have just stated. This is noted in Appendix G of our Safety Evaluation on page 68.

Following the pre-hearing conference we had initial discussions with Drs. Newmark and Hall on the question of

eb2

damage resulting from a gas pipeline failure at the Russellville nuclear plant. Dr. Newmark reaffirmed that it was the judgment of his staff that damage to the containment structure resulting from a brittle fracture or explosive rupture of this gas pipeline should not be a serious concern with respect to the safety of the facility, and that this judgment is based on the following facts:

One, at the closest point the pipeline and the containment structure are separated by 600 feet.

Two, that this pipe has no curves or bends in the vicinity of the plant which would tend to straighten the pipe and thus amplify the pipe end movement.

And, three, that experience has shown that without such amplification the pipe end movement resulting from ruptures of either larger diameter pipes operating at higher pressures has never been greater than 30 to 40 feet from the point of rupture.

Therefore, this 10-3/4-inch diameter 500 pounds per square inch pressure pipe would not be expected to have significant movement.

MR. ENGELHARDT: That completes our supplemental statement.

MR. BRIGGS: What kind of weather conditions were looked into? Were conditions of inversion examined for the case where gas would be carried to the reactor building and

eb3

would possibly form a flammable mixture at the reactor building?

MR. HOLMES: May I have a minute?

(Pause.)

MR. HOLMES: The density of natural gas is about .6 of that of air and it is felt that this natural gas will go through the atmosphere far enough so that an inversion process will not kick it back down on the site area.

MR. BRIGGS: You mentioned the buoyant effect, the fact that the gas has a density of .6. Have you considered the mixing that would occur with the surrounding air as the gas issued from this break in the pipeline and actually made calculations of what would happen? Or have you just considered that the buoyant effect would carry this up and there would be no danger to the plant?

MR. HOLMES: It was just considered that the buoyant effect would carry it up and there would be no danger.

MR. BRIGGS: So no calculations have been made on what kind of mixing would occur?

MR. HOLMES: No, sir.

DR. QUARLES: Mr. Schwencer, you mentioned that no significant motion of the pipe would take place, and somewhere along in there you mentioned the possibility of a figure of 40 feet; am I correct?

MR. SCHWENCER: Yes, sir.

eb4

DR. QUARLES: The question is, if this 30- or 40-feet movement took place, and pointed the end of the pipe towards the plant, did you consider what effect this would have on the analysis that Mr. Holmes alluded to?

MR. SCHWENCER: I think our answer to this is that we stated that these were with larger diameter pipes operating at higher pressures, and that we did not consider this pipe, which is a 10-3/4 inch pipe, to have significant movement. I think the answer to your question is that we did not consider that a movement of this pipe would be significant, and that point was not analyzed in detail.

DR. QUARLES: Well, what do you mean by "significant"? Do you mean it would not move enough that it could possibly point toward the plant?

MR. SCHWENCER: Yes. This would be random which would include pointing towards the containment structure.

DR. QUARLES: That's all.

MR. BRIGGS: Mr. Holmes, was any consideration given to putting valves nearer the site than the present valves are?

MR. HOLMES: No, sir. These isolation valves are on either side of the line relative to the plant. The gas company has informed us that they could shut these two valves off within two hours and we feel like this is a relatively small line, and we feel like there is no danger of this eb5 within the two hours of the three areas that we have investigated.

I might say that we have an existing complex of steam generating units and they are all fired by natural gas. We have gas lines that we have been living with for years and we are used to gas lines and really we have no fear of this line.

MR. BRIGGS: What kind of guarantee do you receive that the line will be shut off within two hours?

MR. HOLMES: Only their word that this was within their capability. We did not get a guarantee from them.

But may I say that these valves are in close proximity to the Russellville area and the Russellville people have an operation in the Russellville area and we feel like even the two hours has a good deal of conservatism in it; that because of the close proximity and the ease with which they can get to these valves that they surely should be able to cut this off within two hours.

MR. BRIGGS: You say "close proximity". How far are the valves from the plant?

MR.HOLMES: One must be-- As I recall, one is about a mile away and the other would be about six miles away.

MR. BRIGGS: So there would be no great problem of people at the plant getting to the location of the valves with a matter of 30 minutes or so; is that right?

eb6

MR. HOLMES: The Arkansas Power and Light people?

MR. BRIGGS: I mean people --

MR. HOLMES: Or people form the Arkansas-Louisiana Gas Facility?

MR. BRIGGS: People from the Russellville Nuclear Unit.

MR. HOLMES: We are not likely to take the initiative of cutting these lines off without some check with the gas company.

MR. BRIGGS: Yes, I understand that.

DR. QUARLES: Is there any possibility that some natural catastrophe would make it difficult or impossible to get to these valves? What would be the effect of that kind of a situation?

MR.HOLMES: No, sir. Both of these lines are not on our property. There should be no problem.

DR. QUARLES: No danger of roads being flooded out or blocked by tornadoes?

Really, the question is or the thrust of the question is what would happen if one of them were not cut off? Could you continue to have this flame going for an indefinite period?

MR. HOLMES: I don't know enough about the Arkansas-Louisiana gas distribution system to know the answer to this.

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rms 1

MR. ENGELHARDT: Mr. Chairman, may I raise a point with Mr. Holmes on this matterof the gas line.

CHAIRMAN WELLS: Yes, Mr. Engelhardt.

MR. ENGELHARDT: Mr. Holmes, did I hear you correctly when you stated that the main valves of this line were 6 miles and 1 mile from the plant?

MR. HOLMES: These were estimated figures.

MR. ENGELHARDT: I am looking here at the PSAR, Supplement No. 3, 2.11-1 where it says there is a main line valve located 1.9 miles northwest from the plant and another valve at 2.2 miles southeast. Is there some inconsistency here?

MR. HOLMES: Your dimensions are given from the plant. As I understood his question, I was giving estimates from where the Arkansas-Lousiana Gas people might come from to shut them off.

I don't know. I would have to check those.

MR. ENGELHARDT: It wasn't clear to us just what figure you were using on that point.

DR. QUARLES: I would like to go back to the same line that I was questioning a moment ago. My question doesn't depend on whether the valves are on your property or the gas line property. The question really is: Suppose no one cut one of these valves off and the gas line actually continued beyond your two-hour limit that you assumed in your previous answer? Then what?

MR. HOLMES: I am going to call on Mr. Larry Marsh with Bechtel Corporation for an answer to this one.

MR. MARSH: I think I get the question as being that we could have a continuous flow of the amount we have quoted, some 48 million cubic feet per day.

Now, the source of gas to this line has two points of distribution, one to the southeast and one to the northwest.

Now, these sources are limited in their capability,and subsequent to our study based on 48 million we determined from Arkansas-Louisiana Gas that the source capacity could not exceed 42 million.

Therefore, as this gas continues to leak from the system the initial pressure will tend to die, and so the flow rate will be a decreasing amountwith time.

Therefore, I feel that the extent of the energy release will diminish with time. And the interval between the accident and the closing of the valves becomes less significant with time.

Does that help any, sir?

DR. QUARLES: Yes. You understood the question perfectly. Now,let's see if I got the answer right.

(Laughter.)

It is your belief that even if the valve were not closed that the decrease in pressure and so on from the source would take care of the situation so it could die, let us say, a natural death with no harm to the plant?

That was what I was trying to get. Thank you, Mr. Marsh.

CHAIRMAN WELLS: Will you continue with the next response, please, Mr. Jewell.

MR. JEWELL: Mr. Holmes, will you please tell us what is presently known about the background radiation at the site of the Russellville Nuclear Unit, how much this background radiation is expected to increase during normal operation and how this increase compares with the experience at existing nuclear power plants.

MR. HOLMES: The background radiation level at the plant site is not known at present. The only data of this nature available were obtained by the Arkansas Department of Health from samples of milk, air river water and rain water taken at Little Rock, Fort Smith and Fayetteville, Arkansas.

Milk samples from the Fort Smith and Little Rock areas during 1967 and 1968 averaged from 16 to 45 picocuries of strontium 90 per liter of milk. And an air sample taken recently at Fayetteville, Arkansas, gave a gross beta activity of .1 to .2 picocuries per cubic liter of air.

Rain samples at Little Rock in 1968 gave 3 to 37 picocuries of gross beta activity per liter of water.

Samples of Arkansas River water taken at Little Rock in early 1968 indicate tritium activities of 60 to 200 picocuries per liter of river water.

The Arkansas Department of Health, Radiological Health Department, plans to include the Russellville area in their statewide sampling program which includes domestic water supplies, forage crops and milk.

In addition to the Department of Health program, Arkansas Power and Light has a detailed program underway designed to determine the radiation background at and near the plant site.

Little Rock University and Arkansas Power and Light have entered into a 10-year agreement in which the University will conduct a program to gather data on the environmental characteristics of the Dardanelle Reservoir. This program will aid in determining what effects the Russellville Nuclear Unit will have on the temperatures, biological, chemical and radiological conditions of the reservoir.

