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

CHAIRMAN

February 7, 1991

The Honorable William L. Clay
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Clay:

I am responding to your letter of December 19, 1990, in which you expressed concern regarding the degree of control exerted by the Union Electric Company in the construction of the Callaway Plant, Unit No. 1. The Nuclear Regulatory Commission (NRC) agrees that the adequacy of construction control at nuclear power plants is an important issue.

The basis for your concerns about Union Electric's control over construction at Callaway appears to be the Missouri Court of Appeal's discussion of construction control in a personal injury action brought by a construction worker at Callaway. The issue addressed in that private litigation did not involve whether Union Electric properly assured the quality of the work on Callaway, but whether the company had control over the safety of construction personnel. The question of occupational safety (i.e., protecting construction workers from injury caused by unsafe working conditions unrelated to radiation hazards) does not fall within the scope of the NRC's responsibility. The NRC does not review the occupational safety practices followed during construction because the NRC's statutory mandate does not extend into this area of worker safety. The Missouri appellate court, at pages 17-18 of its opinion, recognized this distinction and stated that the (NRC) construction permit required Union Electric to have control over the quality of the work, not the safety of construction personnel.

The NRC and its predecessor, the Atomic Energy Commission (AEC), have long had a working arrangement with the Occupational Safety and Health Administration (OSHA) to cooperate in the areas of radiation protection and occupational safety. This cooperation started in the early 1970s and was in effect during the construction phase of the Callaway facility. In January 1983, this arrangement was incorporated into the NRC's inspection procedures. As shown in the recent Memorandum of Understanding between the NRC and OSHA dated October 21, 1988 (Enclosure 1), NRC personnel may identify nonradiological personnel safety concerns within the area of OSHA responsibility and bring such matters to the attention of a holder of a CP and, when appropriate, to OSHA's attention (55 FR 43950).

Union Electric's actions in overseeing Callaway's construction, including its contractual relationship with the Daniel International Corporation, did not constitute a failure to control construction under the NRC's regulations and standards. The NRC's regulatory role is to ensure that commercial nuclear power plants are designed, built, and operated in a manner that will adequately protect the public from any radiation hazards thereby created. Our requirements for issuance of a construction permit (10 CFR Part 50) specify the

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information that must be submitted by an applicant. A brief discussion of this process is provided as Enclosure 2. The Atomic Safety and Licensing Board's (ASLB's) decision authorizing the issuance of a construction permit for Calloway is provided at Enclosure 3. In granting an operating license, the Commission must find that the facility has been constructed in accordance with the construction permit and that the facility can and will be operated in such a manner as not to endanger the public (10 CFR 50.57). The quality of the construction of Callaway was a particular issue in litigated proceedings before the grant of the operating license, and the ASLB determined that Union Electric properly controlled the quality of construction to permit the granting of an operating license. A copy of the ASLB's December 13, 1982 Decision (16 NRC 1826) in the Callaway Proceeding is provided as Enclosure 4.

Enclosure 5 provides more detail on Union Electric's control of construction of the Callaway facility and the steps taken by the NRC to ensure that Union Electric properly controlled the quality of construction.

Please contact me if you have any further questions on this matter.

Sincerely,

Kenneth M. Carr

Kenneth M. Carr

Enclosures:
As stated

cc: Representative Philip Sharp

MEMORANDUM OF UNDERSTANDING

BETWEEN

THE U.S. NUCLEAR REGULATORY COMMISSION

AND

THE OCCUPATIONAL SAFETY AND HEALTH ADMINISTRATION

PURPOSE AND BACKGROUND

1. The purpose of this Memorandum of Understanding between the U.S. Nuclear Regulatory Commission (NRC) and the Occupational Safety and Health Administration (OSHA) is to delineate the general areas of responsibility of each agency; to describe generally the efforts of the agencies to achieve worker protection at facilities licensed by the NRC; and to provide guidelines for coordination of interface activities between the two agencies. If NRC licensees observe OSHA's standards and regulations, this will help minimize workplace hazards.

2. Both NRC and OSHA have jurisdiction over occupational safety and health at NRC-licensed facilities. Because it is not always practical to sharply identify boundaries between the nuclear and radiological safety NRC regulates and the industrial safety OSHA regulates, a coordinated inter-agency effort can ensure against gaps in the protection of workers and at the same time, avoid duplication of effort. This memorandum replaces an existing procedure for interagency activities, "General Guidelines for Interface Activities between the NRC Regional Offices and the OSHA."

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ENCLOSURE 1

HAZARDS ASSOCIATED WITH NUCLEAR FACILITIES

3. There are four kinds of hazards that may be associated with NRC-licensed nuclear facilities:

   a. Radiation risk produced by radioactive materials;
   b. Chemical risk produced by radioactive materials;
   c. Plant conditions which affect the safety of radioactive materials and thus present an increased radiation risk to workers. For example, these might produce a fire or an explosion, and thereby cause a release of radioactive materials or an unsafe reactor condition; and,
   d. Plant conditions which result in an occupational risk, but do not affect the safety of licensed radioactive materials. For example, there might be exposure to toxic nonradioactive materials and other industrial hazards in the workplace.

Generally, NRC covers the first three hazards listed in paragraph 3 (a, b, and c), and OSHA covers the fourth hazard described in paragraph 3 (d). NRC and OSHA responsibilities and actions are described more fully in paragraphs 4 and 5 below.

NRC RESPONSIBILITIES

4. NRC is responsible for licensing and regulating nuclear facilities and materials and for conducting research in support of the licensing and regulatory process, as mandated by the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and the Nuclear Nonproliferation Act of 1978; and in accordance with the

National Environmental Policy Act of 1969, as amended, and other applicable statutes. These NRC responsibilities cover the first three nuclear facility hazards identified in paragraph 3 (a, b, and c). NRC does not have statutory authority for the fourth hazard described in paragraph 3 (d).

NRC responsibilities include protecting public health and safety; protecting the environment; protecting and safeguarding materials and plants in the interest of national security; and assuring conformity with antitrust laws for certain types of facilities, e.g., nuclear power reactors. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience; and confirmatory research.

OSHA RESPONSIBILITIES

5. OSHA is responsible for administering the requirements established under the Occupational Safety and Health Act (OSH Act) (29 U.S.C. 651 *et seq.*), which was enacted in 1970. OSHA's authority to engage in the kinds of activities described below does not apply to those workplace safety and health conditions for which other Federal agencies exercise statutory authority to prescribe and enforce standards, rules or regulations.

Under the OSH Act, every employer has a general duty to furnish each employee with a place of employment that is free from recognized hazards that can cause death or serious physical harm and to comply with all OSHA standards, rules, and regulations.

OSHA standards contain requirements designed to protect employees against workplace hazards. In general, safety standards are intended to protect against traumatic injury, while health standards are designed to address potential overexposure to toxic substances and harmful physical agents, and protect against illnesses which do not manifest themselves for many years after initial exposure.

OSHA standards cover employee exposures from all radiation sources not regulated by NRC. Examples include x-ray equipment, accelerators, accelerator-produced materials, electron microscopes and betatrons, and naturally occurring radioactive materials such as radium.

It is estimated that the Act covers nearly 6 million workplaces employing more than 80 million workers. Federal OSHA covers approximately three-fifths, or four million, of these workplaces. States which operate OSHA-approved job safety and health programs, or "Plans," cover the remainder.

OSHA State Plan States are encouraged, but not required, to delineate their authority for occupational safety and health at NRC-licensed facilities in the same manner as Federal OSHA.

The OSHA areas of responsibility described in this memorandum are subject to all applicable requirements and authorities of the OSH Act. However, the industrial safety record at NRC-licensed nuclear power plants is such that OSHA inspections at these facilities are conducted normally as a result of accidents, fatalities, referrals, or worker complaints.

INTERFACE PROCEDURES:

6. In recognition of the agencies' authorities and responsibilities enumerated above, the following procedures will be followed:

Although NRC does not conduct inspections of industrial safety, in the course of inspections of radiological and nuclear safety, NRC personnel may identify safety concerns within the area of OSHA responsibility or may receive complaints from an employee about OSHA-covered working conditions. In such instances, NRC will bring the matter to the attention of licensee management. NRC inspectors are not to perform the role of OSHA inspectors; however, they are to elevate OSHA safety issues to the attention of NRC Regional management when appropriate. If significant safety concerns are identified or if the licensee demonstrates a pattern of unresponsiveness to identified concerns, the NRC Regional Office will inform the appropriate OSHA Regional Office. In the case of complaints, NRC will withhold, from the licensee, the identity of the employee. In addition, when known to NRC, NRC will encourage licensees to report to OSHA accidents resulting in a fatality or multiple hospitalizations.

When such instances occur within OSHA State Plan States' jurisdiction, the OSHA Regional Office will refer the matter to the State for appropriate action.

7. OSHA Regional Offices will inform the appropriate NRC Regional Office of matters which are in the purview of NRC, when these come to their attention during Federal or State safety and health inspections or through complaints. The following are examples of matters that would be reported to the NRC:

   a. Lax security control or work practices that would affect nuclear or radiological health and safety.
   b. Improper posting of radiation areas.
   c. Licensee employee allegations of NRC license or regulation violations.

8. The NRC and OSHA need not normally conduct joint inspections at NRC-licensed facilities. However, under certain conditions, such as investigations or inspections following accidents or resulting from reported activities as discussed in items 6 and 7 above, it may be mutually agreed on a case-by-case basis that joint investigations are in the public interest.

9. The chemical processing of nuclear materials at some NRC-licensed fuel and materials facilities presents chemical and nuclear operational safety hazards which can best be evaluated by joint NRC-OSHA team assessments. Each agency will make its best efforts to support such assessments at about 20 facilities once every five years. Of these facilities, about one-third are in the OSHA Plan States. OSHA will also assist in promoting such participation by State personnel in OSHA Plan States.

10. Based upon reports of injury or complaints at nuclear power plant sites, OSHA will provide NRC with information on those sites where increased management attention to worker safety is needed. The NRC will bring such information indicating significant breakdown in worker safety to the attention of licensee management and monitor corrective actions. This will not interfere with OSHA authority and responsibility to investigate industrial accidents and worker complaints.

11. Power reactor sites are inspected by NRC Region-based and Resident Inspectors. Personnel from NRC Regional Offices routinely conduct inspections at most fuel and materials licensed facilities. In order to enhance the ability of NRC personnel to identify safety matters under OSHA purview during nuclear and radiological safety inspections, OSHA will provide NRC Regional personnel with basic chemical and industrial safety training and indoctrination in OSHA safety standards, consistent with ongoing OSHA training programs. To enhance the ability of OSHA and State Plan personnel to effectively participate in the Operational Safety Team Assessments, NRC will provide training in basic radiation safety requirements, consistent with ongoing NRC training programs. Details of such training will be as mutually agreed by the NRC Technical Training Center and the OSHA National Training Institute.

12. Resolution of policy issues concerning agency jurisdiction and operational relations will be coordinated by the NRC Deputy Executive Director for Operations, and by the OSHA Director of Policy. Appropriate Headquarters points of contact will be established.

13. Resolution of issues concerning inspection and enforcement activities involving both NRC and OSHA jurisdiction at NRC-licensed facilities will be handled between NRC's Office of Enforcement and OSHA's Directorate of Compliance Programs. Each NRC and OSHA Regional Office will designate points of contact for carrying out interface activities.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr.
Executive Director for Operations

FOR THE OCCUPATIONAL SAFETY AND HEALTH ADMINISTRATION

John A. Pendergrass
Assistant Secretary

October 21, 1988

Principal Architectural and Engineering Criteria

Required for a Construction Permit

The Nuclear Regulatory Commission's (NRC's) requirements for issuance of a construction permit (CP) are contained in Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) and control the level and amount of information that must be submitted by an applicant. For example, Section 50.34(a) requires that an applicant supply, among other things, the preliminary design of the facility, including the principal design criteria; the design bases; information regarding the materials of construction and general arrangement of the facility; a description of the quality assurance program to be applied during construction; and a preliminary analysis and evaluation of the performance of structures, systems, and components of the facility under normal operating conditions and in the event of postulated accidents.

This information is submitted in a Preliminary Safety Analysis Report (PSAR) and is reviewed at length by the NRC staff. Specifically, the PSAR for the Callaway Plant described standard construction elements of this facility, such as piping systems, and stated that the design of these components would comply with the requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The design of other elements, such as reinforced concrete structures, was governed by the requirements of the American Concrete Institute, while the design of the electrical systems was governed by the requirements of the Institute of Electrical and Electronic Engineers. Moreover, in its PSAR, the Union Electric Company adopted the NRC guidance presented in its Standard Review Plan (NUREG-0800) and in the NRC's series of regulatory guides. A number of the NRC staff's positions in these guidance documents are more conservative than are the industry's standards.

The items discussed above and the additional information in the Callaway PSAR constituted the principal architectural and engineering criteria for the proposed facility as stated in Paragraph 1.B of the CP issued on April 16, 1976. A quality assurance program (QAP) proposed by Union Electric was an integral part of the principal engineering criteria and was intended to ensure that the plant was designed and constructed as authorized by the Callaway CP.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

LBP-76-15

ATOMIC SAFETY AND LICENSING BOARD

Samuel W. Jensch, Chairman
George C. Anderson
Lester Kornblith, Jr.

In the Matter of

Docket Nos. STN 50-483
STN 50-486

UNION ELECTRIC COMPANY

(Callaway Plant, Units 1 and 2)

April 8, 1976

Upon application for construction permits for Callaway Plant, Units 1 and 2, the Licensing Board issues its initial decision, making determinations of law and fact, and authorizing the issuance of construction permits for both units.

ATOMIC ENERGY ACT: SCOPE OF INFORMATION REQUIRED FOR LICENSING

As a prerequisite to licensing, an applicant is required by 10 CFR 50.33(f) to provide information to show that it possesses the funds necessary to cover "estimated" construction costs and related fuel cycle costs. While a firm fuel-supply contract might provide the best estimate of costs, there is no express requirement in the Commission rules that an applicant possess such a contract in order to receive a construction permit.

TECHNICAL ISSUES DISCUSSED: site meteorology; SNUPPS application

APPEARANCES

Gerald Charnoff, Esq., Thomas A. Baxter, Esq., Joseph E. Birk, Esq., Charles E. Bremer, Esq., on behalf of Union Electric Company, Applicant

Dennis J. Tuchler, Esq., David J. Letvin, Esq., on behalf of Coalition for the Environment, and Utility Consumers Council of Missouri, Joint Intervenors

Dr. Vern R. Starks, on his own behalf, Intervenor

ENCLOSURE 3

[illegible], Esq., and W. H. England III, Esq., on behalf of the Missouri Public Service Commission

Lawrence Brenner, Esq., **Stuart A. Treby**, Esq., **Geoffrey P. Gitner**, Esq., **Gregory Lewis**, Esq., and **Auburn Mitchell**, Esq., on behalf of the Regulatory Staff of the Nuclear Regulatory Commission

INITIAL DECISION

(Construction Permit)

I. BACKGROUND

1. On August 8, 1975, the Atomic Safety and Licensing Board (Board) issued a Partial Initial Decision—Environmental and Site Suitability Determinations (NRCI-75/8, 319) in the captioned proceeding. The Board's determination, set forth in that decision, of the issues described in 10 CFR Part 51 and §50.10(e)(2)(ii), enabled the Director of Nuclear Reactor Regulation (Director) to grant the Union Electric Company's (Applicant) request for authority to conduct certain preconstruction permit activities described in 10 CFR §50.10(e)(1). On August 14, 1975 the Director granted Applicant authority to conduct such activities. This Initial Decision addresses the radiological health and safety aspects of the same application including the Board's ultimate resolution on the issuance of construction permits. The findings adopted and conclusions reached in the Partial Initial Decision are reaffirmed and incorporated herein except to the extent modified hereafter.

2. The general background of this proceeding is set forth in the Partial Initial Decision. The matters which the Board determined to be in controversy in this radiological health and safety phase of the proceeding were set forth in the Board's Special Prehearing Conference Order of February 19, 1975. Subsequently, however, on July 8, 1975 the Utility Consumers' Council of Missouri and the Coalition for the Environment St. Louis Region (Joint Intervenors) moved for amendment of their contention on financial qualifications and on August 12, 1975 asked that all other contentions concerning radiological health and safety matters be withdrawn. By an Order issued November 10, 1975, the Board granted Intervenors' Motion and acknowledged the withdrawal of all other contentions. As a result, the only matter in controversy in the radiological health and safety phase of this hearing is the Applicant's financial qualifications to design and construct the Callaway facility. The Board's findings on this matter are addressed in Section III, *infra*. On November 18, 1975 the Board issued an Order directing the resumption of evidentiary hearings in accordance

consideration of the aforementioned matters.

3. In an undated motion filed in September 1975, Joint Intervenors requested the Board to reassess the cost benefit balance reached in the August 8, 1975, Partial Initial Decision, to reflect allegedly changed circumstances with respect to the estimates of fuel costs for the Callaway Plant. In its "Order Directing Presentation of Further Evidence Respecting Costs and Benefits for the Proposed Nuclear Facility at the Sessions of Hearings to Resume on Radiological Safety Considerations," dated November 10, 1975, the Board directed that such further evidence be presented at the hearings on radiological health and safety matters. The Board did not stay its earlier Partial Initial Decision. Our consideration of the additional evidence received with respect to fuel costs is set forth in Section IV of this decision.

4. An additional matter considered by the Board was the question of whether the potential lack of a long-term fuel contract, resulting from the stated position of the Westinghouse Corporation that it is excused from fully performing its obligation to supply uranium for the facility, required withholding of a construction permit. The views of Mr. Anderson and Mr. Kornblith are set forth in Section V, *infra*. The dissenting views of Mr. Jensch are set forth following the majority opinion.

5. The evidentiary hearing was convened on December 9, 1975 in Clayton, Missouri pursuant to the Notice of Hearing published August 30, 1974 (39 *Fed. Reg.* 31690). Thereafter, sessions were held on December 10 in Clayton, Missouri, and on December 11 and 12, 1975 and January 29, 1976, in St. Louis, Missouri. The record of the hearing includes the testimony of witnesses for Applicant, the Staff, and Joint Intervenors, and exhibits. A list of the exhibits offered by the parties and received into evidence by the Board in this portion of the hearing is set forth in Appendix A to this decision. The Public Service Commission of Missouri participated in the hearing as an interested State pursuant to the provisions of 10 CFR §2.715(c).

II. FINDINGS OF FACT: RADIOLOGICAL HEALTH AND SAFETY ISSUES

A. REVIEW OF THE APPLICATION BY THE REGULATORY STAFF

6. The Applicant filed its application for licenses to construct and operate the Callaway facility on June 21, 1974. The application was submitted and accepted for review under the Commission's standardization policy statement of March 5, 1973 (SER, §1.1). This policy permits "simultaneous review of the safety related parameters of a limited number of duplicate plants which are to be constructed within a limited time span at a multiplicity of sites (*Ibid*). The Applicant is one of five utilities who have joined together under the acronym

a standard plant design for review under this policy (*Ibid*). The other utilities who have submitted such applications are Kansas Gas and Electric Company and Kansas City Power and Light Company, Northern States Power and Rochester Gas and Electric Corporation.

7. The Callaway Plant application includes a SNUPPS Preliminary Safety Analysis Report ("SNUPPS PSAR") (Exhibit #30), which describes those portions of the Callaway Plant which are standard to the SNUPPS plants, and a Callaway Plant Site Addendum to the SNUPPS PSAR (Exhibit #31), which sets forth the specific site and related design information, and applicant-related information for the plant. These documents contain comprehensive technical information relevant to radiological health and safety. In addition to a description of the site and the basis for its suitability, this information includes a description of the plant design; an analysis of the safety related structures, systems and components; analyses of postulated accidents and the engineered safety features provided to limit their potential effects; a summary of the Applicant's program for quality assurance; the Applicant's technical qualifications; the Applicant's financial qualifications; and considerations related to the common defense and security of the United States. The Board finds that this application and the PSAR, including the SNUPPS-PSAR and those portions of the Reference Safety Analysis Report (RESAR-3 Consolidated Version together with appropriate parts of Amendment 6 to RESAR-3) incorporated therein by reference, together with the testimony presented during the hearing, properly describe the facility in accordance with the Commission's regulations and the Notice of Hearing in the captioned matter.

8. The Staff performed an independent review of the information provided in the application and carried out its own analyses and investigation. On August 7, 1975, the Staff issued its Safety Evaluation Report (SER) (Following Tr. 2159) and subsequently, on November 21, 1975, issued Supplement 1 to the Safety Evaluation Report (Following Tr. 2159) which summarized the results of the Staff's evaluation of additional information submitted by the Applicants since the issuance of the SER. On January 23, 1976, the Staff issued Supplement 2. The Safety Evaluation and the Supplements thereto, delineate the results of the Staff's technical evaluation of the Callaway Plant design and the scope of technical matters considered by the Staff in its evaluation of the application. The Staff's evaluation addressed the radiological health and safety aspects of the proposed facility including site characteristics, reactor design, safety systems, quality assurance matters, conformance to general design criteria and Commission regulatory guides, financial qualifications and matters concerning the common defense and security of the United States. As a result of reviewing the information set forth in the application, the Staff concluded that the issuance of construction permits for the Callaway facility will not be inimical to

the public health and safety and the common defense and security of the United States. (SER and SER Supplement 1, §21.0).

9. Information concerning the radiological health and safety matters set forth in the application was also reviewed, independent of Staff action, by the Advisory Committee on Reactor Safeguards (ACRS) in accordance with the requirements of the Atomic Energy Act as amended, 42 USC Section 2232. The views of the ACRS were set forth in a letter dated September 17, 1975 to the Chairman of the Nuclear Regulatory Commission. The ACRS concluded that Units 1 and 2 of the Callaway Plant can be constructed with reasonable assurance that they can be operated without undue risk to the public health and safety if due consideration is given to certain matters which the ACRS believed could be resolved during construction. (*See* Appendix B to Supplement 1 of the SER). The Staff, in the second SER Supplement, has addressed the Committee's comments. In addition, the Board examined witnesses for the Staff and Applicant with respect to these comments (Tr. 2882-2914; Supplemental testimony of Schwoerer, following Tr. 3003).

10. The Board finds that the Applicant has adequately considered the radiological health and safety aspects of construction of the proposed facility and that the Staff's review and evaluation of the information set forth in the application is adequate.

B. THE PLANT SITE

11. The Board in its Partial Initial Decision of August 8, 1975, made extensive findings of fact concerning the suitability of the proposed site for the Callaway facility (NRCI-75/8 at pp. 327 - 335). At that time, the Board concluded that based on the available information and the Staff's review to date, there was reasonable assurance that the proposed Callaway site is one which is suitable for nuclear power reactors of the general size and type proposed for Callaway from the standpoint of radiological health and safety. The Staff has further reviewed the characteristics of the Callaway site in light of the particular design proposed for the plant. On the basis of its further evaluation, the Staff concluded that these characteristics are acceptable (SER, §2.1).

12. The Board has reviewed its previous findings on site suitability in the light of the particular design of the Callaway plant and the further evidence adduced in the current phase of the hearings. The additional evidence includes the Applicant's description of the site (Exhibit 31, §2) and the results of the Staff's technical review of the site characteristics (SER and SER Supp., §2). The Staff reviewed the population density and use characteristics of the environs of the site, and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology, to determine that these characteristics have been adequately described, that they have been given appropriate con-

sideration in the design of the Callaway plant, and that they conform to the Commission's reactor site criteria, 10 CFR Part 100, taking into consideration the facility design and proposed engineered safety features.

13. The Staff, in the course of its safety evaluation, received and evaluated reports from independent outside consultants concerning the potential for subsidence or collapse at the plant site which could be caused by postulated cavities hundreds of feet below the site. At the Board's request, Drs. Cording and Nieto, the Staff's consultants, testified with respect to their investigation and the support for the Staff's conclusion at SER Supp., §2.5.3. The Staff's consultants testified that a postulated cavern beneath the Callaway site would have to be in excess of one hundred feet in diameter to present a potential for subsidence (SER Supp., §2.5.3). The Staff's consultants further testified that there was a very low probability of the existence of such large caverns beneath the Callaway site (Tr. 2495, 2496). The Staff's consultants based their conclusions partially upon direct evidence from the Missouri Geological Survey (Tr. 2495). Based on its review and the independent analysis provided by its consultants, the Staff concluded that the potential for subsidence at the Callaway plant site is very remote and does not constitute a hazard to the proposed plant (SER Supplement, §2.5.3). The Board agrees and finds that the potential for subsidence at the Callaway site is remote and does not constitute a safety hazard. Further, the Board finds that the Callaway site is acceptable from the standpoint of its geological and seismological characteristics.

14. At the request of the Board, the Staff presented two witnesses, a Staff meteorologist and a consultant from the National Oceanic and Atmospheric Administration, to provide additional information on the Staff review of the site meteorology. Their testimony included identification of the meteorological data available, the methods used to evaluate the data, and the adequacy of the data. Although two years of meteorological data have been collected at the site, they testified that one year of hourly data would encompass all of the primary meteorological cycles and that the year-to-year variablity should be small because the annual amount of energy received from the sun, the driving force for atmospheric motions and stability, is relatively invariant. Whether or not that year of data is reasonably representative of expected long-term conditions at the site can be determined by a comparison between the site data and long-term data collected at nearby weather stations. When a second year of data at the site is available, these can be objectively compared to the first year by computing short-term relative concentrations for each of years. A small year-to-year difference between computed relative concentrations can be expected. This was done for the Callaway site and no meteorological or climatological conditions were identified that would indicate that diffusion calculations based on one year of on-site data are not reasonably representative of long-term conditions at the Callaway site. Analysis of the second year of data showed a 10% difference in

computed relative concentration values compared to the first year. The witnesses considered this difference to be small (they thought that the values over the long-term might vary up to 30 or 40 percent) and concluded that the second year of on-site data confirms that the first year of data collected at the Callaway site is reasonably representative of the long-term conditions to be expected (Tr. 3139-3179).

15. The Staff and the Applicant evaluated the responses of the Callaway facility to various potential accident conditions including a full spectrum of plant conditions (SER, §15.1). The spectrum of accident conditions evaluated included all design basis accidents such as the loss-of-coolant accident, steamline break accident, steam generator tube rupture, fuel handling accident, rupture of a radioactive gas storage tank in the gaseous radioactive waste treatment system, and control rod ejection accident, as required by Commission regulations. The Staff determined that the calculated potential offsite dose which would result from the occurrence of such accidents would be within the Commission's guidelines concerning site suitability set forth in 10 CFR Part 100 (SER and SER Supp., §15).

16. Based on the findings set forth in our earlier decision and the additional evidence adduced at the current hearing sessions, the Board finds that the Callaway facility can be constructed and operated at the location proposed without causing undue risk to the public health and safety.

C. DESIGN OF THE FACILITY

17. The Applicant has described in detail in its PSAR proposed design of the Callaway facility and the Staff has provided a summary description in the SER. Two 4-loop pressurized water reactor nuclear steam supply systems will be utilized at the Callaway plant. Each will have a core power level of 3411 MWT. Uranium dioxide pellets enclosed in Zircaloy tubes with welded-end plugs will be assembled into 17x17 elements to comprise the reactor core. The reactor core, which will utilize water both as a moderator and a coolant, will initially consist of three regions each containing fuel of a different enrichment of uranium-235. The reactor coolant pressure will be established and maintained by an electrically heated pressurizer which will also provide a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation. After being heated by the core, the water will flow through four steam generators where its heat will be transferred to the secondary system. The water will then flow back to the pumps and the cycle will be repeated. The heat energy transferred into the secondary system will be used in the form of steam to drive a steam turbine and generator to produce electrical energy. Control rod movement and regulation of the boric acid concentration in the reactor coolant will control the operation of each of the reactors. A reactor protection system that automatically initiates appropriate action whenever a condition monitored by the system

approaches pre-established limits will be provided. This system will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required (SER, §1.2).

18. A containment structure will house the nuclear steam supply system for each of the Callaway units. These structures will be designed so as to confine safely, within the leakage limit of the containment, the radioactive material that could be released in the event of an accident. Components of engineered safety features and various related auxiliary system will be housed in an auxiliary building adjacent to the containment structure for each unit. The fuel handling buildings, which will also be located adjacent to each containment structure, will each house a spent fuel pool and a facility for the storage of new fuel. The radioactive waste treatment systems will be housed in the radwaste building (*Ibid.*).

19. The Staff and the Applicant have evaluated the postulated effects of forces imposed by several conceivable environmental hazards. These hazards include the safe shutdown earthquake, the design basis wind, the design basis tornado, and missiles generated from within the Callaway facility but outside of containment. The Staff has determined that all structures, systems and components important to safety that must be designed to withstand the effects of a safe shutdown earthquake and remain functional have been classified properly as seismic category 1 items. These items will be designed to withstand the effects of forces imposed by a safe shutdown earthquake. Moreover, the design wind velocity and design basis tornado have also been adequately determined and all seismic category 1 structures that will be exposed to these forces will be designed to withstand the effects of such forces. The Staff has concluded that the engineering design can reasonably assure that the seismic category 1 structures will withstand such environmental forces (SER and SER Supp., §3). The Board concurs in this conclusion.

20. The Staff also reviewed the Applicant's procedures for determination of the loadings on seismic category 1 structures induced by the design flood or highest ground water level specified for the Callaway plant. The Staff concluded that these procedures are acceptable in that they provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces. The Board concurs in this conclusion. Moreover, the information provided by the Applicant and the Staff's review thereof also provides reasonable assurance that the forces associated with missiles generated from internal sources and from outside of containment will not cause or increase the severity of any accident (SER, §3).

21. On the basis of these determinations the Staff has concluded that the proposed Callaway facility can be designed, constructed, and operated to meet the requirements of the General Design Criteria set forth in 10 CFR Part 50, Appendix A of the Commission's regulations (SER, §3.1).

22. Several engineered safety features have also been included in the design of the Callaway facility. The objective of these features is to provide sufficient redundancy to overcome the effects of single failure of any component or system and to prevent the loss of capability to achieve safe shutdown of the reactor. In order to be effective, these systems will be designed as seismic category 1 systems and are required to function even with complete loss of offsite power (SER, §6.1).

23. A steel-lined, pre-stressed, post-tensioned, concrete containment structure is one of the engineered safety features which has been included in the Callaway plant design. The design of this structure is such that it will safety confine the radioactive material that could be relased in the event of an accident. In the event of an accidental coolant release, a containment spray system will operate to provide borated water containing sodium hydroxide to remove heat and radioactive iodine. A containment ventilation system, consisting of four fan coolers located within the containment structure, will be used during normal plant operation. During accident conditions the containment fan coolers can maintain the containment pressure below the containment design pressure even in the event of a single active failure in either the spray system or the fan cooling system (SER, §1.2).

24. The emergency core cooling system is another engineered safety feature. This system has been designed to provide emergency core cooling during postulated accident conditions in which it has been assumed that mechanical failures have occurred in the reactor coolant system piping with a resulting loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS in combination with the containment, containment cooling system and auxiliary feedwater system, will also be designed to protect against the consequences of a steamline break (SER, §6.3). The Staff has reviewed the information provided by the Applicant in this connection and has concluded that the ECCS for Callaway complies with the final acceptance criteria for such systems described in 10 CFR Section 50.46 (SER and SER Supp., §11).

25. The separate radioactive waste systems for the two units and the common offsite radiological monitoring system have been described by the Applicant and the estimated doses from anticipated releases of effluents have been calculated by the Applicant (Exhibits 30 and 31, §11). The Staff has also evaluated these systems and the resultant radiation exposures (SER and Supplements, §11). The Staff's initial assessment was performed to determine conformance with the design objectives of "Concluding Statement of Position of the Regulatory Staff," Docket No. RM-50-2, dated February 20, 1974 (SER, §11). At the time of issuance of the SER, the Staff concluded that the systems met those objectives, but was in the course of making a new assessment against the later requirements of Appendix I to 10 CFR Part 50, which had become effec-

tive on June 4, 1975. At the same time, the Staff was reassessing the parameters and mathematical models used in calculating releases of radioactive materials in effluents pursuant to the requirements of Appendix I (SER, §11). On September 4, 1975, the Commission amended Appendix I to allow an optional method of compliance (subsequently accepted by Applicant) which provided for compliance without a required cost-benefit analysis if the radioactive waste management system met the guidelines of the abovementioned "Concluding Statement." At the time of issuance of the SER Supplement 1 (November 21, 1975), the Staff had completed its reassessment of the parameters and mathematical models used and had requested and received from the Applicant additional information needed to reassess the systems against the new parameters and mathematical models, but had not yet completed its review of that information (Supp. 1, §11). During January 1976 the Applicant revised its waste management systems to include certain additional equipment, as described in Revision 14 to the SNUPPS PSAR, dated January 14, 1976. The Staff reported, in SER Supplement 2, dated January 23, 1976, and in the prepared testimony of two witnesses (Tr. 3184-3185) on its review of the revision and on its evaluation of the revised systems with respect to Appendix I. The Staff's evaluation of the proposed liquid and gaseous radioactive waste management systems showed them to be capable of meeting the criteria given in Appendix I of 10 CFR Part 50 for keeping releases of radioactive materials to the environment "as low as is reasonably achievable," and, accordingly, the Staff found the proposed systems to be acceptable.

26. The Staff has also evaluated Applicant's radiation protection program (Exhibits 30 and 31, §11). The review covered Applicant's radiation protection design features, including shielding and the layout of the facility, the area monitoring program, which details radiological and airborne radioactivity monitoring features, the ventilation systems which will be designed to provide a suitable radiological environment, and the health physics program. This review has shown that occupational radiation exposures can be controlled to meet the requirements of 10 CFR Parts 20 and 50 (SER, §12).

D. THE APPLICANT'S TECHNICAL QUALIFICATIONS AND QUALITY ASSURANCE PROGRAM

27. The Applicant is one of five utilities which have joined to form a SNUPPS Project Organization for the purpose of the management of design and procurement of the standard portions of the individual SNUPPS plants. Management of engineering activities related to the Callaway plant will be administered by the Applicant's Nuclear Engineering Department. The manager of nuclear engineering reports to the Vice President-Engineering and Construction who is responsible for direction of all activities which involve engineering, construction,

testing and preparation of the Callaway facility for commercial operation (SER, §13.1; Exhibits 30 and 31, §§1.4.1, 13.1). The nature of the SNUPPS organization makes available to the Applicant the operational experience and expertise of the other four SNUPPS utilities, in addition to the Applicant's own expertise in this area (Tr. 2978-2980).

28. The SNUPPS organization will also assist the Applicant in the design, construction and operation of the Callaway facility. The SNUPPS project organization has retained the Bechtel Power Corporation to provide architect-engineer services for the standard portions of the SNUPPS plants. Bechtel has also been retained to design those site-related seismic category I structures which are outside of the scope of the standard plant design for the Callaway plant. Moreover, the Westinghouse Electric Corporation has been retained to design, manufacture and deliver the nuclear steam supply system to the Callaway site. Additionally, Applicant has retained another architect-engineer organization, Sverdrup and Parcel and Associates, Inc., for technical and engineering services for those non-seismic category I portions of the facility not included in the standard portion of the Callaway plant (SER, §1.4; Exhibits 30 and 31, §1.4).

29. On the basis of its review, the Staff concluded that the Applicant is technically qualified to design and construct the Callaway facility (SER, Section 21). The Board concurs in this conclusion and finds the Applicant so qualified.

30. The Staff reviewed and evaluated the Applicant's program for quality assurance. The Staff's evaluation is based upon its review of the information presented in the Applicant's PSAR and detailed discussions with the Applicant and the SNUPPS Project Organization to determine compliance on the part of the Applicant, the SNUPPS Project Organization and the principal contractors involved with the requirements of Appendix B to 10 CFR Part 50 (SER, §17.1).

31. The SNUPPS Quality Assurance ("QA") Committee, consisting of one QA representative from each SNUPPS utility, develops the QA manual of procedures, reviews and approves Bechtel and Westinghouse QA programs and verifies their adequacy for the project, provides formal audits of the SNUPPS Project Organization, and evaluates the effectiveness of the QA program implementation. The SNUPPS Executive Director is responsible for the implementation of the QA program of the SNUPPS Project Organization through the QA Manager. The organizational level of the QA Manager provides him with adequate independence and he reports to a sufficiently high management level to accomplish his objectives. The QA manager implements the SNUPPS Project Organization QA Program through his staff. Each member of the QA Committee can initiate stop work action concerning activities managed by the SNUPPS Project Organization through the Executive Director. Additionally, the QA manager has stop work authority over the SNUPPS Project Organization staff and activities and can initiate stop work action for Bechtel activities through the Executive Director. The Staff in its review of the QA

program determined that (1) the QA organizations for the SNUPPS Project Organization are sufficiently independent of the organizations whose work they assure; (2) they have clearly defined authorities and responsibilities; (3) they have adequately defined personnel qualifications; (4) they are organized so that they can identify QA problems in the SNUPPS Project Organization as well as in the principal contract organizations; (5) they can initiate, recommend, or provide solutions; and (6) they can verify implementation of solutions (SER, §17.2). Upon this basis, the Staff concluded and the Board finds that the quality assurance program for the SNUPPS Project Organization complies with Appendix B of 10 CFR Part 50 and is acceptable.

32. The Applicant will have direct control over construction activities at the Callaway site. The Applicant's Executive Vice President is responsible to the President for quality assurance, engineering, construction and operation of the Callaway plant. The QA manager is responsible for directing the QA program. He is responsible to the Vice President-Engineering and Construction, who in turn is responsible to the Executive Vice President. The Staff has reviewed the Applicant's program for quality assurance and found that it has clearly defined responsibilities and authority for the QA organization. Moreover, the Staff determined that the Applicant's provisions for implementing its QA program, which includes corporate level management involvement, authority from the Vice President to enforce QA requirements, and QA stop work authority, are acceptable. Based upon its evaluation of the Applicant's QA organization, the Staff determined that: it is sufficiently independent of the organizations whose work it verifies; it has adequately defined qualification and training requirements for its staff; it is so organized that it can identify QA problems in other organizations performing QA related work; it can initiate, recommend or provide solutions; and it can verify implementation of solutions (SER, §17.5). Therefore, the Staff concluded and the Board finds that the Applicant's organization for quality assurance functions complies with the requirements of Appendix B to 10 CFR Part 50 and is acceptable.

33. The Staff has also evaluated the QA programs of Bechtel Power Corporation (architect-engineer for the standard plant), Westinghouse Electric Corporation (supplier of the nuclear steam supply system), and Daniel International Corporation (construction contractor), and has found those programs to be in compliance with Appendix B to 10 CFR Part 50 (SER, §§17.3, 17.4, 17.6). The Board concurs in these findings.

34. A nuclear reactor inspector from the Commission's Office of Inspection and Enforcement testified during the hearing as to the results of inspections personally performed by him and by others in which the implementation of the QA programs for the Callaway plant was examined (Tr. 2836-2870). He further testified that, subject to the satisfactory correction of several deficiencies that he identified, he had concluded that implementation of the Callaway Plant QA

programs was consistent with and satisfactory for the status of the project. At the hearing session of January 29, 1976, an affidavit was received from the witness attesting to the satisfactory correction of the deficiencies (Exhibit 46). In addition, the affidavit described the inspections of Svedrup and Parcel and Associates (site architect-engineer) and the satisfactory resolution of all deficiencies in its quality assurance program.

35. The Board finds that the Callaway Plant QA programs are in compliance with the requirements of Appendix B to 10 CFR Part 50 and further that they are adequate for the design, procurement and construction of the Callaway plant.

E. CONDUCT OF OPERATIONS

36. The proposed station organization will consist of a technical staff of approximately 145 persons for two unit operation. This technical staff will be under the direction of a plant superintendent and an assistant plant superintendent, one of whom will hold a senior operator license. A superintendent of operations, a superintendent of technical support, and a superintendent of maintenance responsible for plant maintenance, and a training director will report to the assistant plant superintendent. Shift crews composed of licensed operators and technical staff will also be provided. The Applicant's requirement for each job category used at the plant will conform to the minimum requirements of the American National Standards Institute Standard ANSI N18.1, 1971 (Selection and Training of Personnel for Nuclear Power Plants.). A training program will be established to provide plant personnel with sufficient knowledge and operating experience to start up, operate, and maintain the plant in a safe and efficient manner. The training program will include the use of a full-scale simulator control room which will be built for the SNUPPS plants (Tr. 2987). Technical support for the plant staff will be provided by a general office staff of technical specialists maintained by Applicant's Power Operations Group with additional assistance available from its Engineering and Construction Group (SER §§13.1, 13.2).

37. All plant operations are to be performed in accordance with written and approved operating and emergency procedures. These will be prepared in accordance with the guidance in American National Standards Institute Standard ANSI N18.7, 1972 (Administrative Controls for Nuclear Power Plants). Preliminary plans for review and audit of plant operations generally meet the provisions of that Standard. The Applicant has also agreed to keep plant records in conformance with the standard, as well as with Criterion XVII of 10 CFR Part 50, Appendix B (SER, §13.4).

38. The Staff has concluded that the Applicant has established an acceptable technical organization for implementation of its responsibilities for the

design and construction of the Callaway plant, that the proposed plant organization, the proposed qualifications of personnel, and the proposed plans for offsite technical support satisfy the requirements of Regulatory Guide 1.8 (Personnel Selection and Training) and are sufficient to provide acceptable staff and technical support for the operation of the plant, and that the Applicant's proposed plans for preparation, review, approval and use of written procedures and for documentation of operating and maintenance activities are acceptable (SER, §§13.1, 13.2, 13.3).

39. The initial test programs for the Callaway plant will be conducted by Applicant with technical support from the nuclear steam supply system vendor, the architect-engineer, the construction contractor and other vendors. In general, preoperational testing will be completed prior to fuel loading. As the construction of individual systems is completed, preoperational tests are performed to verify, as nearly as possible, the performance of the system under actual operating conditions. Fuel loading begins when all prerequisite system tests and operations are satisfactorily completed. While Applicant will provide additional details of its testing program at the operating license stage, the Staff has concluded that an acceptable test and startup program will be implemented by Applicant (SER, §14; Exhibits 30 and 31, §14).

40. The Board also examined the Applicant's Vice President-Engineering and Construction to obtain the views and plans of top management regarding design, construction and operation of the Callaway plant, particularly with respect to top management responsibility and participation (Tr. 2985-3002).

41. On the basis of the evidence in the record, the Board finds that the Applicant's preliminary plans for the conduct of operations are adequate for this stage of the Callaway Plant project.

F. EMERGENCY PLAN

42. The Applicant's preliminary plans for coping with emergencies include the establishment of an emergency organization which will consist of both on-site and off-site Union Electric Company personnel as well as various public and private agencies. The Applicant has identified the notification responsibilities within the organization to assure prompt and effective communication between interfacing groups. In the State of Missouri the Office of the Adjutant General has been identified as the organization having primary responsibility for radiological emergency planning. Formal training and periodic drills to familiarize plant personnel with the contents and requirements of the emergency plan, site evacuation procedures, and other emergency activities are included within the Applicant's training program. The Applicant will also provide training assistance to such outside agencies as the police department, the fire department, hospital personnel, and ambulance drivers whose services may be required in

emergency situations. Additionally, these organizations will participate in periodic emergency drill excercises (SER, §13.3).

43. The on-site emergency control center will be the plant control room which has been designed for continuous occupancy during the course of an accident. Two off-site control centers will be established in opposite directions from the plant site. The plant emergency facilities will include first aid and decontamination facilities for the treatment of contaminated personnel. The Applicant has made initial contact with four ambulance services for emergency transportation assistance and with five area hospitals for off-site emergency treatment of individuals. The various plant features to assure evacuation capability include radiation emergency alarms, site evacuation alarms, adequate communications systems and sufficient evacuation routes. The Staff has performed analyses to confirm the practicability of evacuation of the Callaway plant environs, as an emergency measure, and has determined that appropriate criteria have been identified to permit design of an acceptable evacuation plan. The Staff has concluded that the Applicant has established emergency plans which meet the requirements of Part II of Appendix E to 10 CFR Part 50 and that the Applicant's emergency program is consistent with facility design features, analysis of postulated accidents and characteristics of the proposed site location, and provides reasonable assurance that appropriate protective measures can be taken within and beyond the site boundary in the event of a serious accident (SER, §13.3). The Board concurs in this conclusion and finds that the Applicant has established adequate emergency planning programs for this stage of the Callaway project.

G. COMMON DEFENSE AND SECURITY

44. The information provided in the application shows that the activities that will be conducted under the permits and licenses applied for by Applicant will be within the jurisdiction of the United States. Moreover, all of the directors and principal officers of the Applicant are citizens of the United States and the Applicant is not owned, dominated or controlled by an alien, foreign corporation or foreign government. The activities which will be conducted do not involve any restricted data, but the Applicant has agreed to safeguard any such data that might become involved as required by 10 CFR Part 50. The Applicant will obtain its fuel from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is contemplated. Upon this basis, the Staff concluded (SER, §19) and the Board finds that the activities to be performed will not be inimical to the common defense and security of the United States.

H. INDUSTRIAL SECURITY

45. The Applicant has provided a general description of plans for protecting the Callaway Plant against potential acts of industrial sabotage. This description includes provisions for the screening of employees at the plant, and for design phase review of plant layout and protection of vital equipment in conformance with Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage." The Staff concluded and the Board finds that the Applicant's preliminary plans for protection of the plant against acts of industrial sabotage are therefore acceptable. Additional details of these plans will be provided at the operating license stage of review (SER, §13.5).

I. RESEARCH AND DEVELOPMENT

46. The nuclear steam supply systems are similar to other large pressurized water reactors now being designed and built by Westinghouse for plants being constructed under Commission construction permits. The Applicant, the ACRS, and the Staff have identified certain on-going investigations to confirm and finalize the design of certain of the plant systems, which include generic design features. These investigations include:

a. 17 x 17 fuel design;
b. Reactor pressure vessel support system;
c. Prevention of turbine missiles; and
d. Environmental and seismic qualification of Class 1-E electrical equipment.

47. The Staff has concluded (SER, §1.7) and the Board finds that the Applicant has identified and will perform development tests necessary for verification of the design and safe operation of the Callaway Plant on a timely schedule. Moreover, if the results of any of this research and development work are not successful, appropriate alternate actions, or restrictions on operation can be imposed to protect the public and safety (*Ibid.*).

III. MATTER IN CONTROVERSY—FINANCIAL QUALIFICATIONS OF THE APPLICANT

48. The determination of whether or not the Applicant is financially qualified to construct and operate the proposed facility is one of the issues set forth by the Commission for our consideration and decision. The only remaining matter in controversy deals with the same subject. The contention, which is set forth in full in Paragraph 61, asserts that, for a number of specified reasons, the Applicant is not financially capable of constructing and operating the proposed Callaway facility. In this section of our decision we will deal first with the

evidence and our findings on the financial qualifications of the Applicant, in general, to construct and operate the facility. Next we consider and rule upon the specific contentions, and finally we will consider the proposed findings of the Joint Intervenors, submitted in January, all of which are directed to this issue.

49. In preparing its Safety Evaluation Report, the Staff performed an analysis of the financial qualifications of the Applicant to determine whether there is reasonable assurance that the Applicant can obtain the necessary funds to cover estimated construction costs and related fuel cycle costs for the Callaway facility as required by the Commission's regulations (10 CFR Part 50, Appendix C and Section 50.33(f)). The Staff's approach focuses primarily on the reasonableness of the Applicant's projected system-wide financing plan and its underlying assumptions (Testimony of Richard Cioni following Tr. 2715 (hereafter Cioni Testimony) at 1).

50. In performing its evaluation, the Staff considered extensive financial information which was provided by the Applicant concerning Union Electric's revenues, the magnitude of its construction program and potential sources of funds (SER Supp. 1, §20). By agreement among the parties, only one version of the sources of fund statement provided by the Applicant was published in the Staff's Safety Evaluation Report Supplement (SER Supp. 1, Appendix E). Due to potential complications caused by regulations of the Securities and Exchange Commission, another version of the sources of fund statement, considered by the Applicant to be proprietary was withheld from public disclosure. The Staff in its evaluation, however, considered both the proprietary and non-proprietary versions of the sources of fund statement (SER Supp. 1, §20). Joint Intervenor's witness stated that he considered the use of a source of fund statement, such as that provided by the Applicant, prudent in the development of a financial plan for the construction of a facility such as Callaway (Tr. 2446).

51. As a result of its evaluation, the Staff concluded that the Applicant is financially qualified to design and construct the Callaway Plant (SER Supp. 1, §20). Joint Intervenors, however, assert that funds to meet these costs are not obtainable. Two points are essential to the Board's resolution of this issue: (1) construction and related fuel costs which will be necessitated by the Callaway project and (2) the reasonableness of Applicant's assertion that these funds are obtainable.

52. The Applicant estimated the cost of the Callaway facility including the nuclear fuel inventory cost for the first core and transmission, distribution and general plant cost to be 1,862.6 million dollars (SER Supp. 1, §20.2). The Staff compared this estimate against a costing model developed by the Energy Research and Development Administration. The Applicant's cost estimate is approximately 5.8 percent above the estimate derived from the costing model (*Ibid.*). Upon this basis, the Staff concluded and the Board finds it is reasonable to use the Applicant's estimate.

53. The Applicant plans to finance the Callaway Plant by the use of internally generated funds,[1] notes payable, and the issuance of debt and equity securities. Available funds from these sources in 1974 totaled 228.8 million dollars and were derived from 42.5 million dollars of internally generated funds, 77 million dollars of first mortgage bonds, 16.5 million dollars of environmental improvement revenue bonds, 77.4 million dollars in preferred and common stock, and a 15.4 million dollar increase in notes payable and other funds. As a result of its evaluation, the Staff determined that the financing projections of the Applicant could be characterized as a reasonable financing plan. Moreover, assuming rational regulatory policies and relatively stable capital market conditions, the Applicant has reasonable assurance of obtaining the funds necessary for construction of the Callaway Plant (*Ibid.*). Joint Intervenors, however, challenge both of these assumptions.

54. Joint Intervenors contend that it is no longer as probable as it was that the Missouri Public Service Commission will grant the Applicant needed rate increases. However, the Missouri PSC is obligated to grant just and reasonable rates so that safe and adequate service might be rendered and the setting of such rates is tantamount to allowing a company a reasonable opportunity to earn a fair return on its investiment (Cioni Testimony at 2). The Staff and Applicant aver that the considerable rate case activity, both currently and in the recent past, demonstrate the more favorable regulatory climate for utilities such as the Applicant (Cioni Testimony at 2 and 3). The Applicant's retail rates are regulated by the Public Service Commission of Missouri, the Illinois Commerce Commission, and the Iowa State Commerce Commission (Testimony of William E. Cornelius following Tr. 2597 (hereafter Cornelius Testimony) at 22).

55. The Illinois Commerce Commission authorized in 1974 a 13.5 percent increase in Applicant's rates (6.4 million dollars). The Applicant now has pending with that Commission a requested rate increase of 24.4 percent (15 million dollars) based upon a test year ending December 31, 1975 (*Ibid.*).

56. The Iowa State Commerce Commission in 1974 authorized a rate increase of approximately 21.6 percent (2.1 million dollars) for the Applicant. On February 20, 1975 the same Commission allowed proposed rate increases of approximately 10.4 percent (1.2 million dollars) to become effective subject to refund based upon the outcome of the proceeding (*Ibid.*).

57. In 1974 the Missouri Public Service Commission authorized a 13.8 percent (39.9 million dollars) increase in Applicant's rates and in 1975 the same Commission issued a report and order approving rate increases for the Applicant of 50.9 million dollars effective January 2, 1976 (*Ibid.*).

58. The Board is of the view that the weight to be attached to such rate case activity is enhanced by efforts of these Public Service Commissions to achieve regulatory reforms designed to overcome problems of regulatory lag and high inflation. These regulatory reforms contribute to a more favorable investment climate for public utilities such as the Applicant. Moreover, they represent a

more realistic approach to rate regulation and constitute a major step toward minimizing the effects of regulatory lag (SER Supp. 1, §20.3). For instance, the use of a forward-looking test year, which has recently been adopted, acts to establish rates on the basis of the capital employed and the expenses anticipated during the period the rates are to be in effect (Cornelius Testimony at 24; Cioni Testimony at 3).

59. Another effort at regulatory reform which is significant is the inclusion of construction work in progress (CWIP) in the rate base. Joint Intervenors contend that increases in the allowance for funds used during construction (AFDC) will adversely affect the Applicant's ability to finance the Callaway project. While it is true that the investment community generally views AFDC as a lower quality of earnings than that generated by operating plants, several regulatory commissions have recently allowed construction work in progress in the rate base during rate proceedings. Indeed, in the recent rate decision of the Missouri Public Service Commission (Cases No. 18,314 and 18,527), the Applicant received its requested rate increases as well as a favorable decision upon the inclusion of construction work in progress within the rate base.[1] A witness for the Applicant who is considered to be an eminent authority in the field of public utility finance and whose writings were cited by Joint Intervenors' witness, testified that he believed the investment community would view such a regulatory reform favorably (Tr. 2928).

60. The Board therefore finds no basis upon which to assert that the Applicant's prospects for obtaining the necessary rate relief are unfavorable and finds that Applicant's and Staff's assumptions that such relief will generally be granted by the public service commissions in which the Applicant does business are reasonable. With respect to the continuing availability to the Applicant of a continuing stable market, the Joint Intervenors principal points are dealt with below in connection with the specific contentions.

61. The specific contention stipulated to by the parties is as follows:

III. Applicant is not financially capable of constructing and operating the proposed Callaway facility as projected, in that

A. Money to be used to pay for the construction of the proposed Callaway

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[1] The Public Service Commission of Missouri, in a Report and Order dated December 22, 1975 (Exhibit 44), approved Applicant's request to include in the rate base CWIP representing expenditures through December 31, 1975, on the Callaway Plant. As a result, Applicant will discontinue capitalizing AFDC on the CWIP included in the rate base. The inclusion of CWIP in the rate base will allow Applicant to generate more funds internally and thereby reduce its reliance upon external financing. If continued over the period of Callaway Plant construction, the inclusion of CWIP in the rate base could increase Applicant's internal cash generation by up to approximately $275 million. Tr. 3129, 3130.

facility will be raised only by a substantial amount of borrowing, and large sales of common stock.

1. An increase of shares of common stock which is projected will require that a large portion of common shares sold will be sold for less than book value.
2. The sale of debentures will make increasingly difficult the maintenance of required ratios between debentures, common shares and preferred shares.
3. Long term borrowing may be curtailed by the inadequacy of the ratio between after-tax earnings and total fixed charges provided for in present indentures.
4. Short term borrowing for current bills during construction will be limited by the ability of Applicant to sell stock, and to acquire long-term loans. Applicant has already suffered limitations on its short term borrowing.
5. Applicant's borrowing will also be limited by the reduction in its rating from AA to A by Moody's and Standard & Poor. Among the reasons given for the reduction of ratings was that the projected Callaway facility put too great a strain on Applicant's financial resources.
6. The unprecedented demand for capital projected over the years by industry, especially energy and utilities, United States Treasury, United States government agencies, and state and local government both increase the price and reduce the availability of capital, making it unlikely that relatively poor risks, such as Applicant, will be able to make either long or short-term borrowings in sufficient amounts to finance the projected facility.
7. The increase in allowance for construction as a proportion of total earnings will limit the sale of stock and of debentures. Allowance for construction, which is a fictitious account regarding assumed return to the plant under construction—were it operating—will inflate earnings reported to shareholders to a far greater extent than they will provide a basis for borrowing or computing required ratios. In 1974, it was reported that $1.37 was earned per share. Less allowance for construction, the amount is more properly $.94 per share.

B. Amounts needed for completion are not provided by the Applicant since no adequate account is taken of probable overruns.

C. Applicant's financial weakness is illustrated by the low quality of its earnings after taxes, as compared to other electric utilities, and Applicant paid very low taxes when compared to other electric utilities.

D. Depreciation accounted for by Applicant is not adequate to replace Callaway Plants 1 and 2.

E. It is no longer as probable as it was that Missouri Public Service Commission will give the rate increases needed by the Applicant to remain in a reasonably healthy condition, whatever its financial ventures.

62. Contention III A identifies in its several subsections the reasons why Joint Intervenors believe the substantial amount of borrowing and the large sales of common stock necessary to raise the money needed for construction of the facility cannot be accomplished. The fact that such borrowing and sales will be necessary is not disputed. The electric utility industry is one of the most capital intensive in the United States and requires $4 of investment to produce $1 of revenues. Consequently, utilities must continually tap the money and capital markets for the funds needed to build the power plants to safety and adequately service their customers (Cioni Testimony at 4). This in itself does not lead to a conclusion that the Applicant will be unable to finance the project. We must look at each of the individual parts of the contention.

63. Intervenors contend that a large portion of the common shares Applicant will offer in financing the Callaway project will be sold for less than book value. Intervenors did not demonstrate by any of their evidence the manner in which this problem would impair the Applicant's capability to finance the Callaway project. While Applicant's common stock is presently slightly below book value, its ratio of market price to book value has recently increased and is anticipated by both the Applicant and the Staff to average at or above book value during the period of construction of the Callaway Plant (Cioni Testimony at 4). The Board finds on this point that there is no basis to conclude that the Applicant's ability to sell common stock will be impaired.

64. Another basis upon which the Intervenor challenged the Applicant's financial capability was the difficulty Intervenor asserted the Applicant would encounter in an effort to maintain responsible capitalization ratios. Intervenors do not assert, however, the manner in which this alleged difficulty will impair the Applicant's capability to finance the Callaway project. Moreover, there is no requirement that this Board is aware of that any specific ratio between debentures, common shares and preferred shares must be maintained by the Applicant as Intervenors apparently contend. The Board does not find any substantial evidence either that the proposed ratios cannot be maintained or that small deviations from the proposed ratios would significantly affect the Applicant's ability to finance the project.

65. Intervenors further allege that the Applicant's long term borrowing capability may be curtailed by its inability to meet indenture coverage requirements. The preponderance of the evidence, however, indicates that the required coverage can be maintained, assuming reasonable rate relief. The Joint Intervenors' apprehensions on this point might stem in part from the belief, as indicated in the contention, that this coverage requirement is based on after-tax earnings, whereas in fact it is based on income before taxes (Cornelius Testimony at 11).

66. Contention IIIA(4) asserts that short term borrowing will be limited by Applicant's inability to sell stock and bonds and that the Applicant has already suffered such limitations. The capability for selling long term securities is dealt with elsewhere. In regard to the second assertion, the Board can find no support in the record. On the contrary, Applicant's testimony indicates it has never been unable to carry out its normal short-term borrowing program (Cornelius Testimony at 13) and Joint Intervenors' testimony deals with projected future difficulties (Testimony of David Gottlieb following Tr. 2163 (hereafter Gottlieb Testimony) at 54-59).

67. With respect to Contention IIIA(5), that borrowing (presumably long-term) will be limited by a recent reduction in bond rating (from AA to A in January, 1974, and to A- in November, 1974, by Standard and Poor and from AA to A in February 1975 by Moody), witnesses for Applicant and Joint Intervenors agreed that the main impact of a lower rating is to increase the interest cost associated with the bonds (Cornelius Testimony at 12; Gottlieb Testimony at 25). Subsequent to the reduction in ratings, Applicant in March 1975 sold an issue of mortgage bonds at an interest rate of 10-1/2 percent (Tr. 2505-2506). Joint Intervenor's witness conceded that there was a good market for bonds rated at single A (Tr. 2326). The evidence, therefore, does not appear to the Board to substantiate the assertion that borrowing will be limited by the reduction in bond rating.

68. Another point upon which Intervenors challenge Applicant's financial capability is the "crowding out" theory. In Contention III (A)(6), Joint Intervenors allege that Applicant is a relatively poor risk and will be crowded out of the capital markets by the unprecedented demand for capital projects over the years by the electric utility industry. In support of its contention, Joint Intervenors cite the low rating of Applicant's bonds. As we have seen, however, Applicant has sold bonds (and also issued stock, as recently as December 1975 - Exhibit 35) despite the rating. It is more likely that capital will become more expensive, not unavailable, if capital becomes scarce in the future (Cioni Testimony at 6). One of Applicant's witnesses, an acknowledged expert in this field, testified that the weaker demanders of capital most likely to be crowded out of the financial markets will be small businesses, local governments, and individuals. He deliberately omitted from the list of weaker demanders of capital the regulated private utilities, which, in his view, are in a preferred position in the capital markets. He testified that under almost any foreseeable scenario, utilities with credit ratings of singe-A or single-A minus will have good access to the capital markets, the only question being the requisite interest rate (Tr. 2920A-2921). The Board finds no evidence in the record that demand for capital will make the Applicant unable to raise the necessary funds.

69. The final part of Contention IIIA asserts that the increase in allowance for funds used during construction (AFDC) will limit the sale of stock and debentures. The evidence does not support the assertion. During the past 5 years, 84% of Applicant's AFDC has been interest and preferred dividend costs of borrowed money. The propriety of capitalizing this has never been seriously questioned. The formula prescribed by the SEC for calculation of earnings for calculating earnings to fixed charge ratio provides for inclusion of 100 percent of AFDC. The definition of earnings under the Applicant's mortgage indenture is somewhat different, but during the last 5 years, 71 percent of the AFDC has qualified (Cornelius Testimony at 17-19). In addition, as we have stated earlier, the most recent rate decision in Missouri allowed applicants to capitalize CWIP, which will avoid any AFDC for that portion of plant. The Board has been unable to find any substantial evidence that AFDC will have a significant adverse affect on the Applicant's ability to finance the plant.