Cooperating in the formulation of the program are the Arkansas Game and Fish Commission, the Arkansas Pollution Control Commission, the Arkansas State Board of

Health, the U.S. Corps of Engineers and the Department of Interior Bureau of Fish and Wildlife.

During initial phases of the program, Little Rock University will carry o[illegible]ll phases of gathering data. A sampling pattern has been set up to guide the data gatherers in their program to obtain statistical data on a number of significant characteristics of the reservoir.

The sampling pattern will also serve as an aid in developing representative background conditions on fish population, abundance of fish food, dispersion of chemical within the reservoir, and background information on the radiation count of the reservoir water and ecology.

In order to provide a comparison, testing will be carried out five years before the plant begins operation and will continue another five years after startup.

The wastes generated by the Russellville Nuclear Unit are based on the design with one percent defective fuel cladding. On this basis the reactivity concentration of fission products excluding tritium in the circulation of water discharge lines is estimated to be 4.4 times 10^{-8} microcuries per cc above background.

This compares with 2 X 10^{-9} to 3 X 10^{-10} microcuries per cc measured at Yankee Rowe and 2 X 10^{-7} to

3×10^{-10} at Indian Point.

Assuming that all the tritium diffuses through the fuel cladding, the tritium concentration in the circulating water discharge is estimated to be 1.7 times 10^{-5} microcuries per cc above background.

This compares with 4.6×10^{-6} microcuries per cc found in Yankee Rowe and 9.6×10^{-7} microcuries per cc at Indian Point.

In summary, the calculated discharge for the Russellville Unit will be well within the limits set by the AEC at 10 CFR 20.

CHAIRMAN WELLS: The Board has no further questions on that particular subject. Would you like to proceed, Mr. Jewell.

MR. JEWELL: Thank you, sir.

The Board has asked for information about the experience of the designer and constructor. Has Arkansas Power and Light Company selected a constructor for this plant?

MR. HOLMES: No, we have not. Bechtel Corporation has been retained to provide construction management on the job. And our contract with Bechtel gives us the option to require Bechtel to do the construction. We have not, however, decided whether Bechtel or somebody else will be the constructor.

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MR. JEWELL: Will you please give us a brief summary of the experience of Bechtel Corporation with pre-stressed concrete vessels of this type of your containment structure?

MR.HOLMES: Bechtel participated in the development of nuclear power plants since 1949; nuclear projects utilizing both tension concrete containment structures designed by Bechtel and licensed by AEC for construction include Turkey Point No. 3 and 4, Palisades, Point Beach 1 and 2, Oconee 1, 2 and 3 Units, and Rancho Seco Unit 1.

The following consultants with international reputations in their respective fields assisted Bechtel engineers in the formulation of all criteria and advised on methods of analysis, and reviewed final designs and specifications:

T. Y. Lin, Kulka, Yang and Associates, pre-stressed concrete consultants, Edward L. Wilson, shell analysis consultant, Kenneth D. Buchert, thin shell analysis consultant, A. E. Mattock, consultant for concrete shear criteria. These consultants are also available for consultation and review of the Russellville containment.

The project design group carrying out the detailed design of the Russellville containment structure has the inherent advantage of accumulated experience for previous designs. They can incorporate in the design the

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experience on the vessels presently being constructed.

Before the Russellville Unit construction is complete, eight practically identical containments will have been built and tested.

Bechtel's present staff designing and supervising the Russellville project include personell who have broad experience in pre-stressed concrete structural design including participation in Palisades, Point Beach and Turkey Point designs.

One member of the staff was associated for three years with research and development, design, construction and testing of pre-stressed concrete structures at design pressures 30 times higher than that for the Russellville containment.

Several members of the design team are active on national technical committees related to pre-stressed concrete designs and have offered articles and reports on the subject in national and international publications.

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MR. BRIGGS: When you talk about pre-stressed concrete design, are you talking about pre-stressed post-tensioned concrete design, or are you talking about other types of pre-stressed concrete design?

MR. HOLMES: Yes, pre-stressed, post-tensioned.

MR. BRIGGS: Does this experience involve actually building and testing units, or have they not yet built and tested such units?

MR. HOLMES: May I call on Mr. Howard Wahl of Bechtel to answer this, please?

MR. WAHL: None of the containment structures under construction have been tested as yet. About two weeks ago the last of the concrete was placed in the Palisades containment, so that structure is ready for post-tensioning; and this will begin in the middle of December.

MR. BRIGGS: Thank you.

CHAIRMAN WELLS: I think you can proceed to the next item.

MR. ENGELHARDT: Mr. Chairman, might it be appropriate at this juncture for me to make a stateent regarding the matter of the technical qualifications of the applicant, and who is considered to be the applicant? This particular answer to the question raises this point. And as you may recall, during the pre-hearing conference

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a member of the Board did inquire as to what the Staff considered in determining the technical qualifications of the applicant. And at that time I indicated that I would be prepared to make a statement with respect to who is included in the term "applicant."

CHAIRMAN WELLS: This is the correct place, it seems to me.

MR. ENGELHARDT: It appears to me to fit into the record as it is developing. And with the Board's permission I would like to make such a statement for the record.

CHAIRMAN WELLS: Please do.

MR. ENGELHARDT: During the pre-hearing conference the Board requested the Staff to present a statement on the point of whether the technical qualifications of the applicant -- which is one of the issues in this hearing -- includes qualifications of the applicant alone or includes qualifications of the applicant together with its principal contractors, employees, agents, and so forth.

It is the staff's interpretation that the finding of technical qualifications of the applicant to design and construct the facility may include the competence of its principal contractors. This interpretation is based on a survey of previous initial decisions of the Atomic Safety and Licensing Boards, which have become final decisions of the Atomic Energy Commission upon the expiration of the

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established review period.

The staff, as in this case, has always relied upon the expertise of an applicant's contractors to establish technical qualifications. Normally part or all of the components of a facility are furnished by a manufacturer, and an architect-engineering firm contributes its expertise to the design and construction of the facility.

In a 1965 case a Board specifically raised the question of whether such reliance was appropriate, and resolved the question in the affirmative. In the initial decision in the matter of Niagara-Mohawk Power Corporation the Board found -- and here I quote:

> "Discussions at some length revolved about the need for evaluating the technical capabilities of the principal subcontractors, particularly the nuclear contractor. Questions were resolved to focus upon the applicant who was specified in the issue, who seeks the license, who recognizes and assumes all primary responsibility for carrying out each and all of the representations and obligations which constitute the basis for granting the license. The Commission looks only to the licensee for compliance with the license requirements. The applicant, a corporation, can and must act through its officers, employees and agents, and duly constituted representa-

tives. Its contractors are among those. But its responsibility as the principal and the licensee remains undiminished. Upon finding the applicant to be an experienced and financially responsible, large utility corporation which has committed its resources, adequate funds and a competent management and operating organization to carry out this project, and absent evidence of deficiencies in the capabilities of its chosen agents, the qualification inquiry under these issues is ended."

Since then the question has not been raised until recently, although several cases have been decided in which the applicant utility was without previous experience with nuclear power plants, and reliance upon its technical competence included those of its principal contractors.

I think this statement would indicate that it appears clear from precedent that the Commission has essentially adopted the position which I have stated; namely, the contractors of an applicant, as well as the applicant itself, must be looked to to determine the issue of the financial qualification of the applicant to design and to construct a nuclear facility.

Mr. Chairman, may I make a correction? I think I stated "financial qualifications" when I meant "technical qualifications" in my closing sentence.

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CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

Mr.Briggs had noticed that also.

Thank you very much for that statement.

Would you like to proceed, Mr. Jewell?

MR. JEWELL: Mr. Holmes, would you please describe the test programs for tendon anchors and liner plate anchors, and give the schedule for completion of those tests?

MR. HOLMES: Anchorages of tendons will develop the minimum guaranteed ultimate strength of the pre-stressing tendon steel. Our specifications require that the post-tensioning system supplier perform tests on the tendon anchorage systems up to the yield point of the tendon steel in order to verify this capability for the tendon system selected.

Assurance will be provided that the components in the tests can be correlated with components to be produced for the containment tendons. This will be done by analysis which define the critical production tolerances and material characteristics, and by assuring that they are sufficiently covered in the tendon tests and any component tests.

All material will be identified to the extent necessary for this specific application by conformance to ASTM specifications, to industry standards for similar detail, or by measured chemical and physical properties necessary to define the materials to a degree equivalent to that given

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by such specifications or standards.

Although the testing of the tendons and anchorage systems does in fact give sufficient assurance that the hardware will develop the ultimate strength of the wire, an attempt will also be made to analytically determine deflections of the tendon hardware, bearing plate, and concrete for the test conditions. Such predictions will then be compared with actual deflections measured during the test. The analysis will be done by aximetric finite element techniques, and consideration will be given to the non-linear stress-strain properties for both the concrete and steel.

Selection of the post-tensioning system supplier is scheduled not later than December 1st, 1968.

After the supplier is selected a period of approximately three to four months is required to conduct the tests and finalize the analysis.

Now I will talk about the liner plate anchorages.

A test program has been worked out to substantiate the analytical constants used in evaluating the adequacy of the liner plate anchorage systems. Various configurations of liner plant anchorage will be cast in concrete to provide composite test specimens as listed:

(1) Two tests with 4 - 12 fillet welds and two tests with angle vertical leg down.

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possible to obtain the following information: spring constants, linear and non-linear; ultimate load capacity; ultimate displacement capacity; total energy available in the system.

This program is expected to be completed by March 1st, 1969. This date is prior to the time when the liner plate anchorages will be installed.

CHAIRMAN WELLS: Mr. Jewell, you may proceed to the next question, if you will.

MR. JEWELL: Mr. Holmes, will you please give us a comparison between training in an operating plant and training in a simulator?

MR. HOLMES: The equivalents between training in operating plants compared with training on simulators can best be judged by a comparison of potential levels of knowledge and variety of experiences gained by the trainees in the two training environments.

Operating training using a simulator, as proposed by B&W is a combination of real nuclear power plant and reactor operating experience and simulator operating experience. Trainees spend a period of time at an operating nuclear plant observing all phases of the operation performed by that plant's personnel. They are not required to actually operate the controls of this plant. They are, however, required to get extensive operating experience on

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some kind of real reactors.

In the case of AP&L this is B&W's Lynchburg pool reactor.

The pool reactor experience enables the trainees to manipulate the controls of an actual reactor and to get the feel of taking the reactor critical and changing power levels. During this training they will also gain an understanding of reactor physics, nuclear instrumentation and health physics, and conduct operating routines not feasible on a commercial power reactor.

The simulator plays a key role by providing power plant operating experience in excess to that possible on a real plant. Only a limited variety of operating experiences are available on a real power reactor. These are limited primarily to start-up, shut-down, and load changes. Questions of nuclear safety and procedural restrictions and the desire for high plant availability prevent more extensive training. Simulator training, on the other hand, provides a complete range of training in both normal and abnormal operating situations without availability restrictions.

The large variety of operating situations and unrestricted availability makes possible repeated drill until the trainee reaches the desired level of proficiency in all phases of plant operations.

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The limited variety of operations and restricted availability in the real plant situation severely limits drill. This forces the trainee to learn most operations by study of written procedures without the opportunity for actual practice. Thus the trainee's proficiency in executing a given operation will not be known until the need for the operation arises in the real plant under emergency conditions.

The B&W simulator is designed to provide maximum realism and matches the response of B&W nuclear steam supply systems being built today. Those power reactors being utilized for operator training are in general of the earlier generation and bear little resemblance to today's large plants.

In summation, the simulator experience is one portion of an extensive training program. The experience level of a graduate of this program will exceed that of an operator following a training program based solely on plant operating experience. Arkansas Power and Light has already decided to utilize the B&W combination of training in real plants plus simulators because of the advantages which I have given.

MR. ENGELHARDT: Mr. Chairman, may I offer a supplement to that statement before the Board raises its questions?

CHAIRMAN WELLS: Yes, indeed.

MR. ENGELHARDT: Mr. Schwencer, what credit does

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But the statement made in the summary, I believe, was that there would be training in an actual plant or simulator training.

Can I assume that your statement says that there will be training on simulators and training in an actual operating plant, rather than "or" training?

MR. HOLMES: Yes, sir. To the extent of what I have said, training in the plant would be of the observation type and learning by reading and studying procedure manuals.

MR. BRIGGS: At an actual nuclear power plant?

MR. HOLMES: Yes, sir.

CHAIRMAN WELLS: Very well. Mr. Jewell, would you continue?

MR. JEWELL: Has the Arkansas Power and Light Company given serious consideration to substituting charcoal absorbers for the proposed chemical spray system for iodine removal?

MR. HOLMES: Arkansas Power and Light believes that there is enough conservatism in iodine release estimates and iodine control through containment leakage ratios and dispersioning effects of the atmosphere to say that iodine removal equipment is not needed. However the AEC staff by imposing additional conservative factors in dose calculations set the requirements for iodine removal.

Our belief is that the chemical means of iodine

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

IN THE MATTER OF)
)
ARKANSAS POWER & LIGHT COMPANY) Docket No. 50-313
)
(Russellville Nuclear Unit))

EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS
DONALD A. NITTI
CHEMISTRY DEPARTMENT
RESEARCH AND DEVELOPMENT DIVISION
THE BABCOCK & WILCOX COMPANY

1. My name is Donald A. Nitti. My residence address is 5301 Fort Avenue, Lynchburg, Virginia, 24502. I am employed by the Research and Development Division of The Babcock & Wilcox Company as a research engineer at the Company's Nuclear Development Center in Lynchburg, Virginia.

2. I graduated from the City College of New York in 1956 with a Bachelor of Chemical Engineering degree.

3. I then worked for the Process Plants Division of the Foster Wheeler Corporation in New York City for three

the AEC Regulatory Staff allow for simulator training?

MR. SCHWENCER: This is a relatively new training device, and it is not considered by the Regulatory Staff as a complete substitute for actual reactor operating experience. We are, however, allowing its use in the training of reactor operators, as in the case here in Russellville.

In allowing this use of reactor simulator training, we eliminate the requirement that two of the trainee's required ten reactor start-ups e performed at a comparable -- and I underline the word "comparable" -- be at the plant. He may, instead, perform all ten start-ups at a test, research, or other power reactor facility. Upon successful completion of such a training program, utilizing reactor simulator training, the AEC would normally grant reactor operator trainees elibility to take the AEC reactor operator examination.

CHAIRMAN WELLS: Thank you.

MR. BRIGGS: The question arose out of a statement I believe that was made in the applicant's summary -- and before we go to that statement I will say that I completely agree that simulator training is very good experience, and in some cases extensive training on the simulator, supplmented by shorter training periods in a reactor plant, is far superior to somewhat longer training in an actual operating plant.

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removal proposed is adequate to bring the dose of iodine to acceptable limits using the AEC Staff's calculation methods. Further, because of known information on the sodium thiosulphate chemical iodine removal system, we are confident that this system will remove iodine from containment atmosphere, and that this system is the most economical.

However because of the AEC Staff's requirements, we are providing space to install charcoal filters if they are required of us.

DR. QUARLES: I think this is a good place to get in a question.

If you use the spray I gather it's very important that the pH be kept fairly high, definitely alkaline. What provisions are made for monitoring the pH and maintaining the necessary value.

MR. JEWELL: Mr. Chairman, it now appears that we need to call upon one of our reserve witnesses. He has not been sworn or qualified. I would like to call Mr. D. A. Nitti as a witness, please. And may we have him sworn?

CHAIRMAN WELLS: Yes, you certainly may.

Whereupon,

DONALD A. NITTI

was called as a witness for and on behalf of the Applicant and, having been first duly sworn by the Chairman, was examined and testified as follows:

(2) Two tests with 6 - 12 fillet welds and two tests with the angle vertical leg down.

(3) Two tests, one with 4 - 12 fillet welds and one with 6 - 12 fillet welds, but with rotation restraining block removed.

The materials used in the test specimen will be similar to those that will be used in the actual structure. The steel plate will be ASTM A-442 or A-36, depending on availability. The concrete mix design will be nearly identical to that which will furnished for the structure. The material supplier has not been selected as yet; however it is felt that both compression strength and modulus of elasticity can be well approximated in the test specimens with local materials.

Physical properties of the steel and concrete will be obtained as follows:

Concrete: Six test cylinders will be required, two at seven days, two at the predicted date of 5000 psi strength date, and two at twenty-eight days. Compression strength test at seven days, test date and twenty-eight days will be done in accordance with ASTM C-39. Modulus of elasticity tests at twenty-eight days will be done in accordance with ASTM C-469.

The results of the test will be a series of load versus displacement plots. From these plots it will be months.

4. From September of 1956 through December 1957, I attended the Graduate School at the University of Cincinnati. My major field was Chemical Engineering.

5. In 1958, I was in the United States Army, Corps of Engineers at Fort Belvoir, Virginia. I attended the Engineer Officers' Basic Course, and then I was assigned to the Engineer Center Regiment as a Company Executive Officer.

6. In 1959, I accepted a position with the Atomic Energy Division of The Babcock & Wilcox Company, in Lynchburg, Virginia. I performed studies relating to reactor system waste disposal, to the growth of radioactive corrosion products in the primary coolant system of reactor plants, and to fuel slurries in liquid bismuth for the Liquid Metal Fuel Reactor Experiment. I also conducted fission product tracer experiments associated with the Gas Suspension Coolant Development Program.

7. In 1961, after completing my Masters thesis, for which I was awarded the Hochsteeder prize, I received my

Master of Science degree from the University of Cincinnati.

8. Also in 1961, I was loaned to Ebasco Services, Inc., in New York, New York, to assist with the design of the USAEC's Advanced Test Reactor. I was responsible for the design and analysis of chemical process systems and for the complete environmental hazards analysis.