70. Intervenors contend in Contention IIIB that probable overruns have not been considered in the Applicant's estimate of cost of construction of the Callaway facility. However, Applicant's most recent cost estimate includes contingency allowances of 63.3 million dollars for Unit 1 and 56.2 million dollars for Unit 2 of the Callaway Plant (Cioni Testimony at 7). Thus, the Board finds that the Applicant has taken the possibility of cost overruns into account in estimating the cost of the Callaway Plant.

71. In Contention IIIC, Joint intervenors allege that the low quality of its earnings and its low tax rate are illustrations of the Applicant's financial weakness. Intervenors, however, have not further asserted the manner in which this alleged financial weakness impairs the capability of the Applicant to obtain the funds necessary for construction of the Callaway project. The Board notes that the Applicant is a "flow-through" company and therefore will flow-through the tax savings associated with accelerated depreciation. This is an accounting technique which is a principal determinant of the Applicant's low tax rate and is further mandated by regulatory commission policy. Additionally, the total taxes paid by an electric utility will vary from year to year as a result of its current business transactions, local tax rates, the investment tax credit applied to electric plants placed in service during the period and many other factors (Cioni Testimony at 7). The Board finds that the relatively low tax rate of the Applicant in and of itself, is an insufficient basis for determining the company's financial strength or weakness.

72. Joint Intervenors allege in Contention IIID that depreciation is inadequate to replace Callaway Plant Units 1 and 2. However, depreciation will be determined by policies established at the time the plant is placed in operation and the relevance of the contention at this time is questionable. Beyond this, however, the contention appears to reflect a misunderstanding of the function of depreciation. Depreciation is not intended as a source of funds to fully replace the property or plant being depreciated. Rather it is a system of accounting which aims to distribute the cost of a plant, less any salvage value, over the estimated life of the plant in a systematic and rational manner. Depreciation

permits expense allocated to a given accounting period to be offset representatively, in part, against the revenues produced through the use of the property or plant during that period (Cornelius Testimony at 20-21). In any event, it is not apparent that the adequacy of the depreciation allowance would affect the Applicant's financing capability.

73. The final contention, Contention IIIE has been addressed earlier.

74. Proposed Finding 3 of the Joint Intervenors asserts that it is not reasonably probable that the Missouri Public Service Commission will continue to approve the rate increases necessary to finance the plant. We reject this finding on the basis of our discussion in Paragraphs 54 to 59 *supra*.

75. Proposed Finding 4 is addressed to the sufficiency of the cost estimate for the plant and is rejected on the basis set forth in Paragraphs 52 and 70 *supra*.

76. Proposed Finding 5 relates to the downgrading of Applicant's credit rating and its effect on interest rates. This finding has, in essence, been incorporated in Paragraph 67 of our findings.

77. Proposed Finding 6 is addressed to the same subject as Proposed Finding 3 and is rejected for the same reason.

78. Proposed Finding 7 is rejected because it, again, is addressed to cost estimates and has been covered in Paragraphs 52 and 70 *supra*.

79. In consideration of all the evidence adduced by all parties in the proceeding relative to the capability of the Applicant to obtain the necessary funds to design and construct the Callaway project, the Board finds that there is reasonable assurance that the Applicant can obtain such funds as are necessary for the construction of the Callaway Plant including related fuel cycle cost and that therefore the Applicant is financially qualified to design and construct the proposed facility.

IV. COST OF FUEL, AVAILABILITY OF FUEL, AND COST-BENEFIT BALANCE

80. The Board has reconsidered its Findings in Paragraphs 69–80 of its August 8, 1975, Partial Initial Decision, relating to nuclear fuel costs and availability in the light of the actions taken by Westinghouse respecting abrogation of parts of its fuel supply contract with the Applicant. The principal questions considered by the Board were whether or not the recent changes in the fuel cost situation, have altered the cost-benefit balance and whether or not the record supports the Board's previous finding that there will be an adequate supply of uranium.

81. The Board's findings in its earlier decision was based in part on the existence of a contract between the Applicant and Westinghouse providing *inter alia* for a 21-year supply of uranium for the plant. The estimates of fuel cycle costs given by the Applicant in its earlier testimony were based on the firm

prices (plus escalation) provided in the contract for the first twelve years of operation and on estimates (based on industry models) developed by its fuel consultant for the balance of plant life.[3] On September 8, 1975, Westinghouse announced that it considered itself "legally excused" from a portion of its obligation to deliver uranium. Westinghouse announced that it intended to perform its contracts to the extent of its uranium presently in inventory or on order by distributing it fairly and equitably among its customers. Westinghouse also announced its intent to invest substantial funds in exploration and production of uranium which would be made available to its customers on favorable terms. Applicant has subsequently filed a civil action against Westinghouse to compel performance of the fuel contract. In an Order entered in that action on February 3, 1976, by the U. S. District Court for the Eastern District of Virginia, Westinghouse was directed to deliver to Applicant (among others) an allocated percentage of Westinghouse's existing uranium supply at the times and prices specified in the contract.[3] While the precise quantity of uranium to be delivered was to be determined after a report was made to the Court on February 16, 1976, the allocation formula set forth in the Order would provide Applicant with 2.0 million pounds of uranium, to be delivered at the times and prices specified in the contract, until the allocated amount is reached. This amount of uranium will be sufficient to fuel the complete first core of Unit 1 of the Callaway Plant, and at least half of the first core of Unit 2 of the Callaway Plant. All or a part of this uranium, however, is not yet in the hands of Westinghouse and its delivery by Westinghouse to its customers is dependent on its receipt by Westinghouse from its suppliers.

82. Both Applicant and Staff presented additional evidence on projected fuel costs based on their evaluations of the current U_3O_8 market conditions and their current predictions of the future prices for uranium. The Applicant reported that U_3O_8 spot market prices for immediate delivery have increased in the past two years from about $6 per pound to about $26 per pound. Applicant has recently issued a letter of intent and is negotiating a contract for 1,000,000 pounds for 1980 delivery at $40 per pound plus interest for three years, or approximately $50 per pound for 1980 delivery payable in 1980 dollars (Supplemental Direct Testimony of Seymour Jaye, following Tr. 2577 (hereafter Jaye Supplemental Testimony), at p. 10; Tr. 3017-3019). Applicant's fuel consultant, based on its industry models, currently projects U_3O_8 prices of $34 per pound in 1980, $42 per pound in 1990 and $76 per pound in 2000. Use of these

[3] The Staff's estimate was based on cost projection by ERDA.

[3] Affidavit of Seymour Jaye, February 4, 1976, with attachment. Applicant's unopposed motion of February 9, 1976, that this affidavit be marked as Exhibit 48 and received in evidence is granted.

projected prices leads the consultant to predict a 1982 nuclear fuel cycle cost of 4.4 mills per KWH and a 20-year levelized cost of 4.6 mills per KWH (Jaye Supplemental Testimony at pp. 2-3). The earlier predictions of these costs were 2.5 and 3.4 mills per KWH, respectively. The allocation of Westinghouse fuel discussed above, if realized, will somewhat reduce these costs. Use of the fuel now under letter of intent, however, will somewhat increase the costs. If both events come to pass, the 1982 cost will be increased by 0.5 mills per KWH and the 20-year levelized cost will be increased by 0.2 mills per KWH (Jaye Supplemental Testimony at pp. 10-11). This change appears insignificant in relation to overall costs.

83. The Staff projected a 1982 U_3O_8 price, based on supply cost and rate of return considerations, of $31.88 per pound. Because of market influences and the possible course of reserve development, the Staff increased this to $40 per pound for use in the cost-benefit calculations (Supplemental Testimony of Darrel A. Nash, following Tr. 3088-B (hereafter Nash Supplemental Testimony) at pp. 9-10). Using this as a basis, the Staff updated its earlier estimate of 5.6 mills per KWH and arrived at an estimate of 7.1 mills per KWH (Nash Supplemental Testimony at pp. 10-11). The Staff further estimates that use of the fuel under letter of intent (but not the Westinghouse allocation) would raise this by 1.0 mills per KWH (Nash Supplemental Testimony at pp. 12-13). The Staff has not calculated (either now or earlier), a levelized fuel cost based on assumed escalation. Rather, it has made its comparisons of power generation costs between nuclear and coal plants on basis of 1982 costs without any escalation. This, of course, favors coal in the comparison, because equal escalation for both fuels would increase coal-fired generation costs faster than nuclear generation costs as a result of the larger fraction of generation costs accounted for by fuel in the case of the coal-fired plants (Pollnow Testimony following Tr. 1459, at p. 26).

84. Both Applicant and Staff have revised the comparisons of 1982 coal versus nuclear generation costs. The Applicant now calculates a cost advantage for the nuclear plant of 8.9 mills per KWH at an 80% capability factor, 8.7 mills per KWH at a 70% capability factor and 8.3 mills per KWH at a 60% capability factor. Using a 59% capability factor for the nuclear plant and 67% for the coal-fired plant, the Applicant finds a 5.4 mill per KWH advantage for the nuclear units (Jaye Supplemental Testimony, Table 2). The Applicant's comparable earlier figures were 10.5, 10.4, 10.0, and 6.9 (Testimony of H. Clyde Allen, following Tr. 1079, Exhibit 5). The Staff's similar calculation shows a nuclear cost advantage 4.4 mills per KWH at a 75% capacity factor, 3.9 mills per KWH at 65% capacity factor and 3.5 mills per KWH at 55% capacity factor (Nash Supplemental Testimony, Table 5). The earlier comparable figures were 5.9, 5.4, and 5.0 (Pollnow Testimony, Table 10). Since one mill per kilowatt-hour at 70% capability factor represents about seven million dollars per year, the advantage of the nuclear plant, although smaller, remains substantial and the previous views of the Board as to the cost-benefit balance remain unchanged.

85. The Board has reviewed the testimony presented earlier with respect to the amounts of U_3O_8 reserves and resources (Testimony of Allen, Exhibit 10; Testimony of Pollnow, pp. 19-20) and considered the additional testimony offered in December and January (Jaye Supplemental Testimony, pp. 3-8; Nash Supplemental Testimony, 3-8; Testimony of Richard H. DeVoto, attached as Appendix A to Jaye Supplemental Testimony; Tr. 3023-3042, 3049-3080, 3093-3095; Exhibit 42). The testimony presented in the current sessions was extensive. The witnesses for both Staff and Applicant discussed the definitions of the various types of resources, the methods by which the sizes of the resources were estimated, and the nature of the inaccuracies involved. Although some of the more recent evidence presented tends to demonstrate that ERDA's resource (as opposed to reserve) estimates may not be precise, adequate reserves and resources appear to be available. Even discounting the "hypothetical" and "possible" resources, it appears to the Board that the amounts of uranium in the "reserves" and "probable resources" categories, even allowing for possible (and likely) inaccuracies, are sufficient to assure with a reasonable probability that adequate fuel will be available for this facility, considering the needs for all of the 236 reactors presently operating, under construction, and planned. Further, this conclusion appears to the Board to be valid regardless of the assumptions that are made regarding uranium or plutonium recycle or enrichment tails assay. While there might be some question regarding the adequacy of uranium resources for the 625 to 1200 reactors that ERDA anticipates might be in existence by the year 2000, this Board need not and has not addressed that issue.

86. Intervenors while not raising any specific contention respecting fuel costs nor contract terms has filed proposed findings based upon the direct and cross-examination of the data presented by the Applicant and the Staff. The foregoing analysis of costs and estimates made by the Board establishes that Intervenors proposed findings are not supported by reliable probative and substantial evidence and are therefore rejected. Intervenors rely upon portions of Applicant's and the Staff's evidence which are out of context of the presentation made by the witnesses. Applicant's witnesses indicated a reliance on domestic, not foreign supplies, and the Staff estimates of costs were based upon the projected trend of market conditions, not estimates of prices that might or might not be included in an additional fuel contract for the Applicant. In addition, Intervenors utilize capacity factors for cost comparisons that are contrary to previous findings made in the LWA portion of the proceedings. Intervenors presented no evidence to dispute these findings. For these reasons, Intervenors findings numbered 8 and 9 are rejected. Intervenor's proposed finding number 10 is the subject of both majority and dissenting opinions and is rejected by the majority.

V. FUEL CONTRACT CONSIDERATIONS

87. At the outset of this proceeding, Applicant verified that it possessed a

valid fuel supply contract covering the supply of uranium for the first 21 years of the plant life. As described earlier, Applicant reported during the course of the hearings that its supplier, Westinghouse Corporation, stated its position that it is excused from fully performing its obligation to supply uranium under this contract. The Applicant disputes this Westinghouse position and court action to resolve the dispute is in progress. The Applicant, meanwhile, has considered other sources of supply, and the record shows that it has issued a letter of intent to execute a contract with Western Nuclear Corporation for 1,000,000 pounds of U_3O_8. At the time of the last evidentiary hearing negotiations were still continuing and a formal contract had not been signed.

88. The Board has considered the question of the necessity of evidence of a firm fuel supply and delivery contract as a part of this proceeding. On January 30, 1976 the Board requested by letter that the parties address this question in their proposed findings and conclusions, and, in a letter of February 11, requested the Staff to provide information as to whether "the generally consistent practice of the Commission is to secure proof of a fuel contract at the time of issuance of a construction permit" and, if so, that the Board be advised of any exceptions to such a practice. Both Applicant and Staff responded to the Board's January 30 request. The Applicant states that it is not aware of any requirement in law that it need prove the existence of a contract nor is it aware that a requirement for such a demonstration would be consistent with past Commission practice ("Applicant's Statement on the Need for Evidence of a Firm Fuel Supply and Delivery Contract" dated February 9, 1976). The Staff took a similar position with respect to the requirements of the Atomic Energy Act and also considered the applicability of NEPA to the question. Acknowledging the relevance of fuel availability and price to both the cost-benefit balance and the comparisons of alternatives, the Staff concluded that the current record was sufficient to demonstrate that the Callaway Plant has a favorable cost-benefit balance and is preferred over the coal alternative ("NRC Staff's Statement on Necessity of Nuclear Fuel Supply Contract" dated February 20, 1976).

89. The Board addressed an additional letter to the Staff on February 27 requesting again that the Staff review the records of applications and other related submittals to provide information on the extent of the practice of applicants providing information respecting the existence of fuel supply contracts in connection with their applications. The Staff responded to this on March 19, 1976, by Affidavit of Harold R. Denton, Director of the Division of Site Safety and Environmental Analysis.* The Affidavit stated, in essence, that the NRC has

*Without objection from the other parties, the Affidavit of Denton is received in evidence and incorporated herein as Exhibit 49.

no requirement that an applicant submit proof of execution of a fuel supply contract with its application, that no mention was found of fuel supply contracts in the Environmental Reports or in the Safety Analysis Reports examined, and that the only mentions found were in the prospectuses routinely requested from applicants in the course of Staff review of financial information. According to the Affidavit, the Union Electric prospectus of March 19, 1975, described both long-term coal contracts and the 21-year Westinghouse nuclear fuel supply contract. The Staff's records for three other proposed facilities were examined. Two of these were in the very early stage of the licensing process. In one case the prospectus stated that no fuel contract had been executed. The second case had not yet reached the stage of Staff review at which prospectus information is required. In the third case, the prospectus stated that a fuel supply contract was "under negotiation." In this latter case a construction permit was granted on February 24, 1976.

90. The Board has reviewed the regulations and the information submitted by the parties. It finds that there is no express requirement that an applicant possess a nuclear fuel contract in order to receive a construction permit. Although it is likely that in many cases an applicant has in fact had such a contract at the time of issuance of the construction permit, it is reasonable for us to conclude that this was done for business reasons rather than to satisfy a regulatory requirement. This view is buttressed by the appearance of the information on contracts not in documents prepared for the Commission, but in prospectuses prepared for other purposes and submitted to the Commission only as ancillary information bearing on financial qualification. Taken in its proper context, this information is provided in the prospectuses for the purpose of advising prospective investors of information relevant to the utility's fuel supply situation and its long-term obligations. In our view, such assertions regarding the existence of fuel supply contracts, even if they are all positive which they are not, do not, simply by their prevalence, become *de facto* regulatory requirements.

91. The requirements relating to fuel that do appear in the rules appear in the portion of the rules covering financial qualifications and require that the applicant provide information to "show that the applicant possesses the funds necessary to cover *estimated* construction costs and related fuel cycle costs" (Section 50.33(f), emphasis added). For this purpose, although a firm contract might provide the best estimate of costs, the Board has found above that the estimates presented on the record, for both construction and fuel cycle costs, are adequate to allow us to make the findings regarding financial qualification that we have made in this decision. Similarly, we have found the estimates adequate

for the findings required by NEPA regarding cost-benefit balance and alternatives.[8]

92. At the suggestion of the dissenting member of the Board, we have reviewed the legislative history of the private ownership act and have received a somewhat different impression from that of our colleague. In our view, the basic purpose of the Congress was to put the provision of nuclear fuel on a commercial basis comparable to that of fossil fuel. This included the ability of the utilities to enter into long-term contracts if they so desired. We find no indication in the history of the intent to make this mandatory, but rather to permit utilities to undertake their normal planning procedures and to deal with suppliers in the normal commercial manner.

VI. CONCLUSIONS OF LAW

The Board has reviewed the entire record in this proceeding, including all of the proposed findings of fact submitted by the parties. Those proposed findings submitted by the parties which are not incorporated directly or inferentially or specifically discussed elsewhere in this Initial Decision are herewith rejected as being unsupportable in fact or in law, or as being unnecessary to the rendering of this decision.

Based on its review, the Board concludes that the Application and the proceedings thereon comply with the requirements of the Atomic Energy Act of 1954, as amended, the National Environmental Policy Act of 1969 ("NEPA"), and the rules and regulations of the Commission. The Board affirms its prior conclusions that the Staff's NEPA review has been adequate and that NEPA, Section 401 of the Federal Water Pollution Control Act, Appendix D of 10 CFR Part 50, and 10 CFR Part 51 have been complied with.

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[8] Our dissenting colleague takes us to task in footnote 8 for an apparent inconsistency between our position regarding the need for fuel contract prices in the *Wolf Creek* case and the lack of need for a contract here. We agree that an explanation is appropriate. The *Wolf Creek* order was prepared and issued by the Chairman with the agreement and general concurrence of the other Board members, but they were unavailable to give it a detailed review. The Chairman is, of course, authorized to do this. Upon careful review, we find that we would not have selected the same words he did. In particular, we disagree with the following:

> "The Board attaches considerable weight to the necessity for actual cost information for the cost-benefit analysis required to be made under NEPA and the Commission's regulations . . . The Board finds it difficult to conceive of a valid cost-benefit analysis being based upon someone's estimate or guess at what the market price is, or might be . . ."

We did not disagree with the ultimate order—that the fuel supply cost terms of the contract be disclosed—for a reason that clearly distinguishes the two cases: in *Wolf Creek* the Applicant was at that time basing its fuel cycle cost estimates on the Westinghouse contract; in *Callaway* it is not.

The Board further finds that the record in this proceeding contains sufficient information to support the following conclusions:

A. In accordance with the provisions of 10 CFR §50.35(a):

(1) The Applicant has described the proposed design of the facility, including, but not limited to the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;

(2) 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;

(3) 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 question associated with such features and components; and

(4) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the Application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

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

C. The Applicant is financially qualified to design and construct the proposed facility.

D. The issuance of permits for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

E. Subject to the conditions set forth in the Partial Initial Decision:

(1) The Environmental review performed by the Staff (pursuant to the National Environmental Policy Act of 1969) and set forth in the final Environmental Statement has been adequate.

(2) Sections 102(2)(A), (C) and (D) of NEPA, Appendix D of 10 CFR Part 50, and 10 CFR Part 51 have been complied with.

(3) The Board has considered the final balance among conflicting Environmental factors, and has weighed the various benefits against costs, taking account of the need for power, and the alternatives to the plant and certain of its design features. As a result, the Board concludes that these considerations favor the issuance of construction permits for the facility.

VII. ORDER

On the basis of the Board's findings and conclusions in its Partial Initial

Decision and this Initial Decision, and pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations, IT IS ORDERED that the Office of the Nuclear Reactor Regulation is authorized to issue to Union Electric Company permits to construct Callaway Plant, Units 1 and 2, consistent with the terms of this Initial Decision.

IT IS FURTHER ORDERED, in accordance with 10 CFR §§2.760, 2.762, 2.764, 2.785 and 2.786 that this Initial Decision shall become effective immediately and shall constitute, with respect to the matters covered therein, the final action of the Commission forty-five (45) days after the date of issuance thereof, subject to any review pursuant to the Commission's Rules of Practice. Exceptions to this Initial Decision may be filed by any party within seven (7) days after service of this Initial Decision. Within fifteen (15) days thereafter [twenty (20) days in the case of the Staff] any party filing such exceptions shall file a brief in support thereof. Within fifteen (15) days of the filing of the brief of the Appellant [twenty (20) days in the case of the Staff], any other party may file a brief in support of, or in opposition to, the exceptions.

THE ATOMIC SAFETY AND
LICENSING BOARD

George C. Anderson, Member

Lester Kornblith, Jr., Member

Dated at Bethesda, Maryland
this 8th day of April 1976.

Samuel W. Jensch, Dissenting:

I agree with my colleagues respecting radiological safety considerations and financial qualifications determinations. I disagree with their conclusion that there is no need at the construction permit proceedings for an adequate and effective fuel supply contract.

In view of the uncertainty of a delivery of an adequate fuel supply for the proposed plant operations, it is my opinion that the present state of the record is sufficient only for the issuance of an additional Limited Work Authorization (LWA). The added authority to Union Electric Company (Applicant) would permit construction to continue, subject to 10 CFR Section 50.10(e), and would also permit Applicant to develop an adequate fuel supply.

The majority opinion herein, as well as the Applicant and the Staff, contends that neither the Atomic Energy Commission had, nor the Nuclear Regulatory Commission has, a requirement that an applicant include a contract for an adequate fuel supply in support of its application at the construction permit

stage of proceedings. The majority imply that a contract should be shown at the operating permit stage; but, again, the Staff and Applicant's argument appears to be that nowhere in the regulations is there a requirement for a contract for an adequate fuel supply.[5] I believe that the regulatory practice established at both Commissions is to expect, and impliedly require, a showing of the existence of an adequate fuel supply so that the finding can be made, at both the construction permit and operating permit stages of licensing, that an applicant can construct and operate a proposed nuclear plant.

Prior to the enactment of the legislation permitting private ownership of uranium fuel for power reactors, the Atomic Energy Commission (AEC) allocated adequate fuel supplies for the life of a proposed nuclear plant. That allocation request was made at the time of the construction permit proceedings. That practice reflects both the request of an applicant for a permit for a long term fuel supply, and the favorable action by the AEC, if safety design and financial requirements are met. When reactor applications began to increase in number, about 1960, the AEC was also confronted with an increasing supply on hand of uranium fuel, so much so, that the AEC restructured its sponsored uranium exploration program by modifying contracts for procurement and extending the dates for receipt of yellow cake and payment therefor.

The legislative history of the private ownership bill reveals that explorers, producers and processors desired long term commitments in order to continue their efforts in developing supplies. The AEC had been considering private ownership of fuel and the combined efforts of the explorers, producers, processors, utility applicants, and the Atomic Industrial Forum resulted in the private ownership bill. The hearings before the Joint Committee on Atomic Energy and the legislative discussions in both the House and Senate are full of expressions of the need for long term commitments being made on the producer side of the supply equation. Such expressions are also equated with other views that the private ownership bill would enable utilities to make long term com-

[5] Many will be amazed to learn that the Regulatory Staff takes the position that there is no requirement for a power reactor applicant to possess a fuel supply contract at any time in the proceedings for a license. The majority opinion implies that the fuel contract need not be demonstrated at the construction permit stage. Since the only other time for evidentiary review is at the operating permit stage, it is inferred that at that time, the contract should be presented. If this is not a correct inference, then it must be that the majority concludes that at no time need a fuel contract be demonstrated in order to procure a license from the Commission. The dissenting opinion, based upon the legislative history, the practice in licensing work, and the necessity for data for findings to comply with the regulations, shows that there is a compelling need for proof of a fuel contract, at the outset, or construction permit stage of the proceedings, just as it is necessary to provide at the outset the proof of financial capability to construct and operate a facility.

mitments for procurement of adequate fuel supplies in order to make accurate analyses of costs of operation over the lifetime of a proposed nuclear facility. The central theme, thus, was for long term commitments of uranium fuel supplies.[6]

Registration statements filed at the Securities and Exchange Commission (SEC) by utility applicants or operators of nuclear power plants reflect varying terms of contracts for fuel supplies. The full disclosure principle of the SEC is of some assistance in providing recognition that long term contracts for fuel supply are part of the showing needed for financial support for the securities proposed to be issued by an S-7 registrant.

In this instant *Callaway* proceeding, as the majority herein point out, Union Electric (Applicant) started out on this project for a construction permit by showing that it possessed a 21-year fuel supply contract with Westinghouse. There is evidence in the record that the 21-year contract is subject to the Westinghouse contention that it is no longer obligated to deliver fuel under that contract because of alleged unanticipated changes in economic and price conditions. Union Electric presented two alternatives: (1) an Order of the United States District Court accepting a stipulation of the parties to require Westinghouse to deliver specified amounts to several utility buyers from supplies

[6]Former AEC Commissioner Robert Wilson was quoted by Senator Morton to express the objective of the legislation as follows:

"Third. It would provide the utility industry and the atomic equipment industry greater assurance as to their *long range costs over the economic life of atomic power plant.*

"Fourth. It would allow, and *eventually require electric utilities* to obtain nuclear fuel under conditions comparable to those for other fuels, and thus permit a more realistic comparison of the true competitive aspects of nuclear and conventional power." (Emphasis added) *Congressional Record*, March 16, 1964, page 5141.

Other expressions urging adoption of the legislation are as follows.

Congressmen Craig Hosmer:

"It is now imperative that this bill be enacted promptly.

* * * *

"First, as nuclear power assumes greater importance in the power economy of the United States, *utility companies and atomic energy industry must be able to plan on a long term basis under normal economic rules. This is particularly true with respect to commitments for fuel.*

"The enactment of this legislation will allow the utility companies to execute *long term contracts* for fuel and thus *to project, with a reasonable degree of certainty, the fuel costs over the life of a nuclear plant.* This long term planning could be done under the same free enterprise conditions which exist in the case of alternate sources of energy." (Emphasis added) *Congressional Record*, August 18, 1964, page 19, 516.

Congressman John Anderson:

"For the utility company, it means a new ability to make *long-term commitments for nuclear fuel under economic* conditions comparable to alternate sources of energy." (Emphasis added) *Congressional Record*, August 18, 1964, page 19, 518.

Westinghouse has "... on hand or on order ..."[7] and (2) Union Electric is endeavoring to execute an additional fuel supply contract, but the record does not reflect, as presumably it would by a supplemental filing, that any further supply has been procured.

The sum of it all is that Applicant does not have an adequate fuel supply contract. The necessity and scope of the contract can be measured by Applicant's endeavors (1) for a 21-year supply at the outset, and (2) the search for some alternative source of supply.

For the license sought in this proceeding, the application is in the usual form; it requests a license to construct and operate a nuclear power facility.

The Atomic Energy Act requires (Section 182) that an applicant for a license shall state "... information of the amount, kind and source of special nuclear material required ...." That requirement apparently developed at the time that the AEC allocated fuel supplies; but, likewise still applicable, the fuel required is that available from contracts executed for a supply. The regulations of the Commission require certain data to be supplied in an application for a construction permit (10 CFR Section 50.35, *et seq.*). Provision is made, however, that even if all data are not supplied, an updating can be made at the time of the consideration of the issuance of an operating permit. The data allowed to be omitted at the construction permit stage are limited and specifically identified; a fuel supply contract is not within the permitted omissions. The updating at the operating permit stage is largely related to design data and research and development additions to the construction permit presentations. The theme asserted in proceedings involving intervenors is that the basic process is the construction permit stage, and at that time contentions that apply to all facets of the proceeding must be asserted by persons, who participate as intervenors, so that the later operating permit hearings should only relate to updating or significant additions to the basic data previously presented. This same rule that applies to intervenors should be equally applicable here. It is a reasonable inference that the required fuel supply contract must be shown at the construction permit stage because of the basic character of fuel supply contract. The majority recognize this basic character but they believe the contract need not be produced now, and thus they would enlarge the scope of the operating permit proceeding.

[7]Union Electric is a SNUPPS applicant. Kansas Gas and Electric is likewise a SNUPPS applicant. The latter interprets the Court Order on the Westinghouse supply in its S-7 SEC registration 1976 statement as follows:

"The Company cannot at this time predict whether the uranium to be delivered pursuant to the Court's order will be sufficient for the initial core load."

The Court Order takes recognition of the fact that Westinghouse apparently does not have "on hand" enough uranium fuel to fulfill its commitments for delivery to the reactor facilities. The Court Order has provided for deliveries for Westinghouse supplies "on order," but that recognition implies the uncertainty whether the supplies "on order" to Westinghouse will actually be delivered or whether economic conditions also relieve the suppliers.

A further aspect on the need for an adequate fuel supply contract is the requirement for an accurate cost-benefit analysis pursuant to the National Environmental Policy Act (NEPA), P. L. 91-190. The majority as well as the Applicant and the Staff are content with estimates of what the market prices of uranium fuel are or will be. The estimates in this record include a wide range and, like many economic predictions, have little validity; NEPA requires the best data that can be made available, and that can be done in this instance by proof of existence of a long term contract.*

The absence of an adequate contract has a particularly acute adverse effect in this proceeding. The effect arises primarily from the rate structure enjoyed by Union Electric. While rates are not jurisdictional issues for the NRC, the effect of operations under that rate structure may have an environmental impact. The rate structure for Union Electric has recently been devised by the Missouri Public Service Commission (MPSC) whose primary and exclusive jurisdiction in the revenue aspect is respected by the NRC. The authority recently granted by MPSC to Union Electric permits construction work progress to be added to the plant account at stated intervals. This procedure permits the rates to be high enough to provide for payment of the portions of the plant added to the plant account.* The significance of this rate structure is that the rate payers are paying for a partially or completed plant whether or not service is rendered to the rate payers. The general rule for utility rates is that the level of rates is determined from the costs incurred to maintain and operate electric generating, transmission and distribution systems that are used and useful in the actual rendition of service. A plant under construction of course is incapable of rendering electric service. A plant without an adequate fuel supply, no matter what may be the hopes of eventually securing an adequate fuel supply is likewise incapable of rendering service. The AEC-NRC approach in most safety and operating contemplations for a nuclear power plant is to assume a "worst" condition; this is termed a "conservative" approach. Applying that principle, it is conceivable that the *Callaway* plant can be fully constructed and paid for by the rate payers who get no or insufficient (below capacity) electric service from the plant for lack of an adequate fuel supply. The NRC review of the environmental impact of a completed skeleton nuclear power plant is a concern to be entertained in this fuel supply uncertainty.