9. In 1963, I returned to The Babcock & Wilcox Company in Lynchburg, Virginia as a research engineer in the Process Development Section of the Chemistry Department at the Research and Development Division's Nuclear Development Center. I was responsible for various aspects of the design, evaluation, and erection of pilot plants devoted to the processing of recycle nuclear fuel, and for process development and improvement.

10. In 1967, I was loaned to the Nuclear Power Generation Department of The Babcock & Wilcox Company's Boiler Division. I was responsible for developing an engineered safeguard system for the removal of radioiodine from reactor building atmospheres. My

responsibilities included the valuation of the scope of the radioiodine problem bas l on current technology, the supervision of a study su contracted to Battelle Memorial Institute, the establ hing of performance criteria, the development of t calculational techniques for the system desi and evaluation, and the identification of areas re iring additional research and development.

11. Later in 1967, I returned to t Nuclear Development Center. I am presently the pr ect engineer responsible for B&W's research and develop nt program on the removal of radioiodine by chem al spray systems.

12. My additional activities inclu membership in the American Institute of Chemical ngineers, Sigma Xi, and Phi Lambda Upsilon.

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MR. JEWELL: I believe copies of that statement have been distributed to members of the Board and to the Staff.

CHAIRMAN WELLS: I wonder if Mr. Nitti understood the question.

Do you wish it repeated?

MR. NITTI: Yes, please.

DR. QUARLES: The question is: How do you monitor the pH of this installation, and how do you maintain it to the necessary degree?

MR. NITTI: I would like to answer one other question that I think was originally in your original question; and that was a requirement that the solution be maintained fairly alkaline, implying that it had to be maintained highly alkaline.

As you know, we do have a research and development program in this area, and we have observed that although the pH does drop upon irradiation we don't encounter any problems -- as a matter of fact we have not encountered any problems whatsoever as far as the solutions, or any properties of the solutions that might jeopardize its performance in a post-accident environment. But we do know that if the solution were allowed to get to too low a pH, this is below 7, in a final condition, or somewhere below 7, it would be possible under certain conditions to expect problems. But we would endeavor to select our concentration in such a way that the pH would never be able to get too acid, or to a condition where the pH dropped to below 7. That would be our first intent as far as the solution composition.

Now to answer your other question: I believe -- and I would like to check first.

(Pause)

There are provisions to take manual samples of the solution to do an analysis to measure the pH.

The second point was about addition of hydroxide or some solution to maintain the pH, and at present there is no intended hardware to accomplish this; primarily for two reasons: one is, we really don't expect to design the solution composition to require this unless the test results did show that there were a possibility of this condition arising.

Secondly, if such a condition were to arise we are confident that it would be somewhere in the neighborhood of twenty to forty days after the accident. We are confident right now of that condition, and probably will never be able to get to a pH below 7 when we finally decide on the optimum solution.

DR. QUARLES: YOu answered my two questions. The first one was from reading the two documents that were distributed this morning. I am confused.

wb17

In the one, Applicant's response to the Board's question on iodine removal, I quote -- it's on page 2:

"This solution has an acid pH--" which would be below 7. "-- is unstable in several respects and is generally not acceptable."

Then referring to the Staff's document on the same subject, on page 3 it discusses test results when the pH was around 9 -- 9.2, implying that this is an acceptable level, and then as you go below this it is then perhaps not as acceptable.

Is my interpretation of these right, that it is most definitely alkaline?

MR. NITTI: Yes, your interpretation is correct.

There might be a slight misunderstanding as to the magnitude of the alkalinity required. The 9.2 is a convenient pH on the sodium hydroxide-boric acid buffer curve. Now if you don't have any buffering capacity in the solution, in the boric acid -- which we refer to there as acid thiosulphate solution, when we start to generate acid through radiolysis; or generate hydrogen ions, more correctly; what happens is, the pH then tends to drop rather easily after rather short irradiation, or radiation of 10^7 or 2×10^7, or something like that.

But if you consider the buffering effect when you add sodium hydroxide you can tolerate a considerable amount

wb18

of hydrogen ion, and we are not changing the pH very much at all.

Now ' question is exactly where you want to start on this curve. And the pH buffer for boric acid - sodium hy' ..ide is pretty much between 9 and 10. So that if you have any buffering capacity you start off somewhere around 9. But the idea is not to fall very far off the plateau of the buffering curve. You don't ever want necessarily -- you want to design the solution do you don't lose all your pH control.

DR. QUARLES: Thank you.

MR. ENGELHARDT: Mr. Chairman, we have a couple of matters that I think might supplement this record with respect to this matter of iodine spray, if this is the right time to insert it in the record.

CHAIRMAN WELLS: Let me learn from my colleagues as to their wish.

I am inclined to say we ought to take a recess now. We have been in session for an hour and a half. And perhaps after that recess we can continue and try to finish the questions on the iodine removal, or finish the other questions. But we will confer on that during the recess.

Is this agreeable, to have the recess now?

DR. QUARLES: Yes.

MR. ENGELHARDT: I just want to clarify one point. The answers we have in mind here are in direct response to

wb19

questions raised by members of the Board at the pre-hearing conference. They are not supplementing anything tht has been said, but are responses in this same area of questions raised by the Board.

CHAIRMAN WELLS: Very well. We will receive these answers after the recess.

We will reconvene at four-ten.

(Recess)

CHAIRMAN WELLS: The hearing will come to order.

Just before the recess, Mr. Engelhardt, I believe you were going to give answers to some aspects of the questions the Board had asked.

MR. ENGELHARDT: Yes, sir.

Mr. Long, what iodine removal factor for the containment spray system is required for the Russellville plant?

MR. LONG: Basing our analytical model on the atmospheric diffusion of the gaseous release from the containment and taking a conservative assumption that a valley located some south-southeast of the site restricted the movement to less than the 22.5 degree sector which we normally use for a normal site, we calculate that a factor of 2.9 will be needed in order to meet the 300 rem dose suggested by the guideline 10 CFR Part 100.

We further stated in our Safety Evaluation on page 29, however, our evaluation of the spray system proposed for this plant using the model which is described in an exhibit we put into the record earlier this morning, that a factor of 4.1 would be obtained for this spray system based on our model.

MR. ENGELHARDT: May I go on with the next question?

CHAIRMAN WELLS: Does it relate to this same general subject?

eb2

MR. ENGELHARDT: Yes, sir.

CHAIRMAN WELLS: Yes, you may go on.

MR. ENGELHARDT: Mr. Long, what research and development is required for the charcoal absorption system, or what evidence do we have that such a system can be provided for the Russellville plant that meets the required iodine removal factor?

MR. LONG: At this point I would like to ask the Board if they would wish for us to supply copies of the document I am about to make reference to, which is our Supplemental Safety Evaluation for the charcoal filter assemblies that were proposed for the Turkey Point plant of Florida Power and Light Corporation?

We just went through a hearing on this last week, as a matter of fact.

CHAIRMAN WELLS: I suppose, Mr. Long, you are surely the world's expert in this. We would be very happy to have copies.

MR. ENGELHARDT: Mr. Chairman, would you like me to identify these as Staff Exhibit 4, or would you prefer to have them, since they are matters of public record and available to the public upon request, just pass them out and not identify them as part of the record?

CHAIRMAN WELLS: I am inclined to think the latter. While they are interesting in connection with this hearing, eb3 they are not necessarily germane.

MR. LONG: To go on with the answer then, as I remember the question propounded by the Board at the pre-hearing conference, it was in relation to what assurance do we have that a charcoal filter system could be installed in this facility if it were deemed necessary at some time in the future.

The system I am speaking of, which related to the Turkey Point plant, is a charcoal filter system only it has no cooling devices in it. It is strictly for the removal of iodine.

Using this as a typical system that could be installed in the Arkansas plant if it were required, we calculated, based on a review of that application, a factor of three and a half for the reduction of iodine within the containment. This puts it in perspective, I believe, with the question originally asked.

CHAIRMAN WELLS: No questions at this point.

MR. ENGELHARDT: That completes this phase of the answers. I believe the applicant has some additional answers to respond to, and then the staff has some further questions to respond to, all on different points other than this matter we were just discussing.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt.

I assume now that we have covered the responses to

eb4

the questions that we have raised in the pre-hearing conference with respect to the iodine removal. Am I correct?

MR. JEWELL: Yes, sir, the applicant has covered all of those questions.

CHAIRMAN WELLS: Let me see if my colleagues on the Board have any questions in connection with the material that you were good enough to supply to us last night.

Dr. Quarles says he has already asked his.

Mr. Briggs, do you have any questions?

MR. BRIGGS: I have a few questions here concerning the applicant's response, the printed response to the questions this morning.

As a matter of clarification on page 3 in the second paragraph there is a statement concerning an NSPP run in accident conditions and the measure of iodine half-life was 31 seconds. It says:

> "These data have been scaled to the Russellville design and they result in an iodine half-life of 23 seconds with the full spray installed capacity operating."

How were these data scaled from the NSPP conditions to the Russellville condition?