It is difficult for me to ascertain how the regulatory finding can be made at the construction permit state (that a facility can be constructed and *operated* at the proposed location without undue risk, etc.) without an adequate fuel supply contract** for the term of the projected life of the facility as identified by the Applicant.

Samuel W. Jensch, Chairman

Dated at Bethesda, Maryland
this eighth day of April 1976

[Appendix A has been omitted from this publication but is available in the NRC Public Document Room, Washington, D.C.]

*The identical constituent members of the *Callaway* Atomic Safety and Licensing Board also serve for the *Wolf Creek* plant (Kansas Gas and Electric Company, *et al*). In the latter proceeding, the Board has ruled that fuel contract prices must be disclosed in order to make a valid cost-benefit analysis. These prices must come from a long term contract since the cost-benefit analysis extends over the term of the facility. In seeking a solution to the economic distress for the *Callaway* fuel supplies, it is not readily apparent why the majority herein chose not to discuss these principles adopted for the *Wolf Creek* facility.

Until this dissenting opinion was prepared, and even during the discussions in prelude to the majority and dissenting opinions, wherein cost benefit subjects were included, the majority expressed no disagreement with the *Wolf Creek* order. The majority now state they disagree with two sentences but reaffirm their agreement with the order to require disclosure of actual fuel prices to be used as costs. The office arrangement in preparation of drafts of orders and decisions (without receipts) indicate that the local member of the Board received a draft, and he concedes he may have received a draft but did not read it. Whatever be the complaint of the majority, it is no adequate explanation for voting on a subject or principle one way in one case, and voting to the contrary and opposite way on the same principle in another case.

*The general rule adopted for rate making,and in effect before some recent innovations, was to permit all costs of plant construction to be recovered by capitalizing interest costs. When the plant was ready to render service, the entire cost was added to the rate base and consumer rates were then established.

**It is somewhat paradoxical to observe the Regulatory Staff's insistence, rightfully, for a long term contract for a cooling water supply for the companion SNUPPS case (*Wolf Creek*) compared with their lack of insistence for proof of a long term fuel contract (which of course is for a reactor facility that needs an adequate supply of cooling water!).

Cite as 16 NRC 1826 (1982) LBP-82-109

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

James P. Gleason, Chairman
Glenn O. Bright
Dr. Jerry R. Kline

In the Matter of **Docket No. STN 50-483-OL**

UNION ELECTRIC COMPANY
(Callaway Plant, Unit 1) **December 13, 1982**

In a Partial Initial Decision, the Licensing Board rules that isolated construction deficiencies do not show a pattern of a programmatic breakdown in Applicant's quality assurance program. The Board determines that pending a resolution of emergency planning contentions and the making of requisite findings by the Director of Nuclear Regulation, the Director would be authorized to issue an operating license for the Callaway Plant, Unit 1.

QUALITY ASSURANCE: CONDITIONS ADVERSE TO QUALITY

A lack of knowledge that quality deficiencies have been recorded by Applicant's construction contractor represents a failure in meeting quality assurance criteria under Commission's regulations in 10 CFR Part 50 Appendix B.

QUALITY ASSURANCE: RECORDS

Documented reinspection results where the objective is to discover the extent of a problem that could affect quality is a requirement of the Commission's quality assurance criteria.

QUALITY ASSURANCE: ACCEPTANCE CRITERIA

Where quality control inspectors provide reports three months after the reported event occurred, under circumstances where the information contained in such reports is similar and only a single inspector noted comments thereon, such documents are considered worthless.

QUALITY ASSURANCE: PROGRAM ADEQUACY

A proof of the adequacy of quality assurance activities can be ascertained by comparing actual performance against functional standards established in the Applicant's program.

TECHNICAL ISSUES DISCUSSED

Construction Deficiencies
- Materials Integrity and Safety
- Concrete Density
- Welding Defects
- Substandard Piping
- Radiographic Technique
- Code Enforcement

APPEARANCES

Thomas A. Baxter, Esq., Richard Galen, Esq., Joseph E. Birk, Esq., for the Applicant

Roy P. Lessy, Jr. Esq., Robert Perlis, Esq., for the United States Nuclear Regulatory Commission

Kenneth M. Chackes, Esq., Kay Drey, for the Joint Intervenors (Coalition for the Environment, St. Louis Region, Missourians for Safe Energy and the Crawdad Alliance)

A. Scott Cauger, Esq., for the Missouri Public Service Commission

ENCLOSURE 4

I. INTRODUCTION

This is a partial initial decision on an application of the Union Electric Company of St. Louis, Missouri for an operating license at its Callaway facility. The Applicant is one of five utilities that jointly submitted construction permit applications under a Standardized Nuclear Unit Power Systems (SNUPPS) option which provides for, within the Commission's standardization policy, a simultaneous review of the safety-related parameters of a limited number of duplicate plants.[1]

The Applicant's construction permit was granted by the Commission on April 16, 1976 and notice of an opportunity for a hearing on the operating license application was published in the *Federal Register* on August 26, 1980.[2] Intervention as parties was granted to the Joint Intervenors (Coalition for the Environment, Missourians for Safe Energy and the Crawdad Alliance) and John G. Reed, an individual. Representatives of local governments and the Missouri Public Service Commission were also admitted to the proceeding. Nineteen contentions on quality assurance, environmental protection and emergency planning issues were admitted by the Board. However, specification of emergency planning contentions was deferred pending development of offsite response plans and the contention on environmental issues has been withdrawn by its sponsor. This decision then is concerned only with the quality assurance controversy.

Of ten contentions alleging failures in the Applicant's and contractor's quality assurance programs, three were eliminated from the proceeding by motions for summary disposition granted by the Board. An evidentiary hearing was held during November and December of 1981 and time was provided for limited appearance statements. The Joint Intervenors (Intervenor), sponsor of this contention, provided no direct testimony, relying on extensive cross-examination for proof of its allegations. The Applicant and Staff provided individual or panel witnesses on all contentions and a complete list of witnesses is included in Appendix I.

[1] See 10 CFR Part 50, Appendix N.
[2] Clarification of this notice was published in the *Federal Register* on November 21, 1980 after Applicant indicated it intended to proceed only with Unit 1

II. OPINION

The Joint Intervenors in this proceeding allege a failure in the Applicant's quality assurance program in that various inspection and surveillance functions were inadequately performed. And as this negligence, Intervenor contends, constitutes a breakdown in meeting or satisfying the criteria of 10 CFR Part 50 Appendix B, the quality of safety-related materials cannot be assured, thus jeopardizing the safe operation of the plant. Accordingly, the Joint Intervenors conclude, the facility should not be licensed to operate. The inadequacies were attributed to the Applicant, the Bechtel Power Corporation, the Daniel International Corporation, and various inspectors among others. In support of their claim of deficient performances, Joint Intervenors submitted six contentions which were at issue in the hearing. We treat the contentions here *seriatim*.[3]

A. 1.A. Embedded Plates

The embedded plate contention consumed a major part of the hearing sessions on Intervenor's quality assurance issue. The deficiencies cited are that defective embeds were fabricated and installed in the plant contrary to the requirements of the quality assurance criteria established in 10 CFR Part 50, Appendix B. In support of its claim, Intervenor states that inspection, surveillance and testing activities were improperly performed and that exceptions to welding code requirements were also improperly permitted.

The essential assertion of the Intervenor is that embedded plates with possible defective welds were installed in the facility and, in the event of an emergency, weld and plate failure could result in the collapse of critical structural members and piping systems necessary for a safe shutdown of the plant.

Embedded steel plates with welded studs or anchors to transmit forces between steel and concrete have been utilized in construction projects like Callaway for many years. *See* Tr. pp. 947-957. At Callaway, the Bechtel Corporation, the Applicant's principal architect/engineer, provided for the use of two different plates in its construction specifications for the facility. The plates' function is to support various members such as steel floor beams, HVAC components and piping and are installed as fixtures in concrete walls through the use of welded steel studs and welded steel anchor rods embedded in the concrete. Studs are welded to embed plates by an automatic machine operation while anchor rods are attached through a manual weld process. (Board Findings 1, 2, and 3)

The Applicant is one of several utilities that participated, under NRC regulations, in an organized Standardized Nuclear Unit Power Plant System (SNUPPS) for the construction and operation of a nuclear power plant. Under the SNUPPS concept and the regulations in 10 CFR Part 50, Appendix B, the SNUPPS organization, the Applicant, Bechtel Corporation and its plate supplier, the Cives Steel Company, and the Applicant's construction contractor, the Daniel International Corporation, have quality assurance programs and responsibilities. (See Schnell Testimony, ff. p. 216 at 4-34. In exercising these responsibilities, the Bechtel Corporation is required to provide a quality surveillance during fabrication of embedded plates, the Cives Company is charged with inspecting all plates to contract and code requirements and the Daniel Corporation had the responsibility of a receipt inspection, generally limited to verifying the quantities of plates received on site and damages occurring during shipping. On July 8, 1977, subsequent to the commencement of the investigations referred to herein, the Daniel Corporation was also directed to perform a quality control receipt inspection of all Bechtel procured safety-related items. (Board Finding 4)

As a result of an NRC routine inspection on June 9, 1977, stop work orders were issued by the Daniel Corporation on installing embedded plates after machine welded plates with apparent defects were discovered. The plates, which did not have full 360 degree flash material, had not been tested as required by the American Welding Society (AWS) code. (Board Finding 5) The orders stopped work, pending a complete reinspection, of all uninstalled plates, manual as well as machine welded. Up to that point, facility construction was less than 10 percent complete and 255 machine welded plates and 225 manual welded plates had been installed in support of safety-related loads. (Board Findings 6 and 7) Staff, Tr. 1397. The Intervenor argues that failures on the parts of Bechtel Corporation and Daniel Corporation employees to discover welding defects and to perform quality control inspections on machine welded plates were infractions of 10 CFR Part 50, Appendix B criteria. *See* Intervenor Proposed Findings, pp. 3-5. Although the Intervenor refers correctly to Licensee's failure in meeting Criterion X as being cited by an NRC inspector (Intervenor's Ex. 28, p. 7) this deficiency was overcome by the NRC's later agreement that plates installed before June 9, 1977 met requirements. (Staff Ex. 6) The Intervenor misconstrues the separate and different inspection responsibilities of the Daniel and Bechtel Corporations. *See* Schnell Testimony, *infra* and Applicant Embed Testimony, ff. 540 at 13-14. In connection with Intervenor's claim that missing documentation regarding the manufacture of embeds was also in violation of quality assurance criteria, this documentation was supplied and the matter closed satisfactorily to NRC inspectors. (Staff Ex. 4, p. 6)

Prior to welding deficiencies alleged to exist in June 1977, the SNUPPS organization quality assurance manager brought to the attention of the Applicant certain deficiencies in Cives manufactured materials that were being reflected in Bechtel inspection documents. The Intervenor alleges that this evidences early warnings of manufacturing defects which did not result, as they presumably should

[3] The complete text of the contentions is included in Appendix II.

have, in corrective changes in procedures by the Bechtel and Daniel Corporations. This could have, the Intervenor believes, resulted in stopping the installation of defective embedded plates by at least six months. On receiving the above information from SNUPPS, the Daniel Corporation was directed to inspect 10 percent of the embed plates on site, which produced a finding that only four pieces out of 374 pieces inspected had deficient welds. However, the defects were considered minor in nature and the conclusion was reached that the material was being manufactured according to the quality required. Tr. pp. 1234-36; Joint Intervenors Ex. 18. One of the pieces found deficient, a door frame, was later found to be acceptable and was installed, an explanation found satisfactory to NRC inspectors. *See* Intervenor Proposed Finding 14 and Intervenor Ex. 34, p. 10. (Board Finding 7)

After the stop work orders were issued in June 1977, the Cives Company and the Daniel Corporation were independently directed to reinspect all plates on site that had not been installed, and the Daniel Corporation also was given a continuing added responsibility to provide complete quality receipt inspection of all safety-related material supplied by the Cives fabricator. (Board Finding 8) Inasmuch as the evaluation efforts involving machine welded and manual welded plates used different methods of analysis and review, the results are discussed separately in the Board's opinion and findings. We do this for purposes of clarity, for the record reflects an ambiguity on occasion when cross-examination fails to distinguish between the two kinds of embeds and Intervenor's proposed findings do not, as required by our regulation (10 CFR 2.754(c)), always cite to the record or make clear that some of the exhibits referred to were admitted solely for purposes of impeachment, *e.g.*, Tr. 592-594. In discussing first the machine welded plates and then the manual welded plates, we also follow here the order in which evidence on those issues was presented during the hearing.

Machine Welded Plates

Welded studs on machine welded embeds were required to be inspected to conform to the requirements of the American Welding Society (AWS) *Structural Welding Code* D1.1-75. (Applicant Embed Testimony at 13) The Cives Company had been inspecting the studs correctly pursuant to a section of the Code that required bend testing only on one out of every hundred studs. Bend testing was performed by striking the stud to a fifteen degree angle in a direction opposite to a missing weld fillet. Subsequent to the stop work orders, Cives was required to conform its inspection to a more rigorous requirement where every stud not having a 360 degree weld fillet would have to be bend tested. (Board Finding 10)

The reinspection efforts of the Cives Company and Daniel Corporation, which called for a visual inspection of all studs and bend testing where 360 degree fillet welds were missing resulted in failure rates of 0.08 percent and 0.11 percent,

respectively. This rate compared favorably with industry standards. (Board Findings 9 and 12) The bend test produces much higher stresses on studs and stud welds than the design loads applied to the embedded plates. (Board Finding 11)

In its proposed findings, the Intervenor suggests that NRC quality assurance criteria require, as a protection against bias, that individuals other than those from companies interested in the outcome should be used to do inspections. They also maintain that a single inspector could not have performed all of the stud inspections recorded on several of the days reported. *See* Intervenor Proposed Finding 24 n.8 and Proposed Finding 25. It seems clear that Criterion X prohibits the use of the same personnel for quality inspections where those employees performed the welding tasks in question. This is not the case here. And further, there is no convincing evidence or testimony that only a single inspector was used on the dates indicated by Intervenor. The studs that failed the bend test procedure — 66 out of 81,673 studs inspected — were replaced with acceptable studs. (Applicant Embed Testimony at 20) The low rate of failure during the reinspection was an assurance to the Bechtel Corporation, which designed and contracted for the plates, that the embeds installed prior to June 9, 1977 met the requirements of the welding code. This confidence was based on the fact that the reinspected plates were fabricated by the same Company in the same time framework using the same procedures as the plates installed before June 9, 1977. (Board Finding 13)

A second step in the review of previously installed machine welded embeds was the performance by Bechtel of an engineering analysis, using data from the reinspection effort, to develop the probability of plate failure. This probability, found to be on the order of one in one billion, was analyzed as being the product of the probability of having a defective stud, the probability of the plate with an assumed defective stud supporting a safety-related load, the probability of that load exceeding the plate capacity due to an assumed adjacent ineffective stud, and the probability of the design load actually occurring. (Board Finding 14)

Although Applicant's expert witnesses testified that the analysis represented a very conservative approach and provided additional assurances of the structural integrity of embeds installed prior to June 9, 1977 (*see* Applicant Embed Testimony at 26), the Intervenor challenges the results on several grounds: first, it cites the fact that the NRC did not rely on the analysis; second, no attempt was made to ascertain how many of the installed embeds may have been fabricated in the same time period as those where the greatest number of defects were found during the reinspection program and finally, the analysis did not consider the possibility of multiple stud failures. The evidence does indicate an apparent lack of confidence in the probability analysis by the NRC investigators (*See* Staff Ex. 6, Schnell letter, April 24, 1980, p. 2), but Staff testimony that the analysis was unnecessary to prove the structural integrity of the installed plates went unchallenged. *See* Tr. Gallagher, pp. 1327-28. The record reflects that plates which were generally interchangeable, when assembled at the site, were mixed so that their use

id not necessarily follow a pattern of first in-first installed. (Board Findings 5-16) Accordingly, no consideration could have been given to the period in which uch plates were fabricated. On the multiple stud issue, the evidence reveals that djacent stud failures — the only significant alignment that could affect the tatistical analysis — would result in a plate failure rate only slightly more than ingle stud rate failures. The Intervenor's attempt to demonstrate a fallacy in the nalysis by using a possible defect in all studs on all plates fails with assumption hat other parts of the analysis remain equal, an assumption that is not possible in act since the other parts are dependent on each other.

As a final step in reviewing the integrity of installed machine welded embeds, he Applicant, at the request of the NRC, had live load tests made on six plates installed prior to June 9, 1977. The plates were randomly selected on the basis of ccessibility and the feasibility of mounting a test rig for the plates and their election was concurred in by NRC officials. The tests, under the supervision of he Applicant's expert witnesses, Drs. Slutter and Fisher from Lehigh University, tilized a thirty-ton hydraulic jack which subjected the plates to loads slightly in xcess of their design loads without any evidence of plate or stud failure, cracking r yielding. The tests were also witnessed by NRC personnel. (Board Findings 7-20) The Intervenor alleges a possible bias tainting the tests on the basis of a rior relationship between Applicant's experts and a company that manufactured achine welded studs and also claims that similar tests on embedded plates erformed for the Applicant several years prior to the instant one, were possibly a dry run." We believe these claims to be without merit since the relationship of ormer consultants to a stud manufacturer on a question of the welding integrity of tuds which pass through an intermediate fabricator does not present any possible onflict of interest and we fail to see any problem even if the Applicant had in fact erformed — which we do not subscribe to as happening — the prior tests as an dditional precaution against failure of the installed plates.

Despite contrary allegations in Intervenor's contention, the testimony indicates hat the location and loads of all machine welded embeds installed before June 9, 977 are known. Of significant importance to the issues in this controversy is the onservative design of embed plates and studs with maximum load capacities roducing a minimum safety factor of at least 2.0 against the applicable limit state f the plates or studs. And since the actual loads applied to an embedded plate are onsiderably less than the allowable load, the factor of safety for a majority of lates is higher than 2.0. Further discussion of this matter and its importance will e found subsequently in this opinion. As a final comment here on machine welded mbeds, the Applicant testified, in response to one of the allegations in Intervenor's contention, that the possibility of a plate failing to support "critical ping" systems was remote since such systems were designed to have a safety ctor of 1.64 against their yielding state and a much greater factor against xceeding their limit state. In addition, the systems required to maintain a safe

shutdown following a design basis accident are composed of two independent subsystems, both of which could not be affected by the failure of a plate in one system. (Board Finding 21)

Manually Welded Plates

As in the case of studs on machine welded plates, all welding of anchor rods to manually welded plates was required to be in conformity with the American Welding Society (AWS) D1.1-75 Code. (Board Finding 22) After June 9, 1977, the Daniel Corporation undertook a quality inspection of safety-related materials supplied by the Cives Company and found some deficiencies in manual welds; the Cives organization was then directed to inspect all manually welded plates not yet installed and to identify the nature, degree and number of all welding nonconformances. (Board Finding 22)

There is adequate testimony in the record that the precise location and loads on all manually welded embeds installed prior to June 9, 1977 are known. *See* Intervenor Ex. 78. Manual welded embeds are used in the Callaway facility to support structural steel framing members (Board Finding 26), and the anchor rods that are attached to them are welded, due to their large size, by manual instead of machine welding processes. (Board Finding 24) The Applicant's prepared testimony, which was not contradicted, indicated that deficiencies in meeting the welding requirements of AWS D1.1 were due to the physical difficulty of a welder maintaining proper access and rod orientation in welding among multiple anchor rods. It was Bechtel's conclusion that the Code requirements were not developed for the type welds involved in the Callaway construction, and as will be discussed later, exceptions to relevant sections of the Code were recommended and approved. (Board Finding 25) Manually welded plates, like machine welded plates, had included in their design load capacity a minimum safety factor of 2.0 against the yield limit state of the plates and tensile capacity of the anchor rods. (Board Finding 27)

The Bechtel Corporation was advised by Cives during its reinspection that welding deficiencies were found in manual plates which were not in conformity to Code welding requirements. The deficiencies discovered were insufficient weld size, unequal leg size, unacceptable weld profiles (excessive convexity) and excessive undercut. (Board Finding 28) Based on this information, Bechtel performed an engineering evaluation on what was considered to be a "worst case" basis — namely, it developed new load carrying capacities by assuming that all anchor rods had ⅝ inch undersized welds for the total perimeter (360 degrees) of the rod, both legs of the weld were undersized and that all rods had a 1/16 inch undercut. (Board Findings 29-30) With these assumptions, Bechtel calculated a reduced load carrying capacity for each plate which, when compared to the actual applied loads, demonstrated that, in all but four cases, where it was equal, the load

carrying capacity exceeded the actual loads. In its embed testimony, the Applicant stated the load carrying capacity exceeded the design loads in all cases. The Intervenor placed in evidence, however, a letter from Bechtel to the Applicant dated April 9, 1980 which indicated that in four instances, the actual loads equalled the load capacity. (Board Finding 31)

Intervenor in proposed findings has framed a substantial number of its criticisms challenging the validity of Bechtel's engineering evaluation. Summarized, these objections amount to the following: first, Bechtel performed its calculations with limited information from the Cives reinspection effort since Cives had completed only its first of four days of reinspection at the time that Bechtel performed its calculations on reduced load capacities; second, the data reported by Cives on undersize and undercut did not reflect Bechtel's assumptions that the maximum undersize was ⅛ inch and undercut was 1/16 inch; third, the reduced load capacities did not produce an adequate safety margin since at least four plates had a reduced capacity equal to the plate loads and fourteen other plates had actual loads within 96 to 98 percent of the reduced design capacity; fourth, if the installed plates had larger deficiencies than the ⅛ inch undersize, which Bechtel reported to be the critical weld parameter, then plate failure of manually welded embeds could be expected; and last, there was substantial evidence of a significant number of manually welded plates reinspected after June 9, 1977 with an average weld undersize greater than ⅛ inch. *See* Joint Intervenors Proposed Findings, p. 20-27.

Considering Intervenor's arguments in order, the hearing record makes clear that Bechtel did not rely on written reports from Cives' reinspection efforts to document their assumptions on undersize. Instead, Bechtel's engineering evaluation was based on information communicated orally. Tr. 724 (Meyers). The Applicant implicitly concedes Intervenor's position when it proposes as it did in its reply findings that it sees nothing wrong beginning calculations using unconfirmed assumptions subject to subsequent confirmation when final Cives data were available. There is a weakness in this response however since it ignores the fact that Bechtel's final August 10, 1977 report on its investigation of welded embeds — which included the results of its engineering evaluation — actually preceded the termination of Cives reinspection program. (Board Finding 33) It appears that Bechtel may have decided that weld deficiencies were not significant after Cives completed its first day of the reinspection effort since there is evidence that its engineering department did not consider the conditions detrimental to the integrity of the embeds by June 30, 1977. Joint Intervenors Ex. 29, p. 4.

On Intervenor's second issue above, it is argued that Cives data reports reflected limited amounts of information concerning the amount of undersize and no information of the amount of undercut or the amount of undercut that extended around the anchor rod's circumference. Applicant's response is that Cives had been directed to identify only the nature and maximum extent and not the number of such deficiencies. It indicates that it merely sought to obtain the amount of

undersize on the one or more anchor rods with the worst undersize condition. See Applicant Reply to Proposed Findings, p. 15; also Tr. 796 (Meyers). For its engineering evaluation, Bechtel sought to determine, based on information received orally as indicated earlier, what the maximum average undersize and undercut were on individual welds. And it was advised that the maximum oversize was always less than ⅛ of an inch and the maximum undercut always less than 1/16 of an inch and that these deficiencies were never around the full circumference of the weld. Tr. 724, 796, and 1241 (Meyers). Bechtel had the information on undersizing verified at a later date by the Cives Company when it subsequently was evaluating supposedly conflicting data supplied by the Daniel Corporation, which is discussed later in this Opinion. Tr. 796 (Meyers); Board Ex. 1, Encl. 2. We conclude that Intervenor's criticism is well founded to the extent that the Bechtel Corporation had no written support for its assumption that the maximum undersize was always less than ⅛ inch but relied solely on oral communications. And we also conclude that Applicant's letter of March 10, 1978 to NRC's Regional Director is somewhat misleading in indicating a documentation that no deficiencies were undersized greater than ⅛ inch. A communication from the Cives Company that its records do not indicate any welds more than ⅛ undersize is not, in our view, "documentation" as that term is normally understood. Applicant Ex. 6, p. 1 of attachment; Board Ex. 1, Encl. 2. Nor can the Board believe that verification of the same matter from field inspection reports referred to by the Bechtel Corporation in its August 10, 1977 Investigation Report on welded studs was other than by oral communication although a different conclusion would be reached by an ordinary reading of that statement. Applicant Ex. 4, p. 3. Whether these amount to a failure in quality assurance requirements we discuss at a later point in this opinion.

The Intervenor's third point, that revised load estimates presented an inadequate margin for error, was repeated in a number of its proposed findings of fact. See Joint Intervenors Proposed Findings 2, 30, 36, 37, 52 and 53. The case essentially states that the Applicant and Bechtel misrepresented the safety issue by indicating that despite the reduced loading capacity a sufficient design margin still existed and none of the plates embedded prior to June 9, 1977 possessed the potential to fail when in actuality four plates had an equal capacity to the reduced margin and 14 others were within a close margin (96 to 98 percent) of their actual loads. The Applicant's response does not substantially contest Intervenor's numbers but relies instead on the design safety factor of at least 2.0 against the yield limit state of the plate and the tensile capacity of the anchor rods, a determination of safety with which the Board concurs. Additional weight must be given to the fact that other conservative assumptions were made in the engineering analysis. Applicant Embed Testimony at 37. The testimony indicates that even though it is accepted engineering practice to load a plate to full capacity, there, nevertheless, remains a margin of safety in the design for embedded plates that at a minimum is slightly less than 2.0. Applicant Ex. 20, pp. 3-4. The fourth argument of Intervenor — it

Bechtel's assumption about ⅛ inch undersize was erroneous, then plate failure would be expected — merges with its fifth point, that there was in fact substantial evidence of a significant number of plates reinspected with average weld undersize in excess of ⅛ inch. We treat both together because they involve discussion of the reinspection effort conducted by the Daniel Corporation, a controversial subject that occupied much of the Board's time and that of the parties during the hearing.

Subsequent to the discovery of possible welding defects in embeds by NRC inspectors on June 9, 1977, the Daniel Corporation was directed by the Applicant to inspect all embeds procured from the Cives Company. This inspection, covering all embeds at the site and those to be received thereafter, was different from and in addition to the Cives site reinspection which was initiated by Bechtel officials. (Board Finding 34) Some time after Bechtel submitted its final report — based on Cives reinspection information — on the welded embed problem, (August 10, 1977), which concluded that the plates were a completely acceptable product, inspection reports compiled by Daniel which contradicted the Cives data were furnished to Bechtel. The Daniel data indicated the existence of a number of plates with weld undersizes in excess of ⅛ inch. (Board Finding 35) There was extensive interrogation during the hearing on Bechtel's claim that it had no knowledge of the conflicting Daniel information prior to Bechtel submitting its final report or that Daniel was even inspecting plates on site. *See* Tr. 797 *et seq.*, (Meyers, Schnell). Regardless of where the truth lies, it is clear to the Board that if Bechtel did not know of Daniel's activities, it should have had such knowledge. The Applicant had directed the performance of Daniel's inspection work and Bechtel was the Applicant's design architect/engineer. It strains our credulity to believe that the Applicant would not advise Bechtel of an order to Daniel which impacted a Bechtel supplier, *i.e.*, Cives. Nor is it easily understood why Cives would have failed to notify Bechtel of Daniel's activities when both Cives and Daniel personnel were involved in a similar activity at the same place at the same time. There is evidence in the hearing that the Applicant knew in August 1977 of the large number of welding defects in manually welded plates that Daniel was discovering and there is also testimony that Daniel and Cives personnel knew of each other's efforts at the time. (Board Findings 36-37) Whether the Applicant or Bechtel knew of the Daniel information or ignored it, it is clear that it was an NRC investigatory team that initiated the review that attempted to reconcile the Cives and Daniel conflicting data. (Applicant Ex. 7, p. 1; Board Ex. 1, Encl. 4, p. 1) An explanation provided by the Applicant and Bechtel for their lack of information concerning Daniel's inspection results is that the surveillance reports from those inspections were accumulated in a nonconformance report procedure for onsite rework, which, being internal to Daniel, would not ordinarily be communicated to others. Nevertheless, due to the severity of the problem being investigated, in the Board's judgment, a lack of knowledge is not plausible.

In an effort to reconcile the differences between the Cives and Daniel inspections, a two month review was undertaken by both Daniel and Bechtel to clarify the inconsistent data. This resulted in a conclusion by Bechtel that due to poor documentation, inconsistencies and errors, an engineering analysis of the Daniel data was inappropriate and, actually, impossible to accomplish. (Board Findings 38-39) The inadequate reporting of data by Daniel was attributed to the fact that its function during inspection was not to determine the amount or extent of weld deficiencies but to record only enough information to provide a basis for rework if that were required. This information was verified during the hearing by two employees from Daniel who were witnesses called by the Board. (Board Finding 40) In its review, Bechtel found that only 8 plate data reports out of 532 showed an average weld undersize greater than ⅛ inch as, I with information further refined by the Daniel review effort, the Applicant found only 10 of 364 embeds with such an average. Both Bechtel and the Applicant concluded that these data reports were too few and too unreliable to be used to characterize the embedded plates delivered at the site. Applicant Ex. 7, p. 3 and Ex. 6, p. 4. The Daniel Corporation also indicated that an engineering evaluation similar to what Bechtel had performed using Cives data, which assumed the maximum undersized condition around the complete weld circumference, would not represent a true picture of actual conditions. And finally, Bechtel, the Applicant and Daniel reinspected 45 plates that had not been repaired which had been previously rejected by Daniel and agreed that the deficiencies observed were less than reported. (Board Findings 42 and 43)

Joint Intervenors argue, in relation to the Daniel's data, that adequate evidence was presented of a significant number of manually welded plates with average weld undersize greater than ⅛ inch. They point to both Bechtel's and Applicant's reviews of the Daniel data which reflected that 8 or 10 plates contained such deficiencies. Further, they insert in proposed findings a composite table purporting to show, based on Bechtel's calculations, a comparison between Daniel and Cives data which shows 26 plates having average weld undersize in excess of ⅛ inch with the Cives data in agreement with Daniel on 13 of the 26 plates. The Board cannot subscribe to Intervenor's arguments on either point. Leaving aside any discussion whether the Daniel data package, Intervenor Ex. 12, which was admitted for a limited purpose, or the composite table which makes its presence felt for the first time in proposed findings can be relied on for probative evidence by the Board, we believe neither can overcome a more fundamental objection. Neither allegation is substantive in the case. In order to arrive at a finding that Bechtel's assumptions were erroneous, based on Daniel data that either 8, 10 or 26 embedded plates possessed average weld undersize greater than ⅛ of an inch, one has to also assume that the weld undersize extends around the entire circumference of the anchor rod. There is no evidence in the record of the validity of that assumption. In addition, we have the testimony of Daniel's Project Manager, who was responsible for the compilation of the Daniel data, that the original inspection

data were not sufficient for the performance of an engineering evaluation. Tr. Holland, p. 1358. Finally, there is uncontradicted evidence from one of Applicant's expert witnesses that even assuming the validity of the welding defects reported in the Daniel data, with such defects going around the full circumference of the anchor rods, it would not have affected the margin of safety or the load carrying capacity of the plates. In the expert's opinion, the weldments in the embedded plates could have been 25 percent smaller than required by the specifications and he testified that the Code was being changed to effect such a revision. (Board Finding 44)

When Bechtel calculated reduced load capacities, in its engineering analysis, it assumed, in addition to the ⅛ inch undersize, a 1/16 inch undercut for all anchor rods. It also concluded that deficiencies in the welds reported by Cives of unequal weld legs and poor weld profile (excessive convexity) did not affect weld performance. Based on this evaluation, and its analysis that safety of the plates was still assured with a reduced design capacity, Bechtel then requested and received the approval of SNUPPS and the NRC for exceptions to the Structural Welding Code. The exceptions permitted additional allowances for the vertical legs of the welds and undercutting, acceptance of unequal legs and the elimination of profile requirements. (Board Finding 45) Exceptions to Code requirements are the responsibility of architect/engineers on construction jobs since the Code requirements are considered as conservative guidelines. Fisher Tr. 773. The Intervenor challenged Bechtel's finding that a 1/16 inch undercut did not affect the load carrying capacity of the anchor rods since its (Bechtel's) analysis ignored the fact that some anchor rods were unthreaded. However, in addition to the fact that there is no evidence in the record to support this argument, the Applicant successfully counters the objection with its reply that an additional margin of safety is provided in the case of unthreaded anchor rods. *See* Applicant Reply to Proposed Findings, 27-28.

After Bechtel and the Applicant concluded that the conclusions reached in the 1977 report on the acceptability of manually welded plates embedded prior to June 9, 1977 was not contradicted by the Daniel data, the NRC ordered additional testing to be performed. It also inspected visually embedded plates substantially loaded by floor slab dead loads without seeing any signs of distress or overstress. (Board Finding 46) In the tests performed by the Applicant's consultants from Lehigh University, the testing was done to demonstrate the structural integrity of the welds in the 45 plates that had been rejected by Daniel but not as yet repaired. Six anchor rods on six different plates were selected by the NRC for bend testing and the direction of the bend was also directed by the NRC. Six anchor rods on six additional plates were selected by the Applicant for tension testing to their ultimate load. The tests were performed on welds selected as having the worst visual defects and appearances. (Board Finding 47) Six anchor rods were bend tested to 30 degrees without any signs of cracking or failure and six rods were tension tested to

failure with three failing in the weld and the other three failing in the rod itself. The ultimate strength of the welds tested was greatly in excess of their design load. The tests were observed by the NRC. (Board Findings 48-49) The Intervenor argues that the welds selected for the tests were not as defective as the worst weld found in the Daniel or Cives inspections nor did the tests deal with the kind of load that the manually welded plates have to support. We believe these allegations unsupportable since the plates were ones previously rejected by Daniel and contained the worst available welds and, finally, the tests subjected the welds to the same shear forces that exist in the embeds that were embedded prior to June 9, 1977. Fisher Tr. 1150-1151.

Based on the engineering analysis of reduced load capacities, the review of the reinspection data of plates manufactured in the same time period, and the additional testing that was done on manually welded embeds, the Applicant and the NRC both concluded that the plates embedded prior to June 9, 1977 were capable of supporting their design loads. (Board Finding 50)

The Board is obligated in this summary of the manual weld issues to express a deep concern over the handling of the embed plate problem by the Applicant, the Bechtel Corporation and the NRC Staff. That concern relates solely to the manually welded plate controversy since we conclude the review procedure for machine welded plates was prepared satisfactorily and followed adequately. We also, however, need to state our apprehensions over the quality of work performance manifested in these proceedings by the Daniel Corporation and Cives Company. The following reflects the Board's uneasiness as a result of the evidence produced in this hearing:

1. Performing an evaluation of reinspection data without written documentation as the Bechtel Corporation did in this case is not only a questionable procedure but a violation of quality assurance requirements.
2. Assuming the truth of statements by the Applicant and Bechtel that they were unaware of Daniel's inspection data for a period of months — statements of which we have some doubt — this lack of knowledge does not foster confidence in either company that they were carefully monitoring construction developments and progress at the Callaway facility.
3. The inconsistent reporting of weld deficiencies by Daniel inspectors demonstrates a supervisory weakness at the Daniel Corporation irrespective of any instructions that its employees were to inspect embeds on an accept or rework basis.
4. The acknowledgment by the Cives Company that it repaired 20 percent of the 400 embeds it reinspected at the plant site raises, in our judgment, serious questions of the quality of work that was received from the fabricator.

5. Finally, we must express an overriding concern that NRC inspection officials permitted the question of the integrity of embeds installed prior to June 9, 1977 to stay unresolved for such a lengthy period of time. Over three years elapsed from the date of reporting the original suspected weld deficiencies until NRC's final evaluation was transmitted. It is clear from the evidence that the issue of the acceptability and adequacy of the manual welded embeds was referred to NRC Headquarters both in November 1977 and April 1978. It is also clear that no action ensued until the Inspection and Enforcement Office of Region III of the NRC again initiated its own review in April 1980. No information has been provided in the record for this inaction on the part of the NRC Headquarters Office. (Board Finding 51)

Quality Assurance Failures

Many of the activities discussed herein were considered by Joint Intervenors to be violations of quality assurance requirements. We agree with certain of the Intervenor's conclusions, but not with others.

The Board does not concur with Intervenor's argument that testimony showing there was some difficulty with the use of embeds at other nuclear facilities or that there were welding problems at Callaway in meeting Code specifications indicates a failure to establish measures for selecting appropriate materials as Criterion III of 10 CFR Part 50, Appendix B requires. As we have indicated, *infra*, the evidence reflects the industrial usage of embedded plates and also reflects that welding codes are designed to be adaptable to revisions in welding requirements.

The Board has already expressed its judgment that the apparent lack of knowledge concerning Daniel's inspection data was [illegible] in communication and under the circumstances here a potentially se[illegible]lown in the fabrication of safety-related materials, *i.e.*, embedded plates. [illegible]igence represents a violation of quality assurance Criterion XVI of Appendix B. Further, it is the Board's view that the failure to require the Cives Company to produce, in its reinspection effort, a more extensive report of the results thereof was a violation of Criterion XVII. The Cives effort was aimed at finding the extent of a problem that could affect quality. The adequate recording of activities affecting quality is one of the objectives of Criterion XVII. Accordingly, it would have been more in keeping with the purposes of effective quality assurance for an adequately documented record to be compiled of Cives reinspection activities.

The Joint Intervenors also cite the Applicant's failure to notify the NRC of the manual embed deficiencies as is required by 10 CFR Part 50.55(e) for significant deficiencies that need extensive evaluation, redesign or repair. Although the Board has expressed its concern with the length of time NRC took to review the Applicant's handling of the manual embedded matter, we are unable to fault the

Applicant with not filing a 50.55(e) report since it completed its original analysis of the problem in a prompt manner. It is a moot question, in any event, since the NRC certainly had notice of possible embedded deficiencies from the date its own inspectors reported welding defects in June 1977.

Outside of the failures reflected herein, the Board finds no other negligence affecting quality assurance requirements on the embed contention. The Applicant did undertake an analysis of the welding deficiencies, effected a revision in Code specifications to accommodate changes in the fabrication of embeds that were found necessary and also imposed stricter inspection requirements. These actions, occurring shortly after the suspected problem of welding defects, were compatible with proper quality assurance activities. Whether the Applicant's actions were adequate to assure the safe functioning of the embeds installed at the facility before June 9, 1977 we discuss in our conclusion below.

Conclusion

As we have indicated above, the Board finds a serious disregard by the Applicant and its major contractors for quality assurance considerations in the handling of certain aspects of the manual welded embed plate problem. The question before us is whether or not these activities constitute such a level of negligence that the quality of safety-related materials cannot be assured and as a consequence, the safe operation of the Callaway plant is thereby threatened. The central issue of this contention then is: If the assumption used by Bechtel in its engineering evaluation, that no weld deficiency was more than ¼ inch undersize was an incorrect assumption, could plate failure among those embedded prior to June 9, 1977 be anticipated. We think not. We conclude the Applicant has carried the burden of proof on this issue on the following grounds:

1. A factor of safety has been designed into the load capacities of both machine welded and manually welded embeds of at least 2.0 against the yield limit state of the plate and the tensile capacity of studs and anchor rods.
2. The actual loads imposed on most embedded plates is considerably less than the allowable design capacity, thereby increasing the design safety factor to more than 2.0.
3. Tension tests to failure were performed on weldments of manually welded plates that had been rejected by Daniel inspectors but not repaired and a safety factor in excess of 3.0 was demonstrated.
4. Evidence that welding code revisions will permit weldments in the future to be 25 percent smaller than those required for the embed plates at Callaway.
5. Testimony of Dr. J. W. Fisher, an expert in weldments and structural analysis from Lehigh University whose opinions and competence were

persuasive to the Board, that, even assuming the validity of the Daniel data with the weld defects reported extending completely around the circumference of the anchor rods, that neither the required margin of safety nor the load carrying capacity of the manually welded embeds would have been affected.

Finally, we also conclude that the Applicant has submitted adequate evidence of the safety of machine welded embeds based on the inspections conducted by the Cives and Daniel organizations, the probability analysis carried out by the Bechtel Corporation and the load tests performed successfully on selected machine welded plates installed before June 9, 1977.

B. I.C.1. Honeycombing In the Reactor Building Base Mat

In this contention Joint Intervenors challenge whether there exists adequate assurance that there are no defects in concrete of the reactor building base mat beyond those already known. They challenge the reliability of the methodology that was used for testing the base mat as well as the inferences drawn from the tests. Additionally, they question the adequacy of the quality control and quality assurance procedures that were used before, during and after the placement of concrete for the base mat. Relevant to this consideration is whether individual deficiencies in quality control are sufficiently numerous and serious that when considered collectively they would indicate a failure or breakdown of the Applicant's quality assurance program as a whole.

The reactor building base mat is a flat circular slab of concrete which is 154 feet in diameter and 10 feet thick. The base mat serves as a foundation and base for the reactor building. The tendon access gallery is located directly below and along the outside edge of the base mat and continues around its circumference. The lower surface of the base mat forms the ceiling of the tendon access gallery. Tendons are steel cables which cross over the reactor dome for the purpose of applying compressional stress to the reactor building shell and dome after the concrete hardens. The tendons are anchored at both ends by steel structures called trumplates. The trumplates are embedded in the concrete ceiling of the tendon access gallery. There are 172 trumplates in the Callaway reactor building. Applicant Base Mat Testimony, ff. 227, at 9-11 and Figure 1; Varela Testimony, ff. 396, at 2-3.

Concrete for the reactor building base mat was placed over a 62-hour period from April 6 to April 9, 1977. Six thousand seven hundred and twenty (6720) cubic yards of concrete were used. Two shifts, each involving approximately 190 construction crafts, engineering, quality control and supervisory personnel, were used in alternate fashion to accomplish the concrete placement. Three NRC Staff inspectors were present during the concrete placement and one or more of these inspectors observed most of the operation. (Board Finding 52)

Honeycombing in the concrete ceiling of the tendon access gallery was found by construction personnel after the concrete had hardened and the forms and shoring were removed. Honeycombing is a defective condition in hardened concrete which consists of small voids dispersed through the concrete giving it a popcorn appearance and, of course, causing it to be weakened. The defect may or may not be a serious safety concern depending on its location and extent. Honeycombing that was found on the ceiling of the tendon access gallery, which is part of the reactor building base mat, was taken as a serious matter by both Applicant and Staff since there were 14 locations where it undermined the base of the tendon trumplates which ultimately would be loaded in excess of 1,400,000 pounds each. (Board Findings 53, 54, 55)

The honeycombed areas were chipped out to sound concrete to determine the extent of the defect at each location and to prepare the cavities for repair. Repairs were done by filling each cavity with a high strength grout which formed a bearing surface for the trumplates having a strength at least as high as the sound concrete of the base mat. (Board Findings 54, 61)

The Applicant had performed nondestructive testing of the concrete in the base mat above the tendon access gallery to determine whether hidden concrete imperfections existed in the interior concrete above the trumplates. Testing was performed with a soniscope which measures the velocity of sound as it travels through concrete. High sound velocity (above 12,000 feet per second (ft/sec)) indicates uniform dense concrete; low velocities (5000 ft/sec) reveals poor concrete containing voids, honeycombing or cracks. Soniscope testing is an accepted technique that has been utilized for more than 15 years. (Applicant Base Mat Testimony, at 23-27; Varela Testimony, at 5)

Multiple soniscope measurements were taken at each of 44 trumplates in the tendon gallery which represent 25 percent of the trumplate locations in the tendon gallery. Seven hundred and sixty (760) individual measurements were made of which 103 were unsuccessful. Unsuccessful shots or signals are caused by poor contact between a transducer and rough concrete or by a minute plane of separation between concrete and a steel plate which blocks the signal but does not indicate internal voids or honeycombing. (Applicant Base Mat Testimony, at 25-28; Applicant Ex. 2, pp. 15-17)

Measurements were taken: (1) vertically through concrete, (2) vertically through concrete and a steel plate which was part of the trumplate, and (3) at an angle through the concrete behind the trumplates. All measurements showed velocities in excess of 15,000 ft/sec. Coefficients of variation ranged between 1.11 and 2.1 percent for these measurements, indicating reliable and reproducible data. (Applicant Ex. 2, pp. 15-19)

The tests showed: (1) concrete above the gallery and the trumplates uniform in composition and strength, (2) concrete tested has a high compressive strength, (3) no evidence of internal honeycombing in concrete in the 44 trumplate areas, and

(4) the results indicate that, based on a 25 percent sample, internal honeycombing probably does not occur in the base slab. (Applicant Base Mat Testimony, at 25-28; Applicant Ex. 2, p. 21) (Board Finding 60)

Staff concluded the test was appropriate and the number of sample locations was conservative. (Varela Testimony at 6, 8; Staff Ex. 5)

Our review of the evidence pertaining to causes of honeycombing, repairs, soniscope testing, and loading of structures in the tendon access gallery lead to the conclusion that there exist reasonable assurances that no concrete defects of importance to safety exist in the tendon access gallery portion of the base mat. Intervenor's issue related to dry-pack repairs was found to be insignificant to the safety of the base mat. (Board Findings 64, 65, 66, 67, 68)

The tendon access gallery, however, constitutes only 19 percent of the entire base mat. Because of the nature of the soniscope which was used for nondestructive testing, it could not be used to test the remaining 81 percent of the base mat. Thus, if reasonable assurance of integrity of the remaining base mat is to be obtained, it must be done by indirect means since few direct observations exist to confirm it.

The indirect evidence for base mat integrity, however, is substantial. The physical evidence from the tendon gallery shows that honeycombing was a surface phenomenon. All that was ever discovered was first discovered by visual observation of surfaces. The soniscope tests for interior defects throughout the full thickness of concrete, on the other hand, did not reveal a single instance of faulty concrete above the tendon gallery. (Board Findings 56, 60)

The cause of surface honeycombing in the tendon gallery is due principally to inadequate vibration of fresh concrete in areas specially congested and hampered by steel embedments. Since no such cause existed in the interior of the base mat, there is no reason for suggesting that concrete was not vibrated adequately during placement. (Board Findings 57, 58, 81)

The base mat placement began on one side of the base mat and progressed to the other without special reference to the tendon gallery. It is unlikely that substantial changes in materials, workmanship, equipment, or methodology could take place unnoticed during such a relatively short intense work period which would lead to either better or worse workmanship focused in the tendon gallery as compared to the rest of the base mat. We therefore conclude that the base mat pour was accomplished with reasonable consistency and uniformity of materials and workmanship from beginning to end. The direct observations of the integrity of interior concrete in the tendon gallery, therefore, apply to the base mat as a whole even though the sampling technique used by the contractor cannot be regarded as a true random sample of the entire base mat.

Similar reasoning leads us to conclude that we cannot rule out with certainty the existence of honeycombing on the lower surface of the base mat (*i.e.*, that portion resting on earth and therefore inaccessible to inspection). We note, however, that

the accessible surfaces of the base mat (top and vertical sides) were inspected and no imperfections were found. (Board Findings 79, 83)

The frequency and magnitude of honeycombing in the tendon gallery gives us a fair sample of what reasonable constancy in materials and workmanship might produce on other lower surfaces in the base mat. Considering, however, that the tendon gallery was more congested with embedded steel items than the lower surface of the base mat in general, we conclude that whatever honeycombing might exist on the remaining lower surface its frequency and magnitude are not likely to exceed those already found. (Board Finding 81)

The honeycombing that was found, however, would be harmless to the overall safety of the reactor building if located in other parts. The concern in the tendon gallery stemmed from the fact that honeycombing undermined the base plates of some special structures — the tendon trumplates — which would ultimately carry very high loads. No such special load bearing structures exist over the lower surface of the base mat in general. The Board therefore concludes that even though honeycombing of the general frequency and magnitude as that already found might exist on the lower surface of the reactor base mat, such imperfections would not jeopardize the function of any special load bearing structure nor would they create a concern for the general integrity of the base mat. (Board Finding 84)

Adequacy of Applicant's Quality Assurance and Quality Control Procedures

The Board has been unable to find serious defects in the overall quality assurance procedures followed by the Applicant in connection with construction of the reactor building base mat. The evidence shows that the essential components of an overall program were present in this case. These components include preplanning and inspection, supervision and inspection during concrete placement, and inspection, testing and repairs of defects after the task was finished. (Board Finding 74)

It appears that Intervenor's objections stem from an unrealistic assumption that the only acceptable performance of a constructor would be flawless work in the first instance. While that is an important goal we conclude that it is an unrealistic approach to safe construction since prevention of flaws is only one aspect of a multifaceted program. In this instance it is apparent that defects in concrete of the reactor base mat occurred in spite of reasonable efforts to prevent them. However subsequent inspections disclosed the defects, they were reported internally and to NRC, tests were made to determine the extent of the imperfections, and repairs were made. (Board Findings, 66-72)

These are actions we expect of a functioning quality assurance program. A serious deficiency in any of the components might lead to serious questions about

the overall program. However, in this case the program worked properly and it led to the desired result which is a structurally sound base mat.

Joint Intervenors assert a number of reasons why the Applicant's testing may not demonstrate the integrity of the tendon gallery. Their assertion that the soniscope method is faulty because it does not take account of the fact that sound waves may go around defects in concrete reflects a misinterpretation of the physical principles of the instrument. The deflection of sound waves around defects in concrete is the phenomenon which enables the detection of such defects. The added time required for sound to traverse a tortuous pathway relative to an unobstructed pathway is what is measured and what leads to the interpretation of reduced velocity and faulty concrete if it exists. (Board Findings 75-76)

Intervenor's argument that the velocity of sound in steel may account for the high sound velocity measured in tests of the base mat is similarly misguided. Expert testimony shows that the interface between steel and concrete often results in a degraded signal or complete obstruction of the signal. Many of the attempted soniscope measurements failed because they were taken from steel surfaces which were interfaced with concrete. While sound might well have a high velocity in steel, we need not take notice of that fact as urged by Intervenor since it is beyond dispute that the base mat consists of concrete containing embedded steel. It is the existence of concrete-steel interfaces which might influence the velocity of sound; the sound signal may be halted by interfaces or simply go around the obstruction. In either case the result could not be an apparent increase in sound velocity. (Board Finding 77)

Intervenor's discussion of the errors possible in aligning a cross hair on an oscilloscope which is necessary to measure the velocity of sound is also without merit. They assert that the testing report does not discuss the margin of error or with what bias such factors might have been resolved. This is simply inconsistent with the facts, since the Wiss, Janney, Elstner report (WJE was the firm selected by the Applicant to perform testing), Applicant Ex. 2, p. 19, lists a table showing average velocities of sound as transmitted through concrete and standard deviations and coefficients of variation for each average. We expect nonsystematic errors of measurement including instrument reading errors to be reflected in the calculated standard deviations and coefficients of variation. The coefficients of variation actually observed range from 1.0 to 2.1 percent. These errors are sufficiently small to conclude with confidence that the measured sound velocities reliably exceed the threshold of concern (12,000 ft/sec) below which the integrity of concrete could be in doubt. (Board Finding 78)

Joint Intervenors assert that the Applicant's quality assurance program failed to provide proper documentation regarding the reactor building base mat problems. They believe that the documentation that exists demonstrates weaknesses in work procedures and quality control procedures that were governing at the time of base mat placement. Joint Intervenors specifically object that: (1) concrete placement

reports were not submitted by all quality control inspectors after the base mat was poured and (2) the documentation of the existence of honeycombing in the reactor base mat was not submitted in a timely fashion since more than a month passed after completion of concrete placement before a nonconformance report was written. (Board Finding 85)

The Staff inspector who was present at the time of concrete placement subsequently cited the Applicant for an infraction because each of the Applicant's quality control inspectors did not submit individual concrete placement reports. After the citation was issued the Applicant undertook to remedy the deficiency in documentation by having each inspector who was present during the time of concrete placement sign a concrete placement report. Each concrete placement report was similar in information content and appearance and was signed by the individual inspectors during a period covering July and the first part of August of 1977. In the Applicant's view the signatures and the absence of comment on the concrete placement reports provides assurance that the individual inspectors observed no deficiencies during the pour. It could not be ascertained directly from the reports, however, what activities the signature of each inspector was verifying that he had witnessed. (Board Findings 86, 87, 88)

The NRC Staff inspector interviewed some of the inspectors after receiving the concrete placement reports and verified that they had observed no deficiencies during concrete placement. (Board Finding 89)

The Board concurs with Joint Intervenors that the procedure followed here was defective. We criticize the documentation procedure but do not find evidence that the inspectors failed to perform their duties at the time the concrete was actually placed. Specifically, we find that the placement reports signed without comment some three months after the event took place to be essentially worthless. (Board Finding 90)

We are unable to determine from the belated reports whether they were signed by the inspectors in a perfunctory manner as simply another burden of paperwork or whether the signatures have genuine meaning. While some interviews were done by the Staff inspector, the interviews were not documented. Thus, no genuinely useful written record exists which would document the observations of inspectors during the placement of concrete in the base mat. Assuming that such documentation is necessary, as appears to be the case from the Staff citation, we should insist that it be substantive and not a mere paper shuffling exercise. In this instance the Board concludes that an appropriate procedure might have been to interview each inspector and produce a written record of the interview for the inspector's signature. (Board Findings 90, 91)

The Board, however, can find no deficiency with regard to the timing of the nonconformance report. Concrete placement was finished on April 9 and a nonconformance report from the Daniel Corporation to Bechtel was filed on May 11, 1977. In the approximate month between completion of the base mat pour and

the filing of the first nonconformance report, the concrete was left to harden, the concrete forms were then removed, inspections were performed and honeycombed areas were chipped to sound concrete. The chipping operation was an essential prerequisite to determining the extent of unsound concrete. In light of the actions that had to be taken, the elapsed time of one month from the termination of the pour appears reasonable. Upon receiving the May 11 nonconformance report, Bechtel rejected it, requesting more detail before disposition. The Daniel Corporation then proceeded to draw detailed maps of the extent of honeycombing in the tendon access gallery and outlined in detail a proposed repair method. The second nonconformance report which was acceptable to Bechtel was filed on June 27, 1977. (Board Finding 92)

The Board finds no reason for concluding that there was a general breakdown in Applicant's quality assurance procedures. The quality assurance procedures employed in this instance worked properly in that precautions were taken to prevent deficiencies; deficiencies that occurred in spite of the precautions were found promptly; appropriate reports and tests were made; and repairs were made which restored the defective areas to original design specifications. While we found deficiencies as regards the handling of concrete placing reports on the part of both Staff and Applicant and in the requirements for testing dry pack concrete, these appear to be isolated matters and not evidence of gross failure of the Applicant's quality assurance program.

Conclusions

The Board concludes that the Applicant took reasonable steps in advance of concrete placement to prevent the occurrence of concrete imperfections. The concrete imperfections occurred in spite of precautions and not due to neglect of quality assurance. When concrete imperfections were discovered they were reported both to Bechtel and the NRC and reasonable plans were established for testing and repair. The soniscopic testing demonstrated that there were no hidden concrete imperfections in the base mat above the trumplates. The surface repairs that were undertaken assured adequate bearing surfaces for the trumplates, which would ultimately carry very high loads. Loads were later imposed on the trumplates, as high as 1,600,000 pounds without failure of trumplates or distress in concrete. The issues raised by Intervenor concerning the use of dry pack are insignificant and have no bearing on safety of the structure. The Board concludes that there is no reason to doubt the integrity of the reactor base mat or the peformance of the tendon trumplates.

C. I.C.2. Honeycombing in Reactor Building Dome

The Board denied a motion for summary disposition of this contention prior to hearing after concluding that there existed a substantial factual dispute as to the integrity of the reactor dome and quality control procedures that were followed for discovery, reporting and resolution of this matter. Nevertheless, Joint Intervenors announced at hearing that while they did not abandon the contention they would not conduct cross-examination of Applicant's or Staff's witnesses since they did not feel that they could contribute to development of a sound record on this subject.

After reviewing the prefiled testimony of both Applicant and Staff the Board concluded that there were remaining matters to be explored and it called the witnesses for Board examination. The Board questioned the witnesses on both quality assurance procedures and the reliability of the testing and analysis that was done that led Applicant and Staff to conclude that there were no undiscovered defects in the reactor dome which would jeopardize its integrity. The Board found the witnesses of Applicant and Staff to be forthright and credible. No evidence was uncovered in the Board examination which was at variance with their prefiled testimony.

In this contention Intervenor challenges whether there exists adequate assurance that there are no imperfections in the concrete of the reactor building dome beyond those already discovered and repaired by the Applicant.

The reactor building dome is the roof of the reactor building. It is constructed in the shape of a hemisphere with an inside radius of 70 feet. The concrete in the dome is 3 feet thick. Layers of reinforcing steel run both horizontally and vertically within the concrete near both the inside and outside surfaces. The inner side of the concrete dome is lined with a one-quarter inch carbon steel liner plate which assures leak-tightness of the building. The liner plate also served as a concrete form for the inside surface of the dome during concrete placement.

Concrete for the hemispherical reactor dome was poured in a series of "lifts" or layers placed sequentially starting from the vertical walls of the reactor building and working upward towards the top of the dome. Most of the concrete was placed without special problem; however, difficulties were encountered as construction approached the top. This portion of the dome was poured without the use of outer forms which would constrain the flow or movement of concrete after placement. The angle of the dome ranged from 45 degrees to near horizontal at the top. At these angles and without forms, the freshly placed concrete tended to slide downward as it was being consolidated by workmen using vibrating machines. The migrating concrete was moved to upper levels by workmen to restore the desired surface. Movement of concrete was a surface phenomenon because of the lack of external constraint at the surface. (Board Findings 94, 95)

When the concrete hardened, workmen noted surface honeycombing in four areas of the dome. Upon chipping these areas to sound concrete, it was found that

there was some loss of bond between concrete and steel reinforcing bars at depths of 4 to 6 inches from the surface. The defects consisted of small gaps ¼ to ½ inch in diameter on the lower side of the reinforcing bar. Later when some blockout forms were removed, three additional areas of honeycombing were found making a total of seven areas known to contain defects. (Board Findings 96, 99)

The Daniel Corporation, acting in its role as constructor for the Applicant, filed a nonconformance report for approval by Bechtel Corporation on November 10, 1980. NRC was not notified at this time since Daniel inspectors determined that the imperfections did not represent a significant deficiency in quality control, construction or engineering and were not reportable under the provisions of 10 CFR Part 50.55e. Bechtel and Union Electric personnel, however, questioned whether a more rigorous investigation of the significance and extent of the imperfections should be performed since there existed the possibility that additional areas of imperfections might be present. Union Electric then notified the NRC Staff on December 5, 1980 of a potentially significant deficiency. (Board Finding 98)

Union Electric and Bechtel decided to conduct tests of the dome using both destructive and nondestructive methods to determine whether there were other areas of concrete imperfections. Techniques used included nuclear densometer testing, boroscopic examination, microseismic (pulse echo) examination, selective excavation and engineering analysis. The nuclear densometer and boroscope examinations did not reveal further evidence of unsound concrete near the outer surface of the dome which is the only area in which they are effective. (Board Finding 100)

The pulse-echo technique was used at 1,671 locations as a means of searching for imperfections throughout the three-foot thickness of the dome. This technique works by generating a sound pulse in the surface of concrete which passes through the test section and reflects from the opposite surface back to a detector at the surface. The time of passage is measured with the aid of an oscilloscope. In normal concrete the oscilloscope display shows two pulses or peaks separated along a horizontal axis. The first pulse is from the initial signal and the second is from the reflected signal. Imperfections are detected because they reflect sound back to the detector from an interior location. This results in an additional pulse being displayed on the oscilloscope screen between the two normally present. The pulse-echo method has been used extensively for similar applications and is reliable and accurate. (Board Finding 101)

The results of the pulse-echo testing showed that of 1671 tests, 28 or 1.68 percent were of possible structural significance. The readings, however, were sporadic in occurrence. None of the areas tested showed a sufficient number of such readings to classify the area as structurally defective. (Board Finding 102)

The Applicant excavated concrete at six test points which had shown defects by the pulse-echo method. At two of these points no imperfections were found, while at four points small air gaps on the downhill side of the reinforcing bar were found.

The excavations confirm that a correlation exists between the pulse-echo readings and the actual imperfections. The nature of the imperfections serve to confirm the original conclusion that they were caused by subsidence. (Board Finding 103)

The results of testing and evaluation of the reactor building dome are contained in Bechtel's "Final Report of Containment Dome Concrete Imperfections at Callaway Unit 1" dated March, 1981. Staff review of that document concluded that more testing was needed. Additional pulse-echo tests were done and Applicant, and Staff concluded that the extent of imperfections was clearly identified. (Board Finding 104)

The effect of the concrete imperfections was to reduce the bonding between the reinforcement bar located near the outside dome surface and the concrete. (Ma Testimony, at 3) This kind of imperfection was not critical because (1) the imperfections are primarily limited to the hoop bars and do not affect the meridional reinforcement; (2) the type of load placed upon the hoop bar (an axisymmetric loading) is such that there is no change of stress along the bar and therefore sporadic imperfections would not affect their load carrying capabilities; and (3) the design margins for the hoop bars and radial ties are more than adequate to overcome any minor lack of bond due to the sporadic imperfections. (Applicant Dome Testimony, at 25)

An engineering analysis made by the Applicant showed that even if the bond between concrete and steel were lost for 50 percent of the entire reinforcing bar near the outer surface of the dome, the structure would retain sufficient strength to meet all design and accident load conditions. The actual loss of bond was minor and sporadic (consisting of ¼ to ½ inch voids along reinforcing bar near the outer surface at 28 locations). The design strength margins of the dome far exceeds the minor loss of strength that could be attributed to the small voids in concrete. (Board Finding 105)

The Staff concluded by independent analysis that the concrete had higher than required compressive strength and that the minor loss of bonding would not affect the structural integrity of the dome since ample margins of safety exist principally because of extra steel reinforcements. (Board Findings 106, 107)

No repairs of imperfections revealed by the pulse-echo investigation were necessary because of the design margins of safety in the structure. The areas which had previously been chipped to sound concrete, however, were repaired by filling with concrete of the same characteristics used for the dome construction after first preparing the cavities to ensure adequate bonding. (Board Finding 99)

Adequacy of Applicant's Quality Control

No evidence was brought out in the Board's questioning of witnesses which would call into question the quality control procedures used by the Applicant in

relation to inspection and discovery of defects, reporting, testing of the dome, or the repairs.