MR. NITTI: Mr. Briggs, they were scaled using the analytical model basically developed by Griffiths and generally used in most of these calculations. What we did was

eb5

we took the ratio of-- The expression is the same expression used on page 4 of the staff's testimony relating to this subject and also the same expression used in Section 14 of the PSAR for calculating the iodine removal constant.

What we did is took the parameters, the physical parameters of geometry and flow rate and vertical fall distance of the droplets under ambient conditions in the respective facilities and took the ratios of these parameters and used it to predict the value, the ratio of lambdas we would expect and from that we evaluated the lambdas we would expect in the Russellville containment building.

MR. BRIGGS: And the height of fall in the nuclear safety pilot plant is how great?

MR. NITTI: It is 15 feet, give or take a couple of feet.

MR. BRIGGS: And the height in the Russellville plant?

MR. NITTI: 99 feet.

MR. BRIGGS: And so the ratio of each that you used there was the ratio of 15 to 99, or 99 to 15?

MR. NITTI: That's right.

MR. BRIGGS: In making this extrapolation then, you assumed that the drop diameters, the drop spectrum was the same in both cases? There is a value D here for drop diameter. You assumed that the drops were the same in the nuclear safety

eb6

pilot plant as in the Russellville plant?

MR. NITTI: Actually, what we did in our calculation was an attempt to very be very conservative. We feel that our calculations are conservative by about a factor of 4. We use a uniform drop size of 1,000 microns in our calculations and we used it as uniform all the way down for the 90 feet. And we don't take credit for defining droplets in the spectrum and we don't expect to have any droplets to be much larger than 1,000 microns.

MR. BRIGGS: But in the extrapolation from the NSPP case to the Russellville case, the extrapolation which resulted in this iodine halflife of 23 seconds, you considered that the drop diameter remained the same; is that right?

MR. NITTI: Yes. Actually what we did in that extrapolation is that particular test is based on the same nozzle that we are considering using in the Russellville containment building and therefore actually the drop diameters as such would cancel out of the expression. We assume the same spectrum and we assume it is constant all the way down.

MR. BRIGGS: So the important numbers came out to be the flow rate and the containment volume on the drop fall height.

MR. NITTI: Yes.

There were several other conservative corrections made. One is we corrected for the difference in density of the atmospheres so that changes the terminal velocity somewhat and the other conservative thing is in the NSPP the droplets are relatively close to the nozzle and travelling somewhat faster than terminal velocity and this would tend to give a slower removal rate than we would expect from droplets falling at terminal velocity. So we don't try to take any credit on that type of thing.

MR. BRIGGS: Well, in the same paragraph it says:

> "The iodine half-life reported in the PSAR is 90 seconds full capacity and 180 seconds at half capacity."

One gets the 90 seconds and 180 seconds by including these conservative assumptions with respect to drop size, is that right?

MR. NITTI: Yes, the 90 seconds and the 180 seconds are based on the fact that there is half the flow in those two numbers. But they are also based on the calculations as presented in Section 14 of the PSAR which assumes uniform drop size of 1,000 microns.

MR. BRIGGS. You have mentioned the uniform drop size of 1,000 microns. What actual drop size do you expect for the Russellville spray system?

MR. NITTI: That particular nozzle has a mass-mean diameter reported to be 700 microns and whenever we have taken--

eb8

If you assume that the drop size distribution is normal and you try to break it up into drop size groups and make the same calculations and then sum up the total mass transfer for each drop size group, it appears that you increase the overall mass transfer capability due to the smaller drops in the spe trum.

So therefore we just assumed that 1,000 microns would be conservative and in fact, we think this is substantiated by the fact that we are reporting one-hundred eighty or ninety seconds and NSPP under similar conditions is measuring a much faster removal rate. And I think if we refine the calculations to be less conservative and more technically correct, we would agree much closer to the Oak Ridge results nad in fact, Oak Ridge and the people at CSE conducting similar experiments feel that their mathematical models are much closer than this agreement would show.

MR. BRIGGS: In any of the tests that have been run so far in spray systems, is there an indication that the drop size changes with fall height?

You mentioned in here the nuclear safety pilot plant and then the containment test system, or whatever the name is of it, Seaforth, I believe, having a greater fall height. Is there any information on what happens to drop size as a function of fall height?

MR. NITTI: I don't believe that these experiments in themselves are capable of resolving small changes in mass

eb9

transfer rates that might be associated with small changes in drop size, but from discussions with manufacturers and from our own analyses we are of the opinion that the drop size will not change much in this environment.

And I would like to point out a few things.

If you assume that the drops are 1,000 microns and you assume that the area they are covering in this building and the flow rate, you would come to the conclusion that these drops are about -- have a population density of about 100 drops per liter, and this keeps these drops quite far apart.

And if you then say, well, your drop density increases with population density, you would assume that you did get some more collision frequency but then, when you take a look at what happens when a 1,000-micron drop collides with a 100-micron drop, the net change is very small in the overall effect. The loss of the 100-micron drop doesn't hurt you much and the increase in size of the 1,000-midron drop is negligible.

So what really changes the spectrum appreciably is collision of drops of equal size or approaching equal size and these drops are falling at about the same velocity so from our analyses we believe that collision frequency is in general a minor problem or no problem, actually. We would not expect to see any.

And in discussing spray nozzle tests with

eb10

manufacturers, they pretty much told us bluntly that the drop size spectrum, no matter how far you get from the nozzle, the measured drop size spectrum would be the same.

Does that answer your question?

MR. BRIGGS: This is based on tests?

MR. NITTI: I think it is based on experience along the same lines that I had outlines as far as the problems of changing the drop size distribution though I cannot say for a fact that this is based on a specific test that was reported or was not reported.

MR. BRIGGS: On page 4 under non-reactive iodine in the second paragraph it states, down near the bottom of the paragraph that certain results have been obtained, and then it says:

> "Some of these experiments have observed greater than 5 percent non-removable iodine. However, these experiments were conducted under conditions that are not applicable to the PWR accident environment."

What were the conditions that these were done under and why do they not apply to the PWR accident?

MR. NITTI: In general, I would say that the values above a few percent, 2 or 3 percent, were obtained under conditions which were oriented more toward gas cooled reactors that had an appreciable amount of CO_2 and no steam and

eb11

conditions of that type or were PWR-type conditions but altered to try to produce methyl-iodide by injecting large amounts of methane or organic vapors or something of that nature. And we don't feel that they are really representative of PWR accident conditions.

MR. BRIGGS: So you feel then that 5 percent is really quite a conservative value for the organic iodine that one would have in the containment?

MR. NITTI: Yes. We don't believe there is any experimental evidence that measures the value as high as 5 percent under any conditions that approximate a loss-of-coolant accident condition in a PWR.

MR. BRIGGS: Well, then, is it reasonable to assume or to state that the dose that would be received at the boundaries of the exclusion area, the two-hour dose, or the dose that one would receive at greater distances over longer periods of time depends upon, one, the iodine that cannot react with the solution which is 5 percent in your case and I believe 10 percent in the staff Analysis and, two, the rate at which you can wash the iodine out of the containment?

This then also depends on how rapidly the system begins to operate, the iodine removal system.

I was a bit confused by the table 1411 in the PSAR. That is the sensitivity analysis. I think I know what was done but I would like to make sure about this.

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In the cases 1, 2, and 3 it shows drop size, fall velocity, temperature, pressure, iodine removal time constant, and then dose. And then under "Remarks" for case No. 1 it says:

"Operation of the reactor building spray system at maximum building temperature and pressure."

-- when one got a dose.

And then it says under case 2:

"Operation of the reactor building spray system after partial cooling, about one hour."

And then under 3 it says:

"Operation of the reactor building spray system after cooling at ambient condition."

-- which then must be some hours later.

Are those remarks really not what is meant? In other words this implies to me that in case 1, the spray started immediately; in case 2, the spray started an hour later; in case 3, the spray started several hours later.

MR. NITTI: No, that is not the interpretation that was intended.

What was intended was to show that the atmosphere, the density of the atmosphere in the containment building affects the dose that we measure and we made our evaluation at the peak temperature and pressure, but as the pressure decays, our mass transfer rate should increase, at least in eb 13

our calculations.

We made very conservative assumptions as to the atmosphere composition to slow down the mass transfer. In other words, we assumed that the atmosphere was air. Air is denser than steam-air mixtures and has a lower mass transport -- a higher mass transport existing. So these calculations were based on air.

Now there are more-- ORNL has developed a computer code that calculates the physical properties of the atmosphere more precisely and if we use the same code, these numbers would be higher removal rates because of the lower density of the atmosphere and we would have lower doses. In other words, the doses that we would measure would be closer to the 54 as shown in case 3 as compared to the value of 53. So this is trying to show the fact that we were conservative in evaluating the mass transfer at the peak temperature and pressure.

MR. BRIGGS: I'm afraid the table didn't make things very clear for the reader, at least one reader.