The evidence from the dome episode shows that the Applicant's actions demonstrated an affirmative commitment to quality of construction. The Applicant and Bechtel might have rested on an easy concurrence with the Daniel Corporation's initial assessment that the defects were minor. (This later proved correct but could not be known with certainty at the time.) The Applicant instead ordered additional testing which would ascertain the full extent of concrete imperfections in the dome. The testing itself was comprehensive in that it utilized several different methods and many individual measurements. As indicated, the pulse-echo testing alone was systematically performed at 1671 points. The defects that were revealed proved in final analysis to be structurally insignificant. (Board Finding 108)

The Staff also demonstrated a skeptical and analytical approach during the resolution of this problem. At one point it requested removal of additional forms in the dome which subsequently revealed more imperfections and at another required even further pulse-echo testing to satisfy itself that the full extent of imperfections were known. The Staff engineering analysis was independently performed rather than relying on review of the Applicant's analysis.

Conclusion

The Board concludes that the Applicant has adequately discharged its burden of proof on contention I.C.2. Contrary to the contention, the full extent of concrete imperfections in the reactor building dome is known from tests. Engineering analysis shows that the integrity of the structure is not in doubt because of the minor nature of the imperfections and the compensating design margins inherent in the structure. Additionally, there is no evidence related to the discovery or resolution of this matter which would suggest an overall failure or breakdown of the Applicant's quality assurance or quality control programs.

D. II.A.1. SA-358 Piping

SA-358 is an ASME material specification for a type of welded stainless steel pipe which is widely used for pipe sizes greater than eight inches in diameter. The pipe is made from plate by forming and rolling the plate into a continuous tubular shape. The resulting longitudinal seam is then welded, usually by the submerged-arc process, with the weld made from both the inside and outside surfaces. (Finding 109)

A Daniel pipefitter in the process of preliminary work on the pipe spool piece prior to fit-up for welding noticed an internal weld surface irregularity in the SA-358 pipe in question. He brought the matter to the attention of Daniel quality

control personnel, who observed the irregularity and possible ovality/thin wall conditions. The inspector had an ultrasonic test performed on the pipe which indicated that a thin wall did exist. A nonconformance report (NCR) was generated and a "hold tag" was placed on the pipe. (Finding 110)

The Material Specification for SA-358 piping allows an outside diameter variation of 1 percent. The pipe in question has a nominal outside diameter of 10.75 inches, not including the allowable weld reinforcement of 0.125 inches. Measurements taken at the request of the NRC Staff showed a maximum outside diameter variation of 0.092 inch or 0.86 percent, within the 1 percent limit. (Finding 111)

The pipe in question has a specified minimum wall thickness of 0.874 inch. An actual minimum thickness of 0.814 inch was found in the pipe's inservice inspection weld preparation area, which had been counterbored. Bechtel performed two calculations as provided in ASME Section III, Article NC-3640 to determine the acceptable minimum wall thickness for this piece of pipe. The calculations yielded acceptable minimum wall thicknesses of 0.711 and 0.795 inch, respectively. Independent calculations were performed which verified Bechtel's findings. (Finding 112) The Board agrees that the counterbored area did result in a wall thickness below the specified minimum thickness, but finds that the actual minimum thickness of 0.814 inch is adequate.

The third irregularity raised in the contention is that the pipe "had rejectable weld defects on the inside of a longitudinal seam weld." The hearing on this claim addressed irregularities, and their potential causes, of several types — including excess reinforcement, overlap, and fissures.

Daniel measured an area of weld reinforcement on the inside of the SA-358 pipe with a reinforcement height of 3/16 inch and documented it in a nonconformance report dated April 30, 1979. While Bechtel initially, erroneously dispositioned this NCR, it is clear that there was a nonconformance as to the reinforcement height, since 1/8 inch is the maximum permitted by SA-358. Apparently recognizing the error, Daniel elected to rework the item in accordance with its approved procedures, to bring the weld into compliance with the ASME code. The excess weld reinforcement was reworked by simple removal of the excess material by localized grinding. Joint Intervenors have alleged that this nonconformance was not repaired or reworked in accordance with documented procedures. However, Daniel could have first initiated a deficiency report for rework (rather than an NCR), and simply reworked the item. Instead, a more conservative approach was taken by first seeking the designer's review of the matter. (Board Finding 113) The Board finds that Daniel acted properly in assuming the responsibility to correct the nonconformance.

The Daniel NCR also identified a condition described as overlap in the same area as the excess weld reinforcement. The overlap apparently was excess weld material which had rolled over onto the surface of the pipe material. Bechtel

advised, in its disposition of the NCR, that overlap is not listed in the ASME code as a rejectable condition for radiography. This is because overlap does not affect the volumetric quality of the weld. Overlap is a condition that occurs at the intersection of the weld with the pipe material surface, but could not propagate through the thickness of the weld because it is in the wrong plane for propagation. Nevertheless, the overlap was reworked by the grinding process discussed above with respect to excess reinforcement. (Finding 114)

Counsel for Joint Intervenors extensively cross-examined the witnesses on the possibility that the weld defect might have been caused by "drop through" or "melt through." In addition, questions were asked on whether there might be cracks or fissures in the weld. (Board Finding 115)

Melt-through occurs in a submerged arc weld of the type used to weld SA-358 piping when total passage of both weld metal and the flux to the other side of the weld occurs. Drop-through is a similar condition, although less extensive. Both melt-through and drop-through are visible conditions. There are no reports that melt-through was visually noticed, and photographs of the weld in question reveal no evidence of drop-through. Not only would melt-through be visible, it would also be detectable on radiographs because of the resultant development of porosity in the weld. Radiographs of the weld reveal no such defect. In addition, testimony indicated that the presence of overlap indicates drop-through could not have occurred. (Findings 116, 117) The Board concludes that no drop-through or melt-through occurred in this weld.

In NRC's investigation of this piece of pipe, an allegation was addressed that the weld was cracked. Photographs of the weld exhibit two fissures which could have been mistaken for a crack. The indications identified as fissures were in the excess material and not in the weld itself. There was testimony that the indications described as fissures were actually the result of overlap where the excess weld metal came out on the surface of the pipe without wetting the pipe. If fissures existed in the weld, they would have been visible in radiographs. Radiographs reveal the weld to be free from defect. (Finding 118) The Board, therefore, finds that there were no cracks or fissures in the weld of the pipe.

The excess weld material was removed from the pipe by grinding, although the pipe would have been able to perform its function without this removal. The Staff reviewed both the pipe itself and radiographs of the pipe taken subsequent to grinding. The Staff found that the pipe in its present condition is free from defects. The Board agrees.

Conclusion

The Board finds that the record developed with respect to SA-358 piping does not reveal a breakdown in Applicant's quality assurance program. The weld defect in question, a relatively minor one, was discovered by Daniel personnel and

dispositioned by Daniel. The weld defect was removed, and the pipe was shown to be adequate with respect to ovality and wall thickness.

E. II.A.2. SA-312 Piping

SA-312 is an ASME material specification for both seamless and welded stainless steel pipe. Welded SA-312 pipe is made from plate by forming and rolling the plate into a tubular shape. The longitudinal seam is then autogenously welded (without filler metal) by the gas tungsten arc method. The weld is made from both the inside and outside surfaces for double-welded pipe. (Finding 119)

Those safety-related systems which contain double-welded SA-312 pipe are designated as ASME Classes 2 and 3 (seamless pipe only was used for systems designated as ASME Class 1). Under the rules of the ASME Code, welded piping is required to meet all the tests and examinations prescribed by Section III. The material specification for SA-312 requires chemical analysis, tension tests and flattening tests to be performed on each lot of pipe. The material specification also requires each length of pipe to be hydrostatically tested. ASME Section III requires that welded pipe for use in Class 2 systems be nondestructively examined by one of the following methods: ultrasonic, eddy current, magnetic particle, liquid penetrant, or radiographic examination. It is usual for pipe manufacturers to select the ultrasonic method for SA-312 pipe as other methods are not suitable for large diameter pipe. (Finding 120)

The problem addressed in Subcontention II.A.2 with SA-312 pipe is centerline lack-of-penetration (CLP). CLP occurs in autogenously double-welded SA-312 pipe when complete through-wall fusion does not occur between the inside and outside welds during welding of the longitudinal seam. A plane then exists in the center of the pipe wall between the two weld passes where the original plate edges are tightly abutted but not fused. (Finding 121)

This problem with SA-312 piping is not limited to the Callaway facility, but is generic in nature. On September 27, 1978, the Arizona Public Service Company informed the NRC that Pullman Power Products (PPP), a fabricator of safety-related piping spools for use in the Palo Verde Nuclear Generating Station, discovered longitudinal weld defects in ASME SA-312 type 304 austenitic stainless steel pipe supplied to Pullman by the Youngstown Welding and Engineering Company (YWEC). The defects were identified during the radiographic examination of circumferential shop assembly welds. Forty-four percent of the completed and partly fabricated subassemblies were rejected, the majority because of CLP. Less than two months later, Southern California Edison reported similar defects in pipe supplied for one of its nuclear facilities. Documentation provided with the pipe indicated that YWEC had performed the required ultrasonic examination, but the rejectable indications had not been identified. (Finding 122)

This determination resulted in a special investigation at YWEC by NRC inspectors. They determined that the apparent cause of the identified defects was nadequate control of welding parameters, including welding current, voltage and ravel speed. The NRC Staff then issued Bulletin 70-03. (Exhibit XI to Staff Exhibit 7). This Bulletin required that licensees: (i) determine whether double-velded SA-312 pipe manufactured by YWEC had been incorporated or would be ncorporated into safety-related piping systems, (ii) identify the system, location, ipe size and pressure/temperature parameters where the double-welded SA-312 ipe was or would be used, and (iii) develop a program for the volumetric xamination of the longitudinal welds and provide suitable corrective action for on-conforming material. (Finding 123)

With this discovery of a potential problem with SA-312 pipe in a number of uclear plants where it was involved, Bechtel determined that a detailed test rogram should be initiated to look into this generic problem. The test program as designed both to assess the ability of ultrasonic examination to detect CLP and assess the effects of CLP on various mechanical properties of double-welded A-312 pipe. The results and conclusions of Bechtel's investigation are contained Bechtel's "Report on Investigation of Weld Imperfections in ASME SA-312 ouble-Welded Austenitic Stainless Steel Pipe for Compliance with NRC I&E ulletin 79-03" (Applicant Ex. 11). (Finding 124)

During this investigation, it was determined that the principal cause of CLP was e wide range of allowable welding parameters permitted by the YWEC qualified elding procedure. The significant welding parameters include arc voltage, mperage, travel speed, and weld head oscillation. Differences in the allowable ttings for these parameters affect the depth of penetration of the upper and lower eld passes. Thus, as amperage is decreased to the minimum allowable setting nder the YWEC qualified welding procedure, less heat is transmitted to the weld rface and the weld is relatively shallower. Similarly, an increase in arc travel eed will result in a shallower weld as will an increase in the weld head oscillation te. The effect of any combination of settings depends on the thickness of pipe ing welded. The range of parameters in the YWEC qualified welding procedure as approved for a thickness range of 1/16 inch to ⅝ inch. Therefore, settings hich would produce acceptable penetration for a 1/16 inch pipe might result in me CLP if used for a thicker walled pipe. An additional factor which can ntribute to CLP is arc misalignment in which the upper and lower weld arcs are t aligned with the longitudinal seam and with each other. As a result, the weld netrations may be sufficient, but the upper and lower weld beads will not meet in e center of the pipe thickness because the point of deepest penetration in the top eld is not aligned with the deepest penetration point in the lower weld. (Board nding 125)

The Bechtel investigation concluded that the ASME Code-required ultrasonic amination cannot reliably detect CLP in double-welded SA-312 pipe. Bechtel's

investigation included review of the ultrasonic testing techniques used by PPP and Ultralabs, and testing of four special test weldments fabricated by Bechtel with intentionally produced CLP, varying in amount from 35 percent to 60 percent. The Code-mandated ultrasonic examination was not able to detect the CLP in the four samples. Bechtel concluded that the two unfused base metal edges of the rolled plate are in such intimate contact that the ultrasonic sound waves are transmitted without interruption across the unfused area and are not reflected back to the ultrasonic transducer and displayed as an indication. Furthermore, the geometry of the CLP is such that even if the ultrasonic sound wave is reflected from a CLP condition, the majority of the energy would not be returned to the transducer and displayed as an indication. Accordingly, Bechtel concluded that the Code-specified ultrasonic examination will not reliably detect the presence of CLP in SA-312 piping. (Finding 126)

As part of its investigation, Bechtel determined the maximum amount of CLP in the SA-312 piping produced by YWEC. Bechtel examined 71 cross-sections of longitudinal welds in over 500 feet of double-welded SA-312 pipe supplied by YWEC to PPP. Of the specimens, 25 showed some degree of CLP. The greatest amount of CLP was 26 percent of the wall thickness of the pipe. There is ample evidence that the extent of CLP that may exist in Callaway SA-312 piping will be no greater than that examined in the Bechtel generic investigation. The Callaway pipe was fabricated by the same process, same machines, same personnel and within the same time period as the pipe supplied to PPP and examined by Bechtel. Furthermore, in intentionally producing test samples with greater than 26 percent CLP, Bechtel was required to use welding parameters outside the range of parameters used by YWEC. (Finding 127)

Bechtel performed tests to determine the effect of CLP on the mechanical properties of the pipe, but, in order to obtain data on welds which contained more than 26 percent CLP, the testing was done on a series of welded plates which were prepared so as to simulate welds in production pipe containing CLP. The same material type was used as the base material from which the SA-312 welded pipe was made. The intentionally produced CLP in the test plates ranged from 11 percent to 47 percent. The yield strength, ultimate tensile strength and elongation were measured, and it was shown that even with 25 percent CLP, SA-312 piping will meet all the ASME mechanical property requirements. Bechtel also had three hydrostatic burst tests performed. The first test was performed on a piece of YWEC pipe containing 15 percent centerline lack of penetration. The other two tests were performed on specially welded pipes with intentionally fabricated CLP of 40 percent and 55 percent. The pipes were plugged at each end and hydrostatically pressurized until fracture occurred. Normal hydrostatic test pressure for this size pipe and schedule is calculated to be 882 psi. The pipe with 55 percent CLP burst at the lowest pressure between 3000-3100 psi. (Finding 128)

In addition to the Bechtel test program, two engineering analyses of SA-312 pipe with CLP were performed by Aptech Engineering Services, Inc. (Aptech). These included a fracture analysis study and a subsequent fatigue analysis. The results of these studies are contained in two Aptech reports introduced into evidence as Applicant Exhibits 12 and 13. The fracture analysis demonstrated that because of the very ductile nature of the stainless steel material used in SA-312 piping, the failure mode of the pipe would not be brittle fracture, but rather, a "leak-before-break" and ductile fracture mode. A limit load analysis was therefore used to calculate critical flaw sizes for a range of pipe stress conditions, pipe diameters and wall thicknesses. These calculations were conservatively confirmed by the actual results of the Bechtel burst tests. Using these results from the Aptech fracture analysis and assuming the highest hoop stress values in piping systems at Callaway containing double-welded SA-312 pipe, it was concluded that the CLP condition of the magnitude identified will not result in the initiation of a leak in such piping and that the possible presence of CLP is not a concern. Testimony at the hearing established that under the design conditions at Callaway, CLP on the order of 85 percent of wall thickness would have to exist before a pipe would leak. Even assuming initiation of a leak, the fracture analysis demonstrated that the critical CLP size (amount of CLP above which catastrophic failure will occur) is greater than the wall thickness of the pipe and thus catastrophic failure cannot occur. (Board Finding 129)

Aptech considered CLP more important from a fatigue point of view than from a fracture point of view because of the SA-312, Type 304 material's ductile behavior. Aptech's fatigue analysis was based on linear elastic fracture principles which link together flaw size, fatigue crack growth rate and applied stresses. The thrust of the analysis was to establish acceptance criteria based on worst-case assumptions. The result of the analysis was a series of flaw size versus life curves for a range of cyclic stresses so that the effect of any amount of CLP in any piping system could be assessed. When the results of Aptech's analysis were compared with actual conditions at Callaway, Aptech determined that the actual combined worst case parameters at Callaway are well below the assumed worst case conditions used during the fatigue analysis. (Finding 130)

In summary, the testing and analyses performed during this generic investigation of the CLP problems established that double-welded SA-312 piping, even with amounts of CLP substantially in excess of that found on production pipe, will function as intended with an adequate margin of safety. However, since it was also established that the ASME Code-required ultrasonic examination was ineffective in detecting the presence of CLP, Bechtel, in its report (Applicant Ex. 11), recommended a two-tiered response to the CLP problem in which the level of further examination for SA-312 piping would depend upon the hoop stresses in the system in which such piping was to be used. (Finding 131)

In Bulletin 79-03A (Exhibit XII to Staff Exhibit 7), the NRC had indicated that 85 percent of ASME allowable code stresses was an appropriate screening mechanism for the use of SA-312 pipe. For systems subject to design stresses less than this level, the NRC found that a satisfactory design margin exists. This conclusion was supported by the Aptech fracture analysis, the Bechtel burst tests, the worst case of CLP actually found, and the level of CLP that can be expected to be detected by nondestructive examination. (Finding 132)

Double-welded SA-312 pipe is used in Callaway in the following systems: residual heat removal system, accumulator injection system, fuel pool cooling system, and the refueling water storage tank. Bechtel performed a series of calculations in responding to Bulletin 79-03A to determine the maximum hoop stresses in any piping systems containing double-welded SA-312 pipe. All of the affected piping systems had hoop stresses less than 85 percent of the ASME allowable.

Based on all the evidence, the Board finds that the designed use of SA-312 pipe at Callaway does not affect the safe operation of the plant.

Intervenor also raised questions about the safety of SA-403 fittings. SA-403 is a specification for wrought austenitic stainless steel pipe fittings, such as elbows, tees and reducers. SA-312 is frequently used as the raw material for such fittings. SA-403 fittings made from double-welded SA-312 pipe would contain CLP to the same extent as the straight-run pipe. (Board Finding 133)

The record reveals that no SA-403 fittings at Callaway made from double-welded SA-312 pipe are included in piping systems which have hoop stresses in excess of 85 percent of the stresses allowed by the ASME code. While fittings theoretically may be subject to different stresses than straight legs of piping, the nature of the systems using SA-403 fittings at Callaway is such that no separate analysis need be made. (Board Finding 134)

The NRC Staff has also concluded, based on the analytical and experimental effort described above, that failure of double-welded SA-312 piping due to the possible presence of CLP is highly improbable. Rutherford Testimony at 4-7.

Conclusion

The Board, therefore, finds substantial evidence in the record to conclude that the double-welded SA-312 piping installed at Callaway is structurally sound and can safely perform its design function. Even if it is hypothetically postulated that a system containing that piping became overpressurized to the point that failure occurred in the longitudinal seam weld, the failure would occur in the "leak-before-break" mode. Under no circumstances would a brittle fracture resulting in a catastrophic failure occur. Rather, a small leak would form in the longitudinal weld and stable propagation would occur only if the internal pressure were maintained or increased. This is unlikely to occur because of the instrumentation

and control systems which would provide plant operators with appropriate information whenever conditions in a safety-related system exceed design conditions, so that appropriate action can be taken.

Contrary to the general allegation of Joint Intervenors, the incorporation into the safety-related systems at Callaway of SA-312 piping which may contain CLP cannot be considered a breakdown in the quality assurance/quality control programs in effect at Callaway. The CLP problem was generic in nature. While its cause may have been inadequate process control by the piping vendor, the means prescribed by the ASME Code to detect this imperfection were later determined to be inadequate.

Furthermore, the Intervenor's contention that the evaluation and acceptance of SA-312 piping with CLP were not performed according to the requirements of the ASME Code has also been shown to be without merit. Apart from the required destructive testing (hydrostatic, tension and flattening) which was performed, all welded pipe which may contain CLP underwent ultrasonic examination and met the ASME criteria for this examination. It was the examinatinn procedure itself which was found to be deficient. It has been established that the use of the efficiency factors in the ASME Code provides a conservative and satisfactory alternative to the ultrasonic examination. Accordingly, all SA-312 piping at Callaway complies with the ASME Code requirements. More significantly, however, the exhaustive investigation of the nature and extent of CLP in double-welded SA-312 pipe, including tensile tests and hydrostatic tests, along with the Aptech fracture and fatigue analyses that were performed, demonstrates that this type of pipe meets the design and service conditions specified in the ASME Code, and will safely and properly perform its intended function throughout the life of the Callaway plant.

F. II.B. Piping Subassembly Deficiencies

As indicated in the contention (Appendix II), Gulf & Western ("G&W") supplied preassembled piping formations for use at Callaway. Preassembled pipe formations are pre-designed, manufacturer-fabricated formations containing piping, fittings, valves, pumps, strainers, tanks and other similar equipment. The formations are discrete portions of piping systems that are completely assembled at the manufacturer's plant, delivered to the construction site and set in place as a unit, rather than being fabricated piece-by-piece at the construction site. The preassembled pipe formations serve the same purpose as all other piping systems in the plant, *i.e.*, to transfer fluids, and are designed and manufactured to minimize the amount of onsite craft labor, thereby realizing cost and schedule efficiencies. (Board Finding 135)

In accordance with ASME Code requirements and Bechtel specifications, nondestructive examinations ("NDE") were performed on welds by G&W in order to detect any conditions not in conformance with ASME Code criteria. Liquid penetrant examinations were performed on all welds in all Class 3 formations, and radiographic examinations were performed on all welds in all Class 2 formations. Pursuant to Bechtel's procedures and specifications, in-process and final surveillance inspections were conducted by a Bechtel supplier quality representative at the G&W facility. The Bechtel representative was to have inspected numerous stages of the fabrication and post-fabrication review and testing of the formations. Upon delivery at Callaway, Daniel personnel performed a receipt inspection of the formations to check for proper paperwork and shipping damage, but did not normally inspect the quality of the welds in the formations. (Board Finding 136)

Potential deficiencies in the G&W formations were first detected by Daniel construction (welding) personnel at the site of Kansas Gas & Electric Company's Wolf Creek plant, which is another SNUPPS unit, in March 1979. The discrepancies were identified while the formation was being installed and were brought to the attention of a Daniel welding inspector who performed a visual examination of the formation and identified possible concerns with respect to both the quality of the formation welds and the quality of the radiographic examination techniques utilized by G&W. This information was passed along to Union Electric to determine if the potential deficiencies applied to Callaway. (Board Finding 137)

Applicant and Daniel personnel proceeded with an audit of G&W formations at Callaway, including a physical review of the formations and welds themselves and a review of G&W radiographs. The results of the audit indicated that there were noncompliances with Bechtel specifications and ASME requirements in the areas of both radiographic technique and weld discrepancies. A SNUPPS audit was then performed at the G&W facility. At the close of the audit, G&W conducted a 100 percent review of the weld radiographs. This review revealed radiographic technique deficiencies which prohibited a definitive determination as to the extent and significance of defects in the welds. (Board Finding 138)

In order to resolve the deficiencies, G&W agreed to review and radiograph all welds, and rework the welds as necessary. Three formations were returned for rework to G&W's manufacturing facility; most formations were to be reworked by G&W personnel at the Callaway site. G&W's performance in these endeavors was monitored by Union Electric, Daniel and Bechtel inspection personnel. The monitoring indicated continuing unsatisfactory performance by G&W. (Board Finding 139)

Following the discovery that G&W's rework effort was unacceptable, Applicant directed G&W to cease its rework efforts and turned the matter over to Daniel. Daniel performed visual inspections of the welds; when necessary, ground the weld surfaces in order to meet NDE requirements and visual acceptance standards; performed the required radiographic work; and rewelded welds that were found to

be rejectable. All repaired welds were then given the appropriate nondestructive examination to ensure that the new welds met applicable acceptance criteria. The NRC Staff has reviewed the condition of the welds after Daniel finished its rework. The Staff found the quality of the repair work to be acceptable and the formations to be now adequate for use at the facility. (Board Finding 140)

Conclusion

The Board finds that the work performed by Daniel and the Staff's review of the pipe and welds in their present condition provide adequate assurance that the preassembled piping formations in their present condition will not affect the safe operation of the plant.

G. Quality Assurance Contention

Joint Intervenors established prior to hearing that their individual contentions were intended to be considered together as a whole to show that a breakdown of the Applicant's quality assurance program occurred during construction at Callaway. The evidence developed on the contentions showed that there were deficiencies in certain elements of the Applicant's QA/QC program. (Board Finding 141)

The Board has considered whether these deficiencies indicate a programmatic breakdown in Quality Assurance and concludes that they do not. We base this conclusion on the fact that (1) an extensive QA/QC program exists, (2) the deficiencies found were disclosed and remedied within the program itself, (3) the reactor building is safely built and (4) the Applicant displayed a generally affirmative commitment to quality in the discovery and resolution of the problems we considered. (Board Findings 142-146)

While we have expressed concern over some of the deficiencies found, we conclude that they are of limited extent and have no broader implications regarding the overall effectiveness of the Applicant's QA/QC program.

III. FINDINGS OF FACT

A. Contention I.A. Embedded Plates

1. Embedded steel plates, to support piping, electrical conduits, cable trays, HVAC components and structural steel framing are utilized in the Applicant's

Callaway plant. (Applicant Embed Testimony, at 11, 28, 34; Staff Embed Testimony, at 2)*

2. The plates are attached to the surface of concrete walls by means of welded steel studs and steel anchor rods positioned in the concrete. (*Ibid.* at 2; Applicant Embed Testimony, at 10-11)

3. Two different types of plates are used in the plant: machine welded plates with studs that are welded by an automatic process, and manually welded plates with anchor rods that are welded manually. (Applicant Embed Testimony, at 12; Staff Embed Testimony, at 3)

4. The facility's architect/engineer, Bechtel Corporation, has responsibility for embed plate design, load capacities and quality surveillance during fabrication; the plate manufacturer, Cives Steel Company, has responsibility for quality inspection of all embed plates, and until July 1977, Daniel International, the facility's construction company, had responsibility for receipt inspection limited to quantities of plates received and shipping damage. In July of that year, Daniel was directed to broaden its duties to include inspecting all safety-related items received at the plant. (Applicant Embed Testimony, at 13-14; Schnell, Tr. 663-666)

5. On June 9, 1977, an NRC Inspector identified machine welded plates at the plant site which lacked full 360 degree weld (flash) material that had not been bend tested to 15 degrees as required by the applicable code. (Applicant Embed Testimony, at 14-15; Staff Embed Testimony, at 3)

6. Prior to June 9, 1977, there had been 255 machine welded plates and 225 manually welded plates installed in safety-related buildings to support safety-related loads in the facility. (Applicant Embed Testimony, at 28, 34; Staff Embed Testimony, at 3)

7. On June 9, 1977, the Daniel Corporation issued stop work orders on installing additional plates. A reinspection was authorized by the Applicant of all machine welded and manually welded plates at the manufacturer's plant and at the Callaway site. (Schnell, Tr. 663; Meyers, Tr. 1227) An earlier reinspection (November-December 1976) provided no indication of defective materials being supplied by the Cives Company. (Intervenor Ex. 18, 19; Starr, Tr. 1451-1452)

8. The Cives Company and Daniel Corporation were separately directed to reinspect the welds on machine welded plates not installed and to bend test any studs to 15 degrees where a visual inspection showed the stud to have less than a 360 degree weld fillet. (Applicant Embed Testimony, at 18-19, 32-33; Staff Embed Testimony, at 4; Schnell, Tr. 661-662)

9. The Cives reinspection demonstrated a failure rate in machine welded studs of 0.08 percent or 66 studs in the 81,673 examined. The Daniel reinspection,

* Applicant Embed Plate Testimony, ff. Tr. 501, hereafter cited as Applicant Embed Testimony; Staff Embed Plate Testimony, ff. Tr. 1251, hereafter cited as Staff Embed Testimony.

which was conducted over a longer period of time, showed a failure of 0.11 percent or 106 studs out of 96,472 studs examined. (Applicant Embed Testimony, at 18-19; Staff Embed Testimony, at 4; Schnell, Tr. 1239-1240)

10. The 15 degree bend test was a more rigorous inspection requirement than that called for by the American Welding Society (AWS) Structural Welding Code, D1.1-75. (Applicant Embed Testimony, at 14-17)

11. The bend tests impose higher deformations and stresses into the studs and welds than the design loads applied to the plates. (Applicant Embed Testimony, at 19)

12. The rate of failure in the machine welded studs reinspected compares favorably with normal industry standards. (Applicant Embed Testimony, at 20; *also see* Staff Embed Testimony, at 4, and Applicant Ex. 4, Appendix A)

13. The reinspected plates were fabricated by the same company in the same time period using the same procedures as the plates which were installed in the facility before June 9, 1977. (Applicant Embed Testimony, at 20-21)

14. Using the failure rate data from the Cives reinspection, the Bechtel Corporation performed an engineering analysis that shows a probability of failure on the order of 1×10^{-9} for a machine welded plate installed prior to June 9, 1977. (Applicant Embed Testimony, at 21-26; Applicant Ex. 4, pp. 2-4)

15. Although the fabrication of a majority of the defective weld studs (59 percent or 39 of 66) on machine welded plates took place during a five-month period in 1976, the plates, on receipt at the plant, were intermingled with other plates and were generally interchangeable. (Thomas, Tr. 1218-19; Applicant Ex. 4, Appendix A)

16. The Staff did not rely on the Applicant's engineering analysis to base its conclusion that plates embedded prior to June 9, 1977 did not jeopardize the safety of the Callaway facility. (Gallagher, Tr. 1327-28)

17. The Staff requested the Applicant to test some machine welded embedded plates installed prior to June 7, 1977 as further evidence that the plates did not represent a safety problem. (Staff Embed Testimony, at 4)

18. Drs. Fisher and Slutter of Lehigh University performed tension tests on six machine welded embedded plates with loads in excess of design capacities without signs of plate failure. Plate selection was approved by the NRC. (Applicant Embed Testimony, at 27-28; Staff Embed Testimony at 4-5; Staff Ex. 6, p. 6)

19. The Staff witnessed the tests performed on the machine welded embeds. (Staff Testimony, Tr. 1418-19; Staff Ex. 6, Attachment E, p. 1)

20. The plates chosen for the embedded plate test were selected randomly on the basis of their accessibility and the feasibility of mounting a test rig on them. (Applicant Embed Testimony, at 27)

21. The precise location within the Callaway plant of machine welded plates installed before June 9, 1977 and the loads they carry are known. Machine welded plates have been designed to provide load capacities with a minimum safety factor

of at least 2.0 against the limit state of the plates or studs. These embeds do not support main floor beams and for critical piping systems are designed with a significant safety factor against failure. (Applicant Embed Testimony, at 28-31)

22. The requirements of the American Welding Society (AWS) *Structural Welding Code* D1.1-75 were applicable to welded studs and anchor rods on all embed plates in the construction of the Callaway facility. (Applicant Embed Testimony, at 13, 18)

23. The Cives Company was directed to reinspect all manually welded embeds not as yet installed at the Callaway site. The precise location of manual welded embeds and the loads they carry are known. (Applicant Embed Testimony, at 32-34)

24. Manual welding is required for anchor rods on embed plates as the rods are too large for automatic welding operation. (Applicant Embed Testimony, at 22)

25. Welding details, required by D1.1-75, are extremely difficult to accomplish by the manual welding process as a welder's orientation and access are hampered by the use of multiple studs. The pertinent sections of D1.1-75 were developed for linear welds and not for the kind of welds involved in the controversy here. (Applicant Embed Testimony, at 35-36; Applicant Ex. 4, p. 1)

26. Manually welded plates at Callaway are used to support structural steel framing members. (Applicant Embed Testimony, at 34)

27. Manually welded plates are designed for loads with a minimum safety factor of 2.0 against the yield limit state of the plate and tensile capacity of the anchor rods. (Applicant Embed Testimony, at 34; Meyers, Tr. 772-777)

28. The Bechtel Corporation was advised early in the reinspection effort that a number of welds did not meet the requirements of D1.1-75 on weld (leg) size, length of weld sizes, weld undercutting and weld profile (convexity). (Applicant Embed Testimony, at 32-35; Applicant Ex. 4, p. 1)

29. Based on "worst case" welding defects, the Bechtel Corporation performed an engineering analysis to evaluate the safety of manually welded plates installed prior to June 9, 1977. (Applicant Embed Testimony, at 37; Staff Ex. 6, p. 8)

30. Relying on reinspection information from Cives that the worst weld undersize was ⅛ inch, two legs of the weld were of unequal length, the welds exhibited excessive convexity, and the maximum undercut was 1/16th inch, Bechtel calculated reduced design capacities for every installed manually welded plate using the assumptions that all anchor rods were considered to have ⅛ inch undersized welds for a 360 degree perimeter of the anchor rod, both legs were considered to be undersized and all rods were to have a 1/16th undercut. (Applicant Embed Testimony, at 35-38; Meyers, Tr. 792; Applicant Ex. 4, p. 2)

31. In comparing the calculated reduced design capacity with the actual applied load on each embedded plate, Bechtel found the reduced capacity still exceeded the design load with the minimum safety factor against exceeding the

atic limit state of the plate of not less than 1.92. On four plates, however, the ign load and the reduced design capacity were the same. (Applicant Embed timony, at 37-38; Intervenor Ex. 78)

32. The Bechtel engineering analysis was found acceptable by the Staff and Applicant's expert consultants, Drs. Fisher and Slutter. (Applicant Embed timony, at 39; Staff Ex. 6, pp. 7-8)

33. The Bechtel engineering analysis was not based on final reinspection data m the Cives Company since the analysis was completed prior to the conclusion Cives reinspection reports. (Intervenor Ex. 22; Applicant Ex. 4; Meyers, Tr. 1. *Also see* Applicant Reply to Proposed Findings, No. 25.)