What was done here was to assume in case 1 that the temperature and the pressure in the containment were 286 degrees and 59 psi, and one then gets a velocity of deposition of 4.84 centimeters per second. Is that right?

BLOOM 25
rms 1

And with that velocity deposition you get an iodine dose of so many rem.

And for case three in fact you calculated that if the temperature were only 100 degrees and the pressure is zero psi you would get a deposition rate of 11.55. And you assumed that at zero time the sprays went on and the deposition rate was 11.55 and then you got an iodine dose of so many rem.

Well, I think it might help to change the remarks a little bit on the table.

MR. ENGELHARDT: Mr. Chairman, on the answer that was given just a few moments ago Mr. Long has a clarification point -- well maybe not clarification -- but which might be useful at this point before we go on to something else.

CHAIRMAN WELLS: Mr. Long, we would be glad to hear it.

MR. LONG: Again, I will make use of my Turkey Point supplemental safety evaluation that was handed out.

The question came up about the experimental data that they felt were not representative of pressurized water conditions and you will find attached to the supplemental safety evaluation an appendix of literature including the conditions for various tests and the amount of organic iodine measured. The documents, or at least some of the tests that they may be referring to, I am sure are included in this list.

Secondly, starting on page 10 of that supplemental safety evaluation we present our reasons for using 10 percent non-removable iodine in the containment. This is our justification for the use of a 10 percent value.

CHAIRMAN WELLS: Thank you, Mr. Long.

MR. BRIGGS: In the staff analysis or the staff's memo on iodine removal by sprays there is some discussion of the drop size spectrum. Could you tell me whether the size spectrum that you use in your calculations is essentially the same as the applicant uses in his?

MR. LONG: As I understand it, no it is not.

MR. BRIGGS: The applicant uses an average size of a thousand microns. And what does the staff use for its analysis?

MR. LONG: We start off with -- I am not sure of the actual micron size number. But then we apply a 10 percent value -- I believe it is explained. I am trying to find it here.

MR. BRIGGS: It is on page five at the bottom of the page.

MR. LONG: We use a 10 percent larger -- this is the geometric mean drop size diameter. And then we apply on top of that a standard deviation of one and a half which the applicant does not do. And because of this we feel we are conservative by at least a factor of two or more over what the applicant has done.

MR. BRIGGS: Using these numbers what -- oh, let's see -- what iodine half-life would you calculate for this system. Do you happen to recall that number?

MR. LONG: I am afraid I can't give you that answer here directly. I cannot name it. But I would have to call back to the office.

MR. BRIGGS: Well, now, let's see here. In making your calculations do you normally calculate these half-lives, or do you somehow translate those half-lives into maximum iodine reduction factors that you refer to on page one?

MR. LONG: I have seen calculations that have been made. We do actually calculate the lambda which was referred to earlier and then convert that to a dose reduction factor which is a matter of transposing it from the inside of the building out to the site boundary.

This would be, of course, the function of the reactor site.

MR. BRIGGS: You refer to experimental results in section six here on page six. There is reference to small scale experiments in the NSPP facility and then to the tests in the CSE installation at Batelle Northwest. In the CSE installation I believe it indicates that during the final spray period the observed half-lives for removal of elemental iodine were considerably longer than the calculated values, or were longer than the calculated values.

Do you know whether they were sufficiently long that one would be concerned about the performance of a spray system for containment like Russellville if the same sort of conditions prevailed?

MR. LONG: Offhand, I cannot remember the numbers themselves. I have heard them presented. I would say no there is no concern over this at this point. My understanding of the CSE experiments, particularly the one referenced here is that there is some question as to whether or not the lower portion of the containment which is compartmented may or may not play a part in trapping the volumes.

And thismay have caused erroneous measurements in the particular experiments. That is one of the explanations. The other is it may be the formation of organic iodines.

MR. BRIGGS: Are the lower compartments, or the lower parts, of the Russellville Containment -- and these are the large containments -- compartmented in a similar way? In other words, they have steam generators and compartments for other types of equipment. Are they at all similar to the CSE?

MR. LONG: I am not sure I can say they are similar, but I would think that the compartmentation of he CSE is more definite than the reactor design because of the secondary shield and the way the primary cavity opens to the secondary shield area and then to the upper containment volume.

Although it is not that well connected that you could really say one is proportionate to the other.

MR. BRIGGS: Is there a continuing program of study at the CSE installation to try to resolve these questions?

MR. LONG: Yes, there is.

MR. BRIGGS: One other question. On the organic iodine it was indicated under item four here on page eight that the removal capability of sodium hydroxide could potentially be raised by the initial surface active reagents. What is the function of those reagents in removing organic iodine?

MR. LONG: I am afraid I can't answer that, for that is strictly a statement made by our people who prepared this statement that deal strictly with chemistry and that is not my field. And I cannot answer the question, sir.

MR. BRIGGS: Does the applicant's chemist happen to know?

MR. NITTI: Well, it is my undestanding, and I think I am correct, that the major problem is that methyl iodide is much less reactive than elemental iodine. And even though for Russellville we used sodium thiosulfate, there is some measurable removal rate and reaction rate with methyl iodide. The problem in general, as pointed out there, is more noticeable in sodium hydroxide in that the limiting factor to mass transfer is the resistance encountered in the liquid film transport and reaction within the drop, though the reaction mechanism is limited by processes within the drop.

And it is the opinion that if you can employ the use of a surface-active reagent that has a reactive group that can react with methyl iodide, you will eliminate the resistance, you will bypass the limiting step in the mass transfer rate, and therefore, have a surface reaction on the drop. And therefore you will greatly enhance the absorption rate of methyl iodide, or the apparent absorption rate.

MR. BRIGGS: The resistance within the drop being a limiting factor, is this the case of methyl iodide absorption?

sir.

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MR. BRIGGS: The resistance within the drop being a limiting factor, is this the case of methyl iodide absorption?

MR. NITTI: That's correct.

MR. BRIGGS: But not in the case of iodine absorption with thiosulfate? Is that right?

MR. NITTI: Well, it is definitely not the case with iodine absorption and especially for thiosulfate. With thiosulfate the reaction is strictly gas phase limited. As fast as you can get the iodine to the surface of the drop, the reaction occurs and iodine is absorbed.

MR. BRIGGS: But in none of the analyses does one depend upon removal of the organic iodine? Is that right?

end 25

MR. LONG: That is correct.

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CHAIRMAN WELLS: I believe, gentlemen, that completes the line of questioning that the Board had on the iodine removal. No others come to mind.

Let me restate what I said this morning: We are deeply grateful to both the applicant and the staff for responding to our requests, and for giving the documents and permitting us to have the opportunity to read them and, therefore, to amplify the record by the additional questions we have been able to put forward.

Mr. Jewell, would you care to proceed with the answers to the other questions?

MR. JEWELL: Mr. Holmes, the Board has indicated an interest in protection against flood. Would you please describe how the plant will be protected against an unlikely flood which might occur if the failure of the Ozark dam were superimposed upon the maximum flooding?

MR. HOLMES: The Dardanelle Dam is designed to discharge 900,000 cfs and hold the normal water level at 338 feet. The highest experienced flood had a peak flow of 683,000 cfs. The maximum probable flood flow is 1,500,000 cfs. A flood of the magnitude of the maximum probable flood would be forecast about five days before it arrives at the plant site. A maximum probable flood and the simultaneous complete failure of the upstream Ozark dam would result in a maximum elevation of 361 feet water level at the plant site.

wb2

The plant grade elevation is 353 feet. The maximum probable flood would be eight feet about plant grade for two to about five days.

During a maximum probable flood the plant will be shut down. The early forecast will provide ample time for precautionary measures which include the checking of critical equipment, power sources and fuel, and sealing off any opening in the flood protected buildings by waterproof doors and covers. Only a limited number of inspection and maintenance personnel will be required during flood.

Boat landings will be possible on the second floor of the administration building, and the roof will serve as an emergency helicopter landing platform.

The distance between the plant and the unflooded shore will never be more than two thousand feet.

The control room and diesel generators are located above flood level. The drive motors of critical service pumps will be mounted on flood protected floors. All other Class 1 equipment is located either above maximum probable flood level or protected by essentially watertight Class 1 structures.

The effect of maximum probable flood on the containment structure has been investigated. The containment is inherantly a leak-tight structure, since the 3'9" walls are pre-compressed due to pre-stressing, and the base slab

wb3

is nine feet thick. It is customary for concrete structures above grade, the exterior walls below grade will be treated with waterproof coating. As a secondary precaution a drainage system will be installed behind the containment liner plate up to flood level, and pressure will be released by this system.

Any seepage water collected by the liner plate drainage system will be gravity drained to the auxiliary building sump. The sump pumps will transfer the seepage water to the clean waste receiver tanks for emergency storage. Water stops will be installed in all construction joints below flood level.

The weight of the containment structure is about 100 million pounds. The bouyant force during maximum probable flood is about 30 million pounds. The factor of safety against floating is 3.3.