34. Subsequent to June 9, 1977, the Applicant directed the Daniel Corpora- n to inspect all safety-related plates at the Callaway site and those to be delivered Cives in the future. (Applicant Embed Testimony, at 40-41; Applicant Ex. 6, 1)

35. After Bechtel's final report on August 1977 concluding that welded studs Callaway were a completely acceptable product, the Daniel Corporation re- rted inspection results contradicting the assumptions used in Bechtel's engineer- analysis. These assumptions were based on Cives reinspection information. pplicant Embed Testimony, at 41; Applicant Ex. 6, pp. 1-2)

36. The Applicant and Bechtel claim to have had no knowledge of the results he Daniel inspection, which began in June 1977, until November 7, 1977. olland, Tr. 1384; Applicant Ex. 7, p. 4)

37. However, the evidence shows Daniel reporting a large number of defects nanually welded plates to the Applicant in August 1977. (Intervenor Ex. 39) testimony also reflects that Daniel and Cives personnel were aware of each er's inspection efforts. (Starr, Tr. 1359-60)

38. Subsequent to the receipt of the Daniel Corporation's inspection reports an estigation commenced, involving the Applicant, the Bechtel Corporation and niel, in an effort to resolve the differences between the Daniel and Cives data on nually welded plates. (Applicant Embed Testimony, at 41; Applicant Ex. 6, p. Applicant Ex. 7)

39. After a series of meetings, the Bechtel Corporation stated its incapability analyzing the Daniel inspection data on manually welded plates due to poor umentation, inconsistencies in reporting and possible errors in the data. (Ap- ant Embed Testimony, at 42; Applicant Ex. 7, pp. 1-4)

40. The reason assigned by the Applicant and Daniel for the lack of complete information was that Daniel was only required to record sufficient information nable the welded plate to be accepted or rejected, and not to provide a complete ure of the amount and extent of welding deficiencies. (Applicant Embed timony, at 43; Starr, Tr. 1357-58; Hofland, Tr. 1380-84)

41. Bechtel completed its review by concluding its previous analysis support- the acceptability of manually welded embeds was not contradicted by the

Daniel's data. This decision was based on the following: first, that after eliminat- ing inconsistent and incomplete data from the Daniel reports, only eight (8) embeds, which have since been repaired, out of 532 reported showed an average weld undersize greater than ⅛ inch; and second, that reinspections of 47 unre- paired but rejected embeds still at the site showed different results from the original Daniel inspections, with the average weld undersize not in excess of ⅛ inch. (Applicant Embed Testimony, at 43-44; Applicant Ex. 7, pp. 2-4; *also see* Applicant Ex. 6, pp. 3-5; Staff Ex. 6, p. 8)

42. The Applicant's review of Daniel's final effort at revising its inspection reports of manually welded embeds found that ten (10) embeds out of 364 reported had an a age weld undersize exceeding ⅛ inch. This number was calculated on the assumption, however, that the undersize indicated extended around the cir- cumference of the stud. (Applicant Ex. 6, p. 4)

43. In its submittal of its final report, Daniel's project manager stated that, since Daniel's inspectors only recorded the greatest, and not the average, under- size, any assumption that the maximum undersize condition went around the complete weld circumference would not represent a true image of the actual conditions. (Intervenor Ex. 14, p. 2)

44. Applicant's expert consultant, Dr. Fisher, testifie , that the manually welded embeds could safely carry their design loads even assuming that the worst weld deficiencies in the Daniel data extended around the circumference of the anchor rods. Dr. Fisher stated these weldments could have been 25 percent smaller and that existing welding codes were being changed to accommodate that addition- al margin. (Fisher, Tr. 742-45, 1136)

45. Based on its investigation and the difficulty of meeting code welding requirements for circular manual stud welding, Bechtel sought and received approval for exceptions to AWS D1-1 for welding between anc ds and plates. The exceptions provided for smaller vertical leg welds, unequal legs, elimination of profile requirements and a 1/16 inch undercut for up to 10 percent of the weld length. The Staff endorsed these exceptions as minor in nature and as not affecting the capacity of the connection. (Applicant Embed Testimony, at 39-40; Applicant Ex. 4, Appendix C; Staff Embed Testimony, at 5)

46. Three years after manually welded embeds had been installed, the Staff (NRC) visually inspected plates substantially loaded with floor slab dead2 ds and reported no signs of distress. (Staff Ex. 6, p. 5)

47. On inspecting the rejected but unrepaired manually welded embeds at the Callaway site, the NRC Staff requested Applicant to perform load tests on selected welds, which appeared to have poor workmanship, in order to test their structural integrity. (Staff Ex. 6, p. 9; Applicant Embed Testimony, at 45)

48. Bend tests to 30 degrees on six anchor rods on six different plates and tension tests on an additional six anchor rods from six other plates were conducted at Lehigh University by Drs. Fisher and Slutter. The bend test plates were selected

by NRC Staff as well as the direction of the bend and the tension test plates were selected by the Applicant and reviewed by the Staff. Both tests were witnessed by the NRC. (Applicant Embed Testimony, at 45; Staff Ex. 6, p. 9 and Attachment E, p. 1; Applicant Ex. 5, p. 1)

49. The bend tests were conducted without any signs of cracking or weld failure and the tension tests, which tested the specimens to failure, demonstrated a minimum ultimate weld strength of 46,200 pounds for welds with a design load strength of 13,650 pounds. The remaining five welds tested showed an ultimate strength of 56,000 pounds or over. (Applicant Embed Testimony, at 46; Applicant Ex. 5, pp. 2-4; Staff Ex. 6, p. 9)

50. The Applicant and the NRC Staff both conclude that based on reinspection of plates fabricated in the same time frame, the engineering analysis of reduced load capacities due to weld deficiencies, and the actual load tests that were performed, the plates installed in the Callaway facility prior to June 9, 1977 were capable of safely supporting their design loads. (Applicant Embed Testimony, at 48; Staff Embed Testimony, at 5; Staff Ex. 6, pp. 9-10)

51. The NRC Regional Office which inspected and investigated events at the Callaway facility referred to NRC Headquarters — for review and a determination of adequacy — the Applicant's report of March 10, 1978, that approved the embeds installed prior to June 9, 1977 as acceptable in meeting design load requirements. The question of adequacy of embeds had been brought to Headquarters' attention at even an earlier date. After NRC Headquarters failed to act, the Regional Office again assumed responsibility for technical review of the report. (Intervenor Ex. 34, p. 4-5; Gallagher, Tr. 1298-1299)

B. Contention I.C.1. Honeycombing in the Reactor Building Base Mat

52. Concrete for the reactor building base mat was placed over a 62 hour period from April 6 to April 9, 1977. One hundred and ninety (190) construction crafts, engineering, quality control and supervisory personnel, were used in shifts to accomplish the concrete placement. NRC Staff inspectors were present and observed most of the operation (Applicant Base Mat Testimony, at 11-12; Staff Base Mat Testimony, at 3, 6-7)[5]

53. Honeycombing in the concrete ceiling of the tendon access gallery was found by construction personnel after the concrete had hardened. Honeycombing is a defective condition that consists of small air pockets in hardened concrete giving a "popcorn" appearance. (Applicant Base Mat Testimony, at 16, 17; Varela, Tr. 401, 402)

[5] Applicant Base Mat Testimony, at Tr. 227 hereafter cited as Applicant Base Mat Testimony. Staff Testimony of A. Varela, ff. Tr. 396 hereafter cited as Staff Base Mat Testimony.

54. All honeycombed areas were chipped to sound concrete, to determine the extent of the imperfections and to prepare the surface for repair. In all, 19 areas of honeycombing which required 24 separate excavations were found in close association with trumplates in the tendon access gallery ceiling. (Applicant Base Mat Testimony, at 15; Staff Base Mat Testimony, at 3)

55. The honeycombing was structurally significant because it called into question the performance of 14 of the 172 trumplates. (McFarland, Tr. 256; Applicant Base Mat Testimony, at 15)

56. The size of the honeycomb areas ranged from less than 1 square foot to a maximum of approximately 22 square feet. Most of the excavations were less than 4 square feet in area. The depth of the individual excavations averaged approximately 10 inches with a localized maximum of 17 inches. Lower layers of reinforcing bar embedded in the concrete were exposed. (Applicant Base Mat Testimony, at 15, 16; Staff Base Mat Testimony, at 3, 4; Staff Ex. 3, at 21, 22)

57. The cause of the honeycombing in the base mat was inadequate consolidation of the concrete during the placement operation. Consolidation is achieved by workmen using handheld vibrating tools which are inserted into the still wet concrete. The vibration causes the concrete to liquify temporarily and to flow around the steel reinforcing bar filling void spaces. (Applicant Base Mat Testimony, at 18; Staff Base Mat Testimony, at 6, 7)

58. Incomplete concrete consolidation was due to localized congestion of reinforcing steel, embedded plates and trumplates in the area above the tendon gallery roof. The area of the trumplates contained more than a normal complement of reinforcing steel. This congestion by steel embedments hampered the placement of the vibrating tools by the construction workers. (Applicant Base Mat Testimony, at 13-14, 29-30; Staff Base Mat Testimony, at 7-8; McFarland, Meyers, Tr. 357-359)

59. Daniel Corporation quality control personnel were present to assure the timely coordination and relocation of vibratory equipment and craft personnel during the pour. The NRC Staff inspector did not note any quality control deficiencies on the part of construction personnel during the pour. (Staff Base Mat Testimony, at 7-8)

60. Soniscope tests of interior concrete above the tendon trumplates showed that concrete was sound and that there were no hidden defects in the base mat above the gallery. (Applicant Base Mat Testimony, at 23-28)

61. Visible honeycombed areas were repaired by pumping a high strength grout into the voids to bond reinforcing steel. The repairs were equal in strength to the original concrete. (Applicant Base Mat Testimony, at 19-21, 25; Staff Base Mat Testimony, at 8; Varela, Tr. 406)

62. A concrete mixture called dry-pack was used to repair noncritical areas between trumplates which were shallow and structurally insignificant. The dry-

pack repairs were done principally for cosmetic purposes. (Meyers, Tr. 375; Applicant Base Mat Testimony, at 21)

63. A stop work order was issued on dry-pack use because no specification required that the dry-pack be tested, and in fact, it was not being tested prior to use. Subsequent tests of the dry-pack material showed that it possessed compressive strength above the minimum required, and Bechtel determined that previously repaired areas could be used as is. (Applicant Base Mat Testimony, at 21)

64. There was a failure on the part of the Applicant to provide a testing specification for the dry-pack which proved harmless since the material was in fact structurally sound. The material was used for cosmetic repairs only.

65. After repair of the honeycombed area and the soniscope testing, a post-tensioning operation was conducted by applying high tension to the tendons and anchoring them in the trumplates. During the post-tensioning operation, a force as high as 1,600,000 pounds was imposed on the area surrounding each trumplate. When the load was transferred to the tendon anchorage, the load on each trumplate was at least 1,400,000 pounds. These are the most severe loads that will ever be imposed on the trumplates. All tendons have been tensioned and anchored in trumplates with no evidence of distress in the concrete. (Applicant Base Mat Testimony, at 31)

66. NRC Staff inspected preparations for concrete placement including adequacy of reinforcing bar installation before concrete placement in the base mat and found no deficiencies. (Staff Base Mat Testimony, at 1; Staff Ex. 1)

67. Each phase of the concrete placement was planned in advance and discussed by the Applicant to assure that participants were aware of their responsibilities, the placement method, and the areas of congestion from reinforcing steel. A scaled model of the reinforcing steel was used during the planning sessions. (Applicant Base Mat Testimony, at 12; McFarland, Tr. 371)

68. Difficulty of concrete workability in areas of congestion had been anticipated in the planning for concrete placement. Engineers, inspectors, and laborers involved in concrete placement and consolidation were positioned within the steel assembly on the first layer of reinforcement to maintain close control of placing requirements and vibration of the concrete. (Applicant Base Mat Testimony, at 11-13; Staff Base Mat Testimony, at 7; McFarland, Tr. 334-339, 353-358)

69. Applicant's quality control inspectors, supervisory personnel and NRC inspectors were present during concrete placement. The Staff inspectors concluded that performance of the quality control personnel to maintain the quality of concrete was satisfactory. (Staff Base Mat Testimony, at 6-8)

70. Imperfections in concrete were reported by Daniel Corporation to Bechtel as required in a Nonconformance Report NCR 2-0653-C-A dated May 11, 1977. Bechtel rejected the report and requested more detail (Joint Intervenors Ex. 4). A second report filed on June 27, 1977, showed detailed sketches of the defective concrete and a repair plan which Bechtel approved. Union Electric notified NRC

by telephone of the concrete deficiency. (Applicant Base Mat Testimony, at 15; Applicant Ex. 1; Staff Base Mat Testimony, at 3, 4)

71. The Applicant conducted additional training of its personnel after the deficiencies were discovered to prevent their recurrence in future concrete placement operations. (Applicant Base Mat Testimony, at 21, 22; Staff Base Mat Testimony, at 8)

72. The concrete testing and repair were inspected and reviewed by NRC regional inspectors and were found to be satisfactory. (Staff Base Mat Testimony, at 5, 8; Staff Ex. 5)

73. The concrete imperfections in the base mat occurred in spite of precautions and not due to neglect of quality assurance.

74. The quality assurance procedures followed in this instance were consistent with the essential elements of a quality assurance program for inspection, identification of nonconformances, repairs of defects, and documentation and reporting. (Schnell Testimony, ff. Tr. 216, at 22-27)

75. Joint Intervenors claim that the concrete in the reactor base may be faulty because the tests performed in the tendon access gallery may be faulty, and might not demonstrate there is no honeycombing other than that initially discovered. (Joint Intervenors Proposed Finding 155)

76. Joint Intervenors' assertion that the soniscope method is faulty because it does not take account of the fact that sound waves may go around defects in concrete reflects a misinterpretation of the physical principles of the instrument. The deflection of sound waves around defects in concrete is the phenomenon which enables the detection of such defects. The added time required for sound to traverse a tortuous pathway relative to an unobstructed pathway is what is measured and what leads to the interpretation of reduced velocity and faulty concrete if it exists. (Pfeifer Tr. 306-307)

77. Joint Intervenors' argument that the velocity of sound in steel may account for the high sound velocity measured in tests of the base mat is misguided. Expert testimony shows that the interface between steel and concrete often results in a degraded signal or complete obstruction of the signal. While sound might well have a high velocity in steel, we need not take notice of that fact as urged by Joint Intervenors (Proposed Findings, p. 93) since it is beyond dispute that the base mat consists of concrete containing embedded steel. It is the existence of concrete-steel interfaces which might influence the velocity of sound; the sound signal may be halted by interfaces or simply go around the obstruction. (Pfeifer, Tr. 308-309) In either case the result could not be an apparent increase in sound velocity.

78. Joint Intervenors' discussion of the errors possible in aligning a cross hair on an oscilloscope which is necessary to measure the velocity of sound is without merit. (Proposed Findings, p. 94) The WJE report (Applicant Ex. 2, p. 19) lists a table showing average velocities of sound as transmitted through concrete and standard deviations and coefficients of variation for each average. Nonsystematic

errors of measurement including instrument reading errors will be reflected in the calculated standard deviations and coefficients of variation. The coefficients of variation range from 1.0 to 2.1 percent. These errors are sufficiently small to assure that the measured sound velocities reliably exceed the threshold of concern (12,000 ft/sec) below which the integrity of concrete could be in doubt.

79. It was not possible to examine directly by nondestructive methods the remainder of the base mat which is not part of the tendon gallery. That portion of the base mat constitutes 81 percent of the entire structure while the tendon gallery constitutes only 19 percent. The exterior surfaces of the entire base mat including the top and vertical walls were inspected and found to be without defect. (McFarland, Tr. 381-382) Portions unavailable for inspection include the lower surface, *i.e.*, that resting on earth, and the interior concrete of the base mat. (Pfeifer, Tr. 246, 247) Since these areas could not be inspected directly, the Applicant relied on indirect evidence to establish their integrity.

80. The Applicant asserts that the random selection of soniscope test locations around the entire 360 degree circumference of the tunnel, the large number of readings taken at these locations, the fact that three different types of soniscope measurements were made yielding uniform results, the high velocities recorded, and the low statistical variation in the data all lead to a high degree of confidence that there is no occurrence of internal honeycombing in the base slab not only above the tendon gallery but also in the remainder of the base mat. (Applicant Base Mat Testimony, at 29-30)

81. Indirect evidence of interior integrity of base mat concrete comes from understanding the causes of surface honeycombing. The inadequate consolidation was due to the high congestion around the trumplates from embedded steel items, which hampered access and visibility of construction personnel when the concrete was being placed. (Applicant Base Mat Testimony, at 18) Although reinforcing bar occurs throughout the interior of the concrete base mat, it is less congested than at the top and bottom of the mat and it presents fewer difficulties of concrete workability. (Applicant Base Mat Testimony, at 13)

82. The soniscope investigation did not reveal a single instance of interior defects in the tendon gallery, and there is no other evidence that inadequate consolidation occurred in areas of low congestion from steel embedments. (Applicant Base Mat Testimony, at 27-28)

83. Honeycombing in concrete of the lower surface of the inaccessible portions of the base mat cannot be ruled out with certainty. However, the lower surface of the remainder of the base mat is less congested with steel reinforcing bar than in areas where honeycombing was found. It was therefore more accessible to vibration by construction crews than the tendon gallery. (Applicant Base Mat Testimony, at 29)

84. The magnitude of honeycombing that was found in the tendon access gallery would be of no safety significance if found elsewhere on other surfaces of

the base mat. The concern for that in the tendon access gallery relates to the possibility of degrading the function of the tendon trumplates and not to the general concrete stress over the entire base mat. (Applicant Base Mat Testimony, at 30; Meyers, Tr. 240)

85. Joint Intervenors object that: (1) concrete placement reports were not submitted by all quality control inspectors after the base mat was poured and (2) the documentation of the existence of honeycombing in the reactor base mat was not submitted in a timely fashion since more than month passed after completion of concrete placement before a nonconformance report was written. (Joint Intervenors Proposed Findings, at 96, 97)

86. The Staff inspector who was present at the time of concrete placement subsequently cited the Applicant for an infraction because each of the Applicant's quality control inspectors did not submit individual concrete placement reports. A single concrete placement report was submitted which was signed by the inspector who was present at the termination of the pour. The concrete placement report did not include the attributes of concrete placement which were to be verified by the quality control inspector. (Staff Ex. 3, at 22, 23)

87. The Applicant and Staff disagreed on the required number of concrete placement reports. The Applicant believed a single concrete placement report signed at the termination of the pour was adequate. The Staff's view is that a concrete placement report was required from each inspector present at the time of the pour. (McFarland, Tr. 330-331)

88. The Applicant undertook to remedy the deficiency by having each inspector present during the time of concrete placement sign a concrete placement report. Each concrete placement report was similar in information content and appearance and was signed by the individual inspectors during a period covering July and the first part of August of 1977. (McFarland, Tr. 328; Joint Intervenors Ex. 5) In all but one case, the concrete placement reports were signed without individualized comment by the inspector. (Each report referenced the same quality control procedure (QCP 109)). In the Applicant's view the signatures and the absence of comment on the concrete placement reports provides assurance that the individual inspector observed no deficiencies during the pour. It could not be ascertained directly from the reports, however, what activities the signature of each inspector was verifying that he had witnessed. (McFarland, Tr. 322-324; 351)

89. NRC Staff interviewed some of the inspectors after receiving the concrete placement reports and verified that they had observed no deficiencies during concrete placement. (McFarland, Tr. 329; Staff Ex. 4, p. 4)

90. The Board concurs with Joint Intervenors that the procedure followed here was defective. We criticize the documentation procedure but do not find evidence that the inspectors failed to perform their duties at the time the concrete was

actually placed. Specifically, we find that the placement reports signed without comment some three months after the event took place to be essentially worthless.

91. The Board is unable to determine from the belated reports whether they were signed by the inspectors in a perfunctory manner as simply another burden of paperwork or whether the signatures have genuine meaning. While some interviews were done by the Staff inspector, the interviews were not documented. Thus, no genuinely useful written record exists to document the observations of inspectors during the placement of concrete in the base mat.

92. The Board finds no deficiency with regard to the timing of the nonconformance report. In the approximate month between completion of the base mat pour and the filing of the first nonconformance report, the concrete was left to harden, the concrete forms were then removed, inspections were performed and honeycombed areas were chipped to sound concrete. In light of the actions that had to be taken, the elapsed time of one month from the termination of the pour appears reasonable. (McFarland, Tr. 255, 256)

93. The quality assurance procedures employed in the construction of the base mat worked properly in that precautions were taken to prevent deficiencies, deficiencies that occurred in spite of the precautions were found promptly, appropriate reports and tests were made except as noted above, and repairs were made which restored the defective areas to original design specifications.

C. Contention I.C.2. Honeycombing in the Reactor Building Dome

94. The top of the reactor building presented an unusual and difficult problem of concrete placement which resulted from angles of placement ranging from 45 degrees to nearly horizontal at the top of the dome and also due to the placement being accomplished without the use of an outside concrete form. (Applicant Reactor Dome Testimony, at 9, 10; Tye, Tr. 2012, 2036)[6]

95. Freshly placed concrete in this region of the dome was vibrated to produce consolidation. Vibration near the outside face of the concrete caused subsidence in a downward direction. Concrete had to be replaced from lower to upper levels while the vibration process continued. (Applicant Reactor Dome Testimony, at 10)

96. After the concrete hardened, four areas of honeycombing were found on the outer surface of the dome. The honeycombed areas were chipped to sound concrete. The chipping process revealed air gaps of approximately ¼ to ½ inch in diameter between the horizontal reinforcing steel and the concrete approximately 4 to 6 inch. from the outer surface of the dome. Chipping was completed November 6, 1980. (Applicant Reactor Dome Testimony, at 11, 12; Board Ex. 6)

97. The gaps were caused by the downward movement or subsidence of the concrete away from the reinforcing bars when it was being placed. (Applicant Reactor Dome Testimony, at 11)

98. The Daniel Corporation filed a nonconformance report to Bechtel Corporation on November 10, 1980. Bechtel and Union Electric personnel questioned Daniel whether more investigation was needed since additional imperfections might be present. Union Electric notified the NRC Staff on December 5, 1980 of a potentially significant deficiency. (Applicant Reactor Dome Testimony, at 12, 13)

99. On December 13, 1980, three additional areas of honeycombing were discovered in the dome area following removal of grease vent blockouts at the request of NRC Staff. At that point there were seven known areas of concrete imperfections in the dome. Repairs of these areas involved chipping and shaping each cavity to receive replacement concrete which was of the same class and mix as originally used. (Applicant Reactor Dome Testimony, at 13-15; Staff Ex. 8, at 4)

100. A number of nondestructive and destructive examinations of the dome were done including nuclear densometer testing, boroscopic examination, microseismic (pulse-echo) examination, selective excavation and engineering analysis. (Applicant Dome Testimony, at 18) The nuclear densometer and boroscopic examinations did not reveal further evidence of unsound concrete. (Applicant Reactor Dome Testimony, at 18-19; Applicant Ex. 19, at 11-15)

101. The pulse-echo technique was used at 1,671 locations as a means of searching for imperfections throughout the three foot thickness of the dome. (Applicant Reactor Dome Testimony, at 19-20; Goddard, Tr. 2056-59) The pulse-echo method has been used extensively for similar applications and is reliable and accurate. (Staff Hawkins Reactor Dome Testimony, at 4; Hawkins, Tr. 2075-76 and 2078)

102. The results of the pulse-echo testing showed that 28 of 1,671 tests or 1.68 percent were of possible structural significance but none of the areas tested showed a sufficient number of such readings to classify the area as structurally defective. (Applicant Reactor Dome Testimony, at 21-22, Applicant Ex. 19, at 22-24)

103. The Applicant excavated concrete at six test points which had shown defects by the pulse-echo method, and excavations confirm that a correlation exists between the pulse-echo readings and the actual imperfections. (Applicant Reactor Dome Testimony, at 22-23; Applicant Ex. 19, at 24-26)

104. Staff review of test results indicated that more testing was needed. Applicant and Staff then concluded after additional testing that the extent of imperfections was clearly identified. (Applicant Ex. 19, at 35-41; Applicant Reactor Dome Testimony, at 24-26; Staff Hawkins Reactor Dome Testimony, at 4; Hawkins, Tr. 2073)

[6] Applicant Reactor Dome Testimony, ff. Tr. 2010 hereafter cited as Applicant Reactor Dome Testimony. Staff Testimony, ff. Tr. 2067 hereafter cited as Staff Ma Reac. — Dome Testimony or Staff Hawkins Reactor Dome Testimony.