Wall thickness of the auxiliary building below flood level will be a minimum of two feet. All outside openings below 368 feet will be provided with water-tight doors of hatches. Water stops will be installed in each construction joint. If local seepage occurs through walls it will be collected by the auxiliary building floor drainage system which discharges into the auxiliary building's sump. The sump pumps will transfer the seepage water to the clean waste receiver tanks for emergency storage.

wb4

The weight of the auxiliary building is 77 million pounds, and the bouyant force is 44 million pounds, a factor of safety against floating of 1.72.

There will not be any cross connections between the drainage systems in the Class 1 structures and any other area of the plant below maximum expected flood level; therefore the flood water cannot enter the Class 1 structures by backing up through the drainage systems.

The diesel generator emergeny fuel tanks and the reinforced concrete compartment housing the c lean waste tanks will be anchored into foundation rock. Part of the intake structure which houses the service water pumps will be raised above flood level. Our investigations into the nature and characteristics of the maximum probable flood and the extensive precautions taken assures that in this very unlikely event the plant will be safety shut down and will not present any hazard to the public.

CHAIRMAN WELLS: Mr. Jewell, you may proceed to the next item.

MR. JEWELL: Mr. Chairman, that completes the answers to the questions which were directed by the Board to the applicant.

CHAIRMAN WELLS: Thank you very much.

Mr. Engelhardt?

MR. ENGELHARDT: Yes. We have two questions that

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were raised by the Board during the pre-hearing conference and a statement from myself regarding the matter that the Chairman expressed some interest in. So, if I may proceed, I would like to dispose of the two questions that remain outstanding, and then we will get to the statement.

Mr. Check, during the pre-hearing conference the Board directed our attention to pages 17 and 18 of the Safety Evaluation in which we speak of a five-year period before radiation effects become critical in the pressure vessel, and then indicate that there are means to mitigate the consequences of such failure if it should occur.

The Board requested that we discuss this matter of mitigating thermal shock.

Would you respond to this inquiry?

MR. CHECK: As the Board recognizes, the chief concern regarding thermal shock is pressure vessel integrity following the injection of emergency core cooling water. The recovery from the loss-of-coolant accident on a PWR is predicated upon covering the core with water.

Now with regard to thermal shock, we of course are hopeful that the work currently underway by the suppliers of pressure vessels and others will demonstrate to our satisfaction that thermal shock of the vessel to the extent of loss of integrity will not occur.

In any event, as stated in our safety evaluation, an

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immediate resolution is not required.

We are, however, considering the alternative, that is, that thermal shock does cause loss of integrity, or that the question remains unresolved.

There are several means which might be employed to deal with the problem.

(a) We might consider raising the inlet temperature of the emergency core cooling water to reduce thermally induced stresses. We are currently assessing what effect a 50 to 75-degree inlet temperature might have. There would, of course, be a corresponding decrease in emergency core cooling performance. But within a year or so we expect to have an understanding of emergency core cooling performance margins which would allow this.

We may in fact find when more realistic heat transfer analyses are performed that the temperature of the initial emergency core cooling water to reach the vessel wall is much higher than is presently assumed, simply because the water is heated during its journey through the hot piping to the vessel.

Item (a) might be considered a preventive device or means.

Item (b) and Item (c), which I will get to in a moment, might be considered mitigating means.

(b) Some means of flooding the cavity which surrounds the vessel so that a water level sufficient to cover the core can be attained whether or not the vessel itself holds water. Another PWR applicant has already presented us with a preliminary description of such a system. The key ingredient here, of course, is a floodable cavity; that is, one which holds water. We have explored this with Arkansas Power and Light Company and are assured that the Russellville cavity is indeed floodable.

Item (c): some external mechanical means to hold the vessel in place even if physical separation were to occur. The capacity of the emergency core cooling system as it is now designed is such that leaks in the pressure vessel equivalent to approximately two to three square feet could be accommodated without a drop in the water level inside the vessel. This two to three square foot area is equivalent to a complete circumferential crack of the vessel approximately one inch wide.

At present we see no reason why any of the above, if proven satisfactory in dealing with the thermal shock question, could not be installed on an operating plant. It would appear that the chief concern of the owner of such a plant would be in accomplishing the plant modification in a manner which minimizes loss of plant availability. To this end the modification could be spread over several refuelings.

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The moment we believe the integrity of the vessel is in jeopardy from thermal shock we will insist that corrective action be taken. However, as long as the question remains open -- and that is indeed the status of it -- we would not be justified in insisting on corrective action while the material properties of the vessel have not been altered sufficiently by radiation to establish the possibility of shock-induced failure, even by today's most conservative analysis.

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DR. QUARLES: Your "C" solution -- do I visualize this correctly that you put a band around it to hold it together?

MR. CHECK: One could give a number of for instances. But speaking most generally, I am speaking of something to hold the vessel, the halves actually in the position they were originally, something to simply guarantee that the vessel pieces will not move with respect to one another to any great degree.

DR. QUARLES: Can you give me a for instance other than a band?

MR. CHECK: A band -- perhaps large tie rods from top to bottom, some restraining devices. One might ever consider steel mesh.

DR. QUARLES: That would answer my question. My question is how do you know for sure which way the crack is going to propagate.

MR. CHECK: I think it is fairly clearfrom the work that is being performed that any crack that does occur, if it is to be considered a serious situation would have to propagate itself circumferentially. It would appear at the beltline of the vessel.

DR. QUARLES: Thank you. That's all I have.

CHAIRMAN WELLS: I don't believe there are any further questions on this subject.

MR. ENGELHARDT: The Board requested the staff to inform it as to what criteria will be used in determining whether it would be necessary to require the applicant to provide additional protection against tornadoes for the fuel storage pool.

Mr. Check, would you respond to that question?

MR. CHECK: The applicant has established a tornado design criteria for all plant structures and equipment important to safety. We have reviewed these and found these acceptable.

With regard to the loss of water from the fuel storage pool, however, we find that evaluating or assessing the potential for loss of water from this pool is extremely difficult. In our judgment, the present limited information based on actual tornado sightings neither allows us to completely dismiss such loss as incredible nor allows us to assign values as to probability and amount of such water loss.

Therefore, we have taken the position which has been accepted by the applicant that the pool must be designed to permit the addition of a means of protecting against water loss at a later date should this be determined by the regulatory staff to be necessary.

In other words, the pool design at this stage should not foreclose on the possibility of additional

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tornado protection.

Now, we are continuing to study this problem as it applies to all reactors located in areas of significant tornado activity.

If the question of tornado-induced water loss has not been resolved prior to the operating license stage of the review, we will require that the protection that we spoke of earlier should be added.

CHAIRMAN WELLS: If there are no further questions on this, you may proceed, Mr. Engelhardt.

MR. ENGELHARDT: If there are no further questions by the members -- the Chairman of the Board did request that I provide and updated status report of the Commission's regulations relevant to the safeguarding of special nuclear material against diversion. And I think that I am prepared to respond to that inquiry.

Public Law 88-489 was enacted in August, 1964, authorizing the private ownership of special nuclear material. Since the enactment of that statute -- as I say in March of 1967 the Atomic Energy Commission promulgated amendments to 10 CFR Part 70 which required Atomic Energy Commission licensees who possess certain amounts of special nuclear material under license including nuclear power reactor licensees to establish and maintain written procedures to control and account for special nuclear material in their possession under license and keep records showing the location and movement of such special nuclear materials and perform at least annually a physical inventory of special nuclear material possessed under licenses, to keep records of such inventory.

Subsequent to the promulgation of these amendments two offices within the Commission were established, one of which is the Division of Nuclear Material Safeguards whose responsibility it is to develop procedures for the protection of special nuclear materials.

Although the amendments that I have just identified have not been amended as yet, there have been certain changes in the organizational means of inspecting facilities to assure that these regulations are in fact being followed and that the licenses which individual operators may possess are being followed in every respect.

Under present procedures initial safeguards inspections of licensed power reactors are conducted by District 2, Safeguards Office of the Division of Nuclear Materials, located in Oak Ridge, Tennessee.

These inspectinns are instituted upon receipt of fuel at the facility and annually thereafter. The inspection objective is to assure that the licensee's material control and accounting procedures are usfficient to account for all special nuclear material in the licensee's possession.

To achieve this objective records are audited, physical inventories are verified, and the system of internal control is reviewed. To verify nuclear material used in a nuclear reactor, independent nuclear material depletion and production calculations are performed. Power generation measurements are reviewed and fuel elements in storage are identified and counted.

I think this is essentially the current status of the Commission's regulatory program regarding safegurding of special nuclear materials.

In summary, I might say that no new regulations have been effectuated or proposed since the last report provided to an Atomic Safety and Licensing Board, but the change has been essentially in administration of the program and the degree and depth to which the licensee is subject to inspection by the Division of Nuclear Materials Safeguards. I think that is the significant change that has occurred since the regulatory staff had an occasion to report on this program to an Atomic Safety and Licensing Board.

CHAIRMAN WELLS: Thank you, Mr. Engelhardt. That has more than answered my question.

MR. ENGELHARDT: I think this completes the regulatory staff's responses to questions raised by the Board at the pre-hearing conference.

CHAIRMAN WELLS: I believe Dr. Quarles has another question.

DR. QUARLES: This is directed to the applicant. It is not even a highly technical question, but I just want to clarify the record.

On page 44 of your summary statement, the last sentence of the complete paragraph on that page states, "Applicant is also a contributor to and participant in Southern Inter-State Nuclear Board and the Atomic Industrial Forum."

Since the Southern Inter-State Nuclear Board is a 17-man board, one member appointed by each of the 17 states, I wonder just what is meant by contributor to and participant in as far as SINB is concerned?

MR. HOLMES: Excuse me jst a minute, sir. May I have a short consultation.

(Pause.)

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The part of the question which addressed itself to participation in this is a fairly easy answer because we do attend their meetings and participate in the Southern Interstate Nuclear Board's activities. I have been to several of these meetings myself, as well as Mr. Philips.

The part that I was not able to answer was the congribution part, but there has been contributions in this line from a more indirect point of view in our effort to work through the Governor and to host the Southern Interstate Nuclear Board meetings in the Arkansas area. We have contributed in this more indirect means.

DR. QUARLES: Thank you. You are really participating in the activities of the Board rather than in the Board itself.

MR. HOLMES: Yes, sir.

DR. QUARLES: And you did contribute I think very significantly to a recent conference held within the past year.

MR. HOLMES: Yes, sir.

DR. QUARLES: Thank you.

CHAIRMAN WELLS: I think the Board should take note at this time again that the applicant was good enough to respond to the questions concerning the research and development by means of submission of written testimony. The board has no further questions on that. We thank you very much for doing this.

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Mr. Briggs?

MR. BRIGGS: In the ACRS letter there is a statement concerning early training and training of a sufficient number of operators. In the PSAR there is a training program schedule shown. I wondered whether this is the training program that the staff now believes to be adequate for the Russellville unit?

MR. LONG: The training program that the applicant has provided in the PSAR is not in question. It was the number of people that were involved. We stated and continued to state that we believed that a minimum manning crew for the plant, operational manning crew of five people is necessary. The applicant proposed four and this is the only difference. We are indicating it now so that the applicant is aware of our concern and will be training sufficient people to take care of this problem if we have it two years from now.

MR. BRIGGS: I see.

I would like to make an observation on the training program. There are two or three things that seem to me to be something that should be of concern to the applicant.

One is, as I see it, the fact that none of the applicant's staff would have more than three months training in an actual operating reactor before the plant goes into operation. It was pointed out that there will be simulator studies and that this will be very thorough, and that there

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is a very thorough training in theory of reactors and design of power plants and that sort of thing.

Yet as I see it, no member of the staff will have more than three months training or three months experience in the operation of a reactor, and it seems rath r doubtful that in this three months experience one would get very vital experience in fuelling of reactors, in management of wastes, in the problems that are encountered in maintenance of the reactors. And I wondered whether that had been given much consideration in the training program schedule that was developed here.

MR. HOLMES: We have proposed this training schedule in comparison with the training schedule of other utilities and we also have made efforts to get the AEC's opinion, the staff's opinion of our training program.

We felt like this was an acceptable amount of time. However, there is some adjustment in this schedule and we realize that it was set early in the game, and that as we go along we will have adjustments to make. We are at this point trying to choose our men for this nuclear unit from existing forces, that is, the people who work in our steam powered plants. These people will be the experienced people and the more talented people.

This program of choosing is underway. We hope that with the combination of these talented people that we can put in there, in this three months we do have an adequate amount of time.

MR. BRIGGS: Well, as I say, the experience that I have had indicates that three months is about time to get acquainted with the plant that you are working in and not very much time to learn about not the operation, the day-to-day operation that goes into producing power but the nasty problems that are associated with controlling the radioactivity that is discharged from the pla . and taking care of the unusual maintenance operations and that sort of thing.

MR. HOLMES: We do have one man whom I mentioned before, Mr. Harvey Miller, who is at the Seaforth facility now, training in preparation for this plant, and the Seaforth facility is in the process of handling fuel so we feel that we have one man who is now gaining experience in this fuel management. He will be in a supervisory bracket at Russellville.

MR. BRIGGS: Now with respect to the number of people that are trained, I observe here that at the time that the plant goes into, I believe, full power operation, there will only be one man per shift who is a licensed operator, and this will be the shift supervisor.

As I see it, it indicates that other shift personnel will be licensed in the plant after operation, but this would be cause for concern, would it not, if the shift supervisor would become seriously ill on a shift and leave

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the shift in the hands of unlicensed personnel?

MR. HOLMES: It was our intention to say in that that we have two licensed people per shift.

MR. BRIGGS: Well, maybe I don't understand the chart. I think it is good practice to have at least two licensed people per shift at the beginning of the operation.

Of course, these are things that would have to be resolved at the time an operating license was granted and is not anything of much concern to the granting of a construction permit, but one likes to, I think, emphasize the importance of experience in starting up a plant like this. It's a very large and not very complicated but rather sophisticated plant and certainly deserves attention.

CHAIRMAN WELLS: I believe this concludes the Board's questions.

Since I see there are in the room at this time some people who were not here this morning, I think it would be useful to let those people know that the purpose of this hearing has been not to make four or five specific findings which would have been the case had it been a contested hearing, but since this hearing was not contested, the Board is obliged only to find that information in the record is sufficient to support the application and that the examination of the regulatory staff has been adequate.

And during the course of the day the Board has asked

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a few questions. More questions were asked at the pre-hearing conference and those questions have by and large been answered this afternoon, as you have observed.

But there is a lot of material in the written record which has not been touched on today but which the Board will review in reaching a final conclusion on the two issues on which we have to make findings.

Now in addition to that the Board is further aided by the applicant and the staff because the regulations provide that they will have the opportunity to propose to the Board at some appropriate time findings and conclusions. I would like to find out at this time what is the will of the applicant and the staff as to the time in which you would like to submit to the Board your proposed conclusions and findings of fact.

MR. JEWELL: Mr. Chairman, the applicant would suggest that findings of fact and conclusions of law be submitted within ten days, and within that period of time an effort will be made by the applicant and the staff to agree upon joint findings of fact and conclusions of law for submission.

CHAIRMAN WELLS: I think I can speak for the Board and say that is agreeable to us if it is agreeable to the staff.

MR. ENGELHARDT: Mr. Chairman, I think I might be

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a little more comfortable if we equated that ten days as to a given date as to exactly when we might accomplish this.

Are you considering the 8th of November as an appropriate date for the filing of these proposed findings? That is approximately ten days.

MR. JEWELL: That's a Friday, isn't it?

CHAIRMAN WELLS: Let's don't file them on Saturday.

MR. JEWELL: May I suggest the 11th of November?

CHAIRMAN WELLS: The 11th of November is satisfactory to the Board.

MR. ENGELHARDT: Yes, sir. It's a holiday but I think we can get together before that time and work out a mutually satisfactory agreement on proposed findings.

CHAIRMAN WELLS: As far as the Board is concerned, any time within this general period would be satisfactory.

I wonder if there are any closing statements?

MR. ENGELHARDT: Sir, may I make one point with regard to the schedule? May I suggest at the same time that we use that date as established for the filing of proposed findings as the date for the filing of any transcript corrections?

CHAIRMAN WELLS: Yes. Could we have the corrections earlier, Mr. Engelhardt?

MR. ENGELHARDT: I am prepared to offer them earlier. I was just thinking of using one date as a convenient

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point of reference but I am prepared to provide the Board -- I am prepared to file transcript corrections by the 6th of November, which is Wednesday.

CHAIRMAN WELLS: The Board can expect the corrections on that date. It does, however, occur to me, if it is convenient to do so, that the 6th might be a better date for transcript corrections because then all parties will be working with a transcript that is correct.

Is that convenient to you?

MR. JEWELL: The 6th is perfectly all right with the applicant for filing corrections to the transcript.

CHAIRMAN WELLS: Very good.

Then that will be the date for corrections to the transcript.

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Does either party have any closing statements?

MR. JEWELL: The applicant does not desire to make any closing statement.

MR. ENGELHARDT: The Staff has no closing statemnt to present.

CHAIRMAN WELLS: I believe then, that all that remains is for me to discharge the pleasant duty of thaking you people who helped me make this hearing possible.

I think first of all I should say thank you, Mr. Bloom, who has been our very efficient reporter.

I would like to thank the applicant and the Regulatory Staff for their patience and their courtesy in supplying the answers we requested.

And, finally, because perhaps these thanks are most deserved, I would like to thank Dr. George Pratt, the President of the Arkansas Polytechnic College, and his assistant, Mr. Bartlett, for making it possible for us to use this room, which is a very excellent facility for such a proceeding.

If there are no other questions, and if my colleagues on the Board have nothing to add, the hearing will be adjourned.

(Whereupon, at 5:40 p.m., the hearing in the above-entitled matter was adjourned.)