105. An engineering analysis showed that the structural integrity of the dome would not be jeopardized when subjected to all design load conditions including postulated accidents even if 50 percent of the mechanical bond were completely lost around the entire perimeter of the bars in the outside layer of reinforcing steel. (Applicant Reactor Dome Testimony, at 26)

106. The Staff performed an independent engineering analysis and concluded that the dome is adequate as built. This is based on findings that the original design contained more than enough extra steel reinforcement to compensate for the small deficiencies found in the concrete. (Staff Ma Reactor Dome Testimony, at 3; Ma Tr. 2077-78)

107. The strength of the concrete in the dome exceeds the design strength by a considerable margin and the overall quality of the concrete is not in question. (Applicant Reactor Dome Testimony, at 11, 12; Staff Ma Reactor Dome Testimony, at 6)

108. Imperfections in the reactor dome were discovered through routine inspections. The imperfections were reported to Bechtel, the architect/engineer, who in turn initiated further examinations. Reports were made to NRC. The extent of defects was fully investigated and repairs were properly made. The Applicant's quality assurance program functioned properly in this instance. (Applicant Reactor Dome Testimony, at 27; Staff Hawkins Reactor Dome Testimony, at 5)

D. Contention II.A.1. SA-358 Piping

109. SA-358 is an ASME material specification for welded stainless steel pipe. (Stuchfield, Tr. 1456, 1545) This material specification provides a series of limits and permissible variations for several dimensional requirements for the finished pipe. (Applicant SA-358 Piping Testimony, at 5)[7]

110. Daniel employees observed a spool piece irregularity and after an ultrasonic test (UT) confirmed a thin wall area, a nonconformance report (NCR) was issued. (Applicant SA-358 Piping Testimony, at 5, 6)

111. NRC Staff and Applicant personnel conducted measurements of the ovality of the pipe, determined that the actual maximum ovality is 0.86 percent, and therefore was within the one percent difference between major and minor outside diameters permitted by material specification SA-358. (Applicant SA-358 Piping Testimony, at 5, 8; Staff Foster SA-358 Piping Testimony, at 2; Staff Beeman SA-358 Piping Testimony, at 2, 3)

[7] Applicant SA-358 Piping Testimony, ff. 1537, hereafter cited as Applicant SA-358 Piping Testimony. Staff SA-358 Testimony, ff. Tr. 1681 hereafter cited as Staff Foster, Beeman or Key SA-358 Piping Testimony

112. The measured minimum wall thickness of the pipe in question was 0.814 inch. (Applicant SA-358 Piping Testimony, at 8, 9; Staff Foster SA-358 Piping Testimony, at 2, 3) Bechtel and the NRC Staff performed independent calculations and concluded a minimum wall thickness of 0.814 inch is acceptable. (Staff Beeman SA-358 Piping Testimony, at 3; Staff Ex. 7, at 8)

113. Daniel measured weld reinforcement on the inside of the SA-358 pipe with a reinforcement height of 3/16 inch and documented it in a nonconformance report. Bechtel initially erroneously dispositioned this NCR. (Applicant SA-358 Piping Testimony, at 9, 10) Daniel elected to rework the item to bring the weld into compliance with the ASME code by localized grinding. (*Id.*, at 16-17; Foster, Key, Tr. 1706, 1707; Laux, Tr. 1625-1627)

114. The Daniel NCR identified overlap in the same area as the excess weld reinforcement. Bechtel advised that overlap is not listed in the ASME code as a rejectable condition for radiography, as overlap does not affect the volumetric quality of the weld. (Applicant SA-358 Piping Testimony, at 11, 12) The overlap was reworked by grinding. (*Id.*, at 16; Foster, Key, Tr. 1706, 1707)

115. Joint Intervenors assert that the overlap condition could have been caused by a "melt-through" or "drop-through" — *i.e.*, the first weld pass from the outside of the pipe melting through the pass made from the inside. (Applicant SA-358 Piping Testimony, at 12; Staff Key SA-358 Piping Testimony, at 2)

116. The overlap indicates that drop through could not have occurred. (Beeman, Tr. 1752) Melt-through would have resulted in a surface condition on the inside of the pipe which would be readily detected. (Applicant SA-358 Piping Testimony, at 13; Stuchfield, Tr. 1563-1564, 1642-1643) If melt-through had occurred, surface porosity would have quickly occurred and no such porosity was detected. (Applicant SA-358 Piping Testimony, at 13, 14) No radiographic reviews of the area by Union Electric and Daniel personnel resulted in this weld being declared unacceptable. (*Id.*, at 15)

117. NRC Staff witnessed radiograph tests (Staff Foster SA-358 Piping Testimony, at 3; Applicant SA-358 Piping Testimony, at 17), inspected the SA-358 pipe, reviewed the pipe's documentation, inspected the radiographs of the weld seam taken after the rework and concluded the weld to be free of defects and within ASME code acceptance criteria. (Staff Foster SA-358 Piping Testimony, at 3; Applicant SA-358 Piping Testimony, at 17; Staff Key SA-358 Piping Testimony, at 1, 2 Key, Tr. 1751. *See also*, Beeman, Tr. 1751-1752)

118. Two fissures in the questionable weld were reported in NRC IE Inspection Report No. 50-483/81-04. (Staff Ex. 7) Applicant's and Staff's witnesses concurred that there is no evidence of fissures. (Applicant SA-358 Piping Testimony at 15, 16; Key, Tr. 1710, 1750; Beeman, Tr. 1712-1714; Stuchfield, Tr. 1648)

E. **Contention II.A.2. SA-312 Piping**

119. SA-312 is an ASME material specification for both seamless and welded stainless steel pipe. (Applicant's SA-312 Piping Testimony, at 19 and Figure 1*; Stuchfield Tr. 1794-1795, 1809-1810)

120. Those safety-related systems at the Callaway Plant which contain double-welded SA-312 pipe are designated as ASME Class 2 or Class 3. For use in Class 2 systems, ASME Section III requires that welded pipe be nondestructively examined, usually by the ultrasonic method. (Applicant SA-312 Piping Testimony, at 16-17; Hurd, Tr. 1782-1788; Stuchfield, Tr. 1824-1825)

121. Centerline lack-of-penetration (CLP) occurs in double-welded SA-312 pipe when complete through-wall fusion does not occur between the inside and outside welds during welding of the longitudinal seam. (Applicant SA-312 Piping Testimony, at 17 and Figure 1; Staff Rutherford SA-312 Piping Testimony, at 3)

122. The CLP problem in SA-312 piping was first identified in the fall of 1978. Imperfections, principally for lack of center line of penetration, were found in the longitudinal welds of double-welded SA-312 pipe being fabricated into piping subassemblies for the Palo Verde Nuclear Generating Station. A similar rejection was reported by Pullman Power Products (PPP) on pipe purchased from Youngstown Welding and Engineering Company (YWEC) for the San Onofre 2 and 3 Nuclear Generating Stations. The Palo Verde and San Onofre owners reported these findings to the NRC. (Applicant SA-312 Piping Testimony, at 17, 18; Staff Rutherford SA-312 Piping Testimony, at 1-2; *see* Applicant Ex. 11, at 1)

123. The NRC Office of Inspection and Enforcement issued I&E Bulletin 79-03 in March, 1979, requiring all operating license and construction permit holders to determine if similar pipe had been or would be incorporated into safety-related piping systems. In response to this Bulletin, Applicant determined that pipe manufactured by YWEC had been used in piping subassemblies fabricated for the Callaway Plant. A complete schedule of the location of all YWEC-supplied pipe was provided to the NRC and a program for ultrasonic examination of longitudinal welds was established as required by the Bulletin. (Applicant Ex. 10; Applicant SA-312 Piping Testimony, at 19)

124. A generic investigation into the CLP problem was undertaken by Bechtel.

125. This investigation determined that the principal cause of CLP was the wide range of allowable welding parameters permitted by the YWEC qualified welding procedure. (Stuchfield, Tr. 1799-1804; Applicant SA-312 Piping Testimony, at 20-21; Intervenor Ex. 61; Egan, Tr. 1807)

* Applicant SA-312 Piping Testimony, ff. 1773; hereafter cited as Applicant SA-312 Piping Testimony. Staff Testimony, ff. Tr. 1898, hereafter cited as Staff Rutherford SA-312 Piping Testimony.

126. The Bechtel investigation concluded that the ASME Code-required ultrasonic examination cannot reliably detect CLP in double-welded SA-312 pipe. (Applicant SA-312 Piping Testimony, at 23; Stuchfield, Tr. 1797, 1827, 1828; *see also* Applicant Ex. 11, at 2, 3, 7, 8)

127. Bechtel determined the maximum amount of CLP in the SA-312 piping produced by YWEC to be 26 percent. (Applicant SA-312 Piping Testimony, at 24-25; Stuchfield, Egan, Tr. 1811-1814; *see also*, Staff Rutherford SA-312 Piping Testimony, at 4; Applicant Ex. 11, at 2-3)

128. Tensile and hydrostatic (burst) tests established that with 26 percent CLP, SA-312 piping will meet all of the ASME mechanical property requirements and that even with 47 percent CLP, yield strength requirements are met. (Applicant SA-312 Piping Testimony, at 25-26; Applicant Ex. 11, at 2, 3, 7) Tests were performed on three SA-312 pipe sections with known CLP of 15 percent, 40 percent and 55 percent. The lowest burst pressure recorded was for the pipe with 55 percent CLP which burst at 3000 psi, far in excess of the ASME Code-required hydrostatic test pressure of 882 psi for the same size and schedule pipe. (Applicant SA-312 Piping Testimony, at 27-28; Staff Rutherford SA-312 Piping Testimony, at 6; Applicant Ex. 11, at 2, 3, 14-19; *see also*, Rutherford, Tr. 1906)

129. Two engineering analyses of SA-312 pipe with CLP were also performed by Aptech Engineering Services, Inc. (Aptech). The first, a fracture analysis, demonstrated that because of the very ductile nature of the stainless steel material used in SA-312 piping, the failure mode of the pipe would not be brittle fracture, but rather, a "leak-before-break" and ductile fracture mode. (Applicant SA-312 Piping Testimony, at 28-30; Staff Rutherford SA-312 Piping Testimony, at 5; Applicant Ex. 12, at p. iii) A load limit analysis showed that the possible presence of CLP is not a concern. (Applicant SA-312 Piping Testimony, at 30-32; Staff Rutherford SA-312 Piping Testimony, at 5, 6; Egan, Tr. 1881; *see* Applicant Ex. 12, at pp. 7-5 to 7-7, 10-1, 10-2)

130. The second fatigue analysis determined that fatigue failure as a result of the possible presence of CLP was not a concern. (Applicant SA-312 Piping Testimony, at 33, 34; *see* Applicant Ex. 13, at 9-1)

131. The testing and analyses performed during this generic investigation of the CLP problems established that double-welded SA-312 piping, even with amounts of CLP substantially in excess of that found in production pipe, will function as intended with an adequate margin of safety. (Applicant SA-312 Piping Testimony, at 34-35)

132. The NRC adopted a Bechtel recommendation in issuing I&E Bulletin 79-03A which made several changes to the directives originally contained in I&E Bulletin 79-03. Inasmuch as all piping systems containing double-welded SA-312 pipe at Callaway are subject to maximum hoop stresses less than 85 percent of the ASME Code-allowable stresses, no further action was required by Applicant.

(Applicant SA-312 Piping Testimony, at 38; Staff Rutherford SA-312 Piping Testimony, at 2; Applicant Ex. 14; Applicant Ex. 11, at 4)

133. Joint Intervenors questioned the use in Callaway piping systems of SA-403 fittings which may contain CLP. SA-403 is a specification for wrought stainless steel pipe fittings, such as elbows, tees and reducers. The double-welded SA-312 pipe used to manufacture fittings is no different than the double-welded SA-312 pipe used for straight-run pipe, and could contain CLP to the same extent as straight-run pipe. (Applicant SA-312 Piping Testimony, at 38, 39; *see also*, Applicant Ex. 16; Joint Intervenors Ex. 64)

134. No SA-403 fittings are used in Callaway piping systems which have hoop stresses greater than 85 percent of the ASME Code-allowable stresses. (Applicant SA-312 Piping Testimony at 39; Stuchfield, Tr. 1777; Hurd, Tr. 1790-1793)

F. Contention II.B. Piping Subassembly Deficiencies

135. Preassembled pipe formations are pre-designed, manufacturer-fabricated formations containing piping, fittings, valves, pumps, strainers, tanks and other similar equipment. (Applicant G&W Testimony, at 8-9)[9]

136. Nondestructive examinations ("NDE") were performed on welds by G&W in order to detect any conditions not in conformance with ASME Code criteria. In-process and final surveillance inspections were conducted by Bechtel, and Daniel personnel performed receipt inspections. (Applicant G&W Testimony, at 10-11; Staff Hansen Testimony, at 4)

137. In March, 1979, a Daniel construction worker informed a Daniel welding inspector at the Wolf Creek site of potential deficiencies in a preassembled piping formation supplied by Gulf & Western. (Applicant G&W Testimony, at 11; Powers, Laux, Tr. 1929, 1930; *see also*, Intervenor Ex. 69)

138. Applicant's Construction QA group conducted an extensive audit of the G&W formations at the Callaway site which determined that the G&W formations exhibited noncompliances to the Bechtel specification and to ASME Code requirements in the areas of both radiographic technique and visible weld discrepancies. SNUPPS QA Committee audit reviewed G&W's manufacturing and inspection activities. G&W agreed to conduct a 100 percent review of the weld radiographs. (Applicant G&W Testimony, at 12, 13; Staff Hansen Testimony, at 2)

139. G&W was required to rework all safety-related formations and onsite rework was monitored by Union Electric, Daniel and Bechtel. Radiographic technique deficiencies and weld deficiencies continued to be encountered and

[9] Applicant's Gulf and Western Testimony (Piping Subassembly Deficiencies), ff. Tr. 1920, hereafter cited as Applicant G&W Testimony. Staff Testimony, ff. 1979, hereafter cited as Staff Hansen Testimony.

G&W was directed to cease its onsite rework efforts. (Applicant G&W Testimony, at 13-16; Staff Hansen Testimony, at 3; Intervenor Ex. 69, Final Report, at 1-2)

140. Daniel assumed responsibility for onsite rework and all repairs have been completed and all welds now meet the applicable criteria. (Applicant G&W Testimony, at 16, 17; Staff Hansen Testimony, at 3; Key Testimony, ff. Tr. 1979, at 2)

G. Contention on the Quality Assurance Program

141. The Board has found deficiencies in certain elements of the Applicant's quality assurance or quality control program. We tabulate our findings below for each contention.

Embedded Plates

1. Bechtel evaluation of weld reinspection data without written documentation was inadequate.
2. There was a lack of awareness on the part of Applicant and Bechtel of Daniel weld inspection data for a period of months, leading to doubts of careful monitoring of construction at Callaway.
3. There was inconsistent reporting of weld deficiencies by Daniel, indicating supervisory weakness.
4. Serious questions were raised as to the quality of welded embeds fabricated by Cives Company.
5. And finally, there was an elapsed time of three years from discovery to final resolution of the embed problem on the part of NRC headquarters officials.

Concrete in the Reactor Base Mat

1. There was inadequate resolution of the inspection documentation problem related to concrete placement reports for the reactor base mat.
2. The use of untested dry-pack concrete for repairs was questionable. (In this instance the failure proved harmless to structual integrity and we consider it here only as incremental evidence pertinent to the effectiveness of the quality assurance program)

Concrete in the Reactor Dome

No defects relevant to the adequacy of the Applicant's quality assurance program were found.

SA-358 Piping

No defects relevant to the adequacy of the Applicant's quality assurance program were found.

SA-312 Piping

No defects relevant to the adequacy of the Applicant's quality assurance program were found.

Piping Subassembly Deficiencies

No defects relevant to the adequacy of the Applicant's quality assurance program were found.

142. In judging the adequacy of the Applicant's quality assurance program we compare actual performance against the functional standards stated in the testimony of Mr. Schnell who is Vice President Nuclear for Union Electric. (Schnell Testimony, ff. Tr. 216, at 23-27)

143. It is uncontroverted that the Applicant has in place a comprehensive QA/QC program at Callaway. (Schnell Testimony, ff. Tr. 261, at 35)

144. There is no evidence from the contentions in this case that all or a substantial part of the overall QA program failed to function during construction at Callaway. The program had effective overall control of construction quality and it coped effectively to resolve problems that were identified in the contentions in this case. However, as noted in our comments on the individual contentions, there were several significant quality control problems.

145. The ultimate resolution of each problem identified in this case assures, however, that there are no safety concerns in the Callaway plant relative to those contentions tried in this case.

146. The deficiencies in quality assurance noted in this decision do not collectively show a pattern of programatic breakdown in the QA program. While there is no cause for complacency regarding the deficiencies noted, particularly those related to the embedded plate problem, we conclude that the deficiencies are isolated problems having no broader implication for the overall effectiveness of the QA program.

IV. CONCLUSIONS OF LAW

The Board has considered all the evidence submitted by the parties and the entire record of this proceeding consisting of the Commission's Notice of Hearing, the pleadings filed by the parties, the transcripts of the hearing and the exhibits received into evidence. All issues, contentions, and proposed findings presented by the parties, but not addressed in this decision, have been found to be without merit or unnecessary to our decision. The findings of fact presented above are supported by reliable, probative and substantial evidence in the record.

The Board has not yet heard evidence with respect to, and this Partial Initial Decision does not address, the emergency planning contentions raised by Intervenor Reed. Based upon a review of the entire record in this proceeding and the foregoing findings of fact, the Board enters the following conclusions of law.

This is a contested proceeding on an application for an operating license for a utilization facility, and the Board has made findings of fact and conclusions of law on the matters put into controversy by Joint Intervenors with respect to construction defects at Callaway and Applicant's quality assurance program. Contentions in regard to radioactive releases have been withdrawn. The matters put into controversy by Intervenor Reed are still pending before the Board. The Board has not determined that a serious safety, environmental, or common defense and security matter exists. *See* 10 CFR §2.760a. Other findings required to be made prior to the issuance of an operating license, except for the remaining matters in controversy, are to be made by the Director of Nuclear Reactor Regulation. *See Id.* and 10 CFR §50.57.

Having decided all matters in controversy raised by Joint Intervenors, in favor of authorizing operation of the facility, the Board concludes that as to the matters decided herein, the Director of Nuclear Reactor Regulation would be authorized, upon making the requisite findings with respect to matters not resolved in this Partial Initial Decision, and subject to the Board's resolution of outstanding matters in controversy, to issue to Applicant a license to operate Callaway Plant, Unit 1. Such authorization is not now granted by the Board, however, and will not be granted until the Board resolves the outstanding matters in controversy or issues a further order to the contrary.

V. ORDER

WHEREFORE, IT IS ORDERED, in accordance with 10 CFR §§2.760(a) and 2.762, that this Partial Initial Decision shall constitute the final action of the Commission thirty (30) days after the date of issuance hereof, unless exceptions are taken in accordance with Section 2.762 or the Commission directs that the record be certified to it for final decision. Any exceptions to this Partial Initial Decision or designated portions thereof must be filed within ten (10) days after

service of the decision. A brief in support of the exceptions must be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff). Within thirty (30) days of the filing and service of the brief of the appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

Glenn O. Bright
ADMINISTRATIVE JUDGE

Dr. Jerry R. Kline
ADMINISTRATIVE JUDGE

James P. Gleason, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland
this 13th day of December, 1982.

APPENDIX I

WITNESS LIST

Contention	Witness	Transcript
Embedded Plates		
Applicant	Donald F. Schnell V.P., Union Electric Co. Bernard L. Meyers Project Mgr., Snupps Bechtel Power Corp.	501

	Eugene W. Thomas Supervisor, Snupps Bechtel Power Corp. Kirit Parikh Civil Eng. Supervisor Bechtel Power Corp. John W. Fisher Professor, Lehigh Univ. Civil Engineering Roger G. Slutter Professor, Lehigh Univ. Civil Engineering	
Staff	Eugene W. Gallagher Civil Engineer, Office of Inspection and Enforcement Nuclear Regulatory Commission	1261
Licensing Board	Harold J. Starr Project Manager Daniel International Corp. John A. Holland Project Piping Engineer Daniel International Corp.	1342
Honeycombing-Base Mat		
Applicant	Bernard L. Meyers Thomas H. McFarland Construction Supervisor Union Electric Company Donald W. Pfeifer Project Manager Wiss, Janney, Elstner & Associates	227
Staff	Anthony A. Varela Civil Engineer Nuclear Regulatory Commission	392
Honeycombing-Reactor Dome		
Applicant	Eugene W. Thomas	2010

	Guy H. Goddard Deputy Group Supervisor Bechtel Power Corp. B. Christopher Tye Process Consultant Bechtel Power Corp. Richard A. Muenow Pres., Muenow & Associates Consulting Engineers	
SA-358 Piping		
Applicant	Michael F. Stuchfield Area Office Manager Bechtel Power Corp. Joseph V. Laux Supervising Engineer Bechtel Power Corp.	1537
Staff	Gordon Beeman Research Engineer Battelle Memorial Institute James Foster Investigator Nuclear Regulatory Commission William Key Inspector Nuclear Regulatory Commission	1681
SA-312 Piping		
Applicant	Bernard L. Meyers Michael F. Stuchfield John D. Hurd Snupps Group Supervisor Bechtel Power Corp. Geoffrey R. Egan Technical Director Aptech Engineering Services	1773
Staff	William R. Rutherford Senior Mechanical Engineer Formerly with NRC and now a Consultant	1897

Piping Subassembly Deficiencies		
Applicant	Joseph V. Laux Bernard L. Meyers Michael F. Stuchfield Robert L. Powers Superintendent, Site Q.A. Union Electric Company Harry J. Porter Quality Manager, Snupps Project Bechtel Power Corp.	1929
Staff	William Key William A. Hansen Senior Resident Inspector Nuclear Regulatory Commission	1977
Quality Assurance		
Applicant	Donald F. Schnell	216

APPENDIX II

INTERVENORS' CONTENTIONS
FAILURE OF THE QUALITY ASSURANCE PROGRAM

Surveillance and inspection functions of Applicant Union Electric Company, and others, including Bechtel Power Corp. (lead architect/engineer), Daniel International Corp. (construction contractor) and Code Authorized Nuclear Inspectors, failed to ensure the quality of safety-related material, structures, systems and components through all phases of their fabrication, construction, testing and inspection contrary to the quality assurance criteria of 10 CFR Part 50 Appendix B. Many vendor-supplied components were on the construction site and were approved for installation before code-defined deficiencies and nonconformances were identified. During construction deficiencies and nonconformances were accepted against code requirements. Without effective surveillance and inspection by the Applicant, and others, of material suppliers, component vendors, and construction contractors, all safety-related material, structures, systems, and components must be considered of questionable integrity. Because effective surveillance and inspection were not performed, the safe operation of the Callaway Plant is in jeopardy and should not be licensed.

Deficiencies and nonconformances which evidence the failure of the quality assurance program include but are not limited to the following:

I. SUBSTANDARD REINFORCED CONCRETE CONSTRUCTION

A. Embedded Plates

Embedded plates, or embeds, so called because they are embedded in concrete, are fixtures installed in concrete walls to support the ends of load-bearing steel beams, piping and other structures. The plates are made of steel with short steel studs welded to one face, like the bristles of a brush. They are mounted flush with the wall surface, with the studs extending into the concrete. The exposed surfaces of the plates serve as point of attachment for girders and other structural members. If an embedded plate tears loose from a wall, the result could be the collapse of an entire floor, breakage of critical pipes in the primary and emergency core cooling systems, and even core melt-down (Class 9 accident).

When the Callaway Plant was approximately five and one-half to seven percent complete, a stop-work order was issued on June 9, 1977, when it was discovered that some of the studs were not properly welded to the embedded plates. (*See* NRC Report No. 50-483/77-10, p. 8). Prior to June 9, 1977, 480 plates had been installed in the plant. (*See* NRC Report No. 50-483/80-14, p. 4). The NRC and the Applicant do not know how many of those 480 plates contain faulty welds, they do not know where those plates are located in the plant, they do not know what loads each plate must bear, and they do not know what the consequences of plate failure would be to the safe operation of the plant and to the health and safety of the public. (*See, e.g.*, NRC Report No. 50-483/80-14 Attachment A — item 17, pp. 4-5 and Attachment B — item 17, pp. 5-6).

The Applicant and NRC staff do know that after the June 1977 stop-work order, many unused plates had to be repaired (*See* NRC Report No. 50-483/77-10, p. 8) or were returned to the manufacturer. There is evidence of multiple defects on some plates. (*See* NRC Report No. 50-483/80-14, Attachment B, p. 3). Although it is not known whether the manufacturer inspected the plates before shipping them to Callaway (*see* NRC Report No. 50-483/80-14, Attachment B, p. 2), none of the 480 installed plates were removed and reinspected, and, none were repaired or replaced.

During the process of evaluating the question whether the embedded plates presented a safety-significant problem, the Applicant improperly determined, with the NRC's apparent approval, that certain exceptions to structural welding code standards would be tolerated. (*See, e.g.*, NRC Report No. 50-483/80-14, pp. 7-10).

We contend that inadequate and incomplete inspection and testing were performed. Omissions include the failure to conduct live-load tests and the failure to consider whether defective plates could withstand the effects of an earthquake as per 10 CFR Part 100, Appendix A, Section VI.

B. Cracks in Concrete*

There exist several cracks in concrete structures at the Callaway Plant that affect its safe operation. Examples include, but are not necessarily limited to, the following:

1. A crack up to ¼ inch wide was discovered in the Reactor Building in the reactor cavity moat area in May 1977, a month after the concrete mat was poured. The crack extended approximately 270 degrees around the circumference. Upon visiting the site in June 1977, an NRC inspector was unable to view repairs performed on this crack because work had progressed to an extent that made physical inspection of the repair impossible. (*See*, NRC Report No. 58-483/77-06, pp. 20-21.)
2. The NRC was notified by a Callaway Plant ironworker in January 1978 that a lift of the north wall of the Control Building had been poured above a part of the wall which contained a crack approximately 12 feet long and 8 inches deep, and which extended from the inside to the outside of the wall and which apparently had been overlooked by the Applicant's quality assurance personnel. (*See*, NRC Report No. 50-483/78-01, p. 20.)

C. Honeycombing

Instances of air pockets or voids, known as honeycombing, have been found in concrete structures at the Callaway Plant. As described in NRC Regulatory Guide 1.55, "Concrete Placement in Category I Structures":

> [T]he presence of numerous concrete voids which have been detected at or near the surfaces of nuclear containment buildings raises concern about the density of portions of these and other concrete structures that cannot readily be inspected. For such unaccessible areas, the only method of assuring a quality concrete structure is through good planning and control of the placement of concrete and all items embedded in it.

The instances of honeycombing at Callaway include but are not limited to:

1. Reactor Building Base Mat

On May 31, 1977, voids described by the NRC as up to six inches, but described by a worker as big enough for a man to crawl into, were found in the tendon access gallery of the reactor base mat. (*See*, NRC Report No. 50-483/77-06, pp. 21-22)

*Eliminated from proceeding by grant of summary disposition

:pairs were undertaken at this time, but during the NRC inspection of August -September 2, a stop-work order was issued because of a discrepancy in work ecifications concerning the testing of dry-pack group. (*See*, NRC Report No.)-483/77-07, p. 13). The stop-work order was lifted on December 7, 1977, after e necessary changes in specifications were made (*see*, NRC Report No. 50-483/ 8-01, pp. 2-3), but no information is available on whether any testing was rformed on repairs done prior to the stop-work order. A report dated August 1, 977, by Wiss, Janey, Elstner and Associates, Inc., described a soniscope study rformed by this firm to determine the possibility of additional honeycombing ithin the 10 foot thick base slab. The study states that, "Based upon a 25 percent mple . . . internal honeycomb *probably* does not occur in the base slab, except at ose 19 areas where honeycomb was visible." (*See*, NRC Report No. 50-483/77- , pp. 12-13, emphasis added.) This assessment of probability is the only surance given that no additional honeycombing exists. According to a letter from mes Keppler, Director, Region III, NRC, to Kay Drey dated January 3, 1979, e tendon access gallery represents nineteen percent of the base mat area. In the me letter Mr. Keppler described the twenty-four large holes which were re- ired, as follows:

> Large voids are defined as those that require approval prior to repair. The largest void in this category was approximately 22 square feet in surface area, and it was irregular in shape. Its maximum depth was 17 inches, and its average depth was 8 inches. The smallest void in this category was approximately 0.25 square feet in surface area, and its maximum depth was 5 ½ inches. The size of the remainder of the voids in this category varied between those previously described.

Reactor Building Dome

Four areas of concrete imperfection in the Reactor Building dome were identi- d by Union Electric personnel during an inspection on August 22 and 27, 1980. ese imperfections were attributed to "the complex nature of those portions of the me slab where the imperfections occurred." However, on December 12, 1980, C personnel noticed that blockouts for the tendon grease vents had not been noved to facilitate inspection, and after the removal of the blockouts on cember 13, three additional honeycomb areas were found. After conducting erviews with UE personnel concerning the three new void areas, the NRC ncluded that, "There appeared no plausible explanation for their occurrence," d that ". . . there was not adequate assurance that the imperfections' existence re limited to only those areas identified." (*See*, NRC Report No. 50-483/80-30, 3-4.)

D. Concrete Cover*

There exist many areas where concrete coverage of reinforcing bars in concrete walls and floors at the Callaway Plant does not adhere to requirements. Bechtel Power Corporation's interpretation of the cover requirements was that minimum cover requirements could be reduced by one-third, but the NRC stated in a meeting between NRC, UE, Bechtel, and Daniel International personnel on January 23, 1978, that no reduction of the two-inch cover minimum is acceptable. However, the NRC indicated that it would be acceptable "if the cover requirements were fully met in the area of the sixth lift, utilizing the fifth lift as a transition area." (*See*, NRC Report No. 50-483/77-11, pp. 10-11.)

Some examples of nonadherence to concrete cover requirements are as follows:

1. At 340 degrees azimuth, vertical reinforcement bars and supporting bars for the horizontal tendon sheathing in the 3rd lift of the reactor containment wall had concrete cover "less than that specified by NRC requirements, but within the concrete cover requirements as interpreted by licenses and contractors." (*See*, NRC Report No. 50-483/77-11, pp. 4 and 9-11.)
2. NRC inspectors observed the preplacement preparation of the fourth lift of the exterior wall of the Reactor Containment Building, finding 14 unacceptable items, in half of which concrete cover was less than the 2 inch minimum required or more than the 9.6 inch maximum required. These items include instances where the concrete cover is as small as 5/8 of an inch (at azimuth 210 degrees) and as great as 12 inches (at azimuth 200 degrees). Some items were corrected, and the rest were within the range judged to be acceptable below the sixth lift because of the one-third placement tolerance. (*See*, NRC Report No. 50-483/78-01, pp. 9-11.)

II. SUBSTANDARD PIPING

A. Material Manufacturing Deficiencies

Safety-related pipe installed at Callaway was manufactured by a company or companies which did not have adequate control of welding parameters. This resulted in known cases of defects which did not comply with the requirements of the American Society of Mechanical Engineers (ASME) Code. The evaluation and acceptance of those defects and deficiencies were not done in accordance with the ASME Code. The safety of pipe installed at Callaway remains in question and

*Eliminated from proceeding by grant of summary disposition.

demands further investigation before an operating license should be issued. For example:

1. In May 1979 a pipefitter discovered and reported a substandard piece of ASME Class II SA-358 piping which had been installed in the emergency core cooling system. The pipe was substantially out-of-round, was machined below the minimum wall, and had rejectable weld defects on the inside of a longitudinal seam weld. (*See*, NRC Report No. 50-483/80-10.) The piping was approved for shipment at the vendor's, was accepted on site, and was installed despite these deficiencies.
2. Substandard fusion welded SA-312 pipe manufactured by Youngstown Welding and Engineering Company and fabricated into safety-related pipe spools by Dravo Corporation has been installed at the Callaway Plant. (*See*, NRC/IE Bulletin 79-03 and 79-03A, and Union Electric letter ULNRC-314 dated May 11, 1979, to NRC - Region III). The evaluation and acceptance of this substandard SA-312 piping were not performed according to the requirements of Section III of the ASME Code.

B. Piping Subassembly Deficiencies

Additional evidence of deficiencies in surveillance and inspection functions include the following: In 1979 it was discovered that pre-assembly piping formations with defective welds from Gulf & Western were accepted and were installed at Callaway. After installation it was also discovered that the vendor had used improper radiographic techniques. (*See*, SNUPPS letter SLNRC-79-20 of November 29, 1979, to NRC — Region I, and Bechtel Final Report of November 28, 1979.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Peter B. Bloch, Chairman
Dr. Jerry R. Kline
Mr. Frederick J. Shon

In the Matter of **Docket Nos. 50-440-OL**
50-441-OL

CLEVELAND ELECTRIC ILLUMINATING COMPANY, *et al.*
(Perry Nuclear Power Plant, Units 1 & 2)

December 13, 1982

The Licensing Board declines to reconsider its earlier decision admitting a hydrogen control contention and reaffirms its earlier decision that petitioners have sufficient basis for the admission of this contention.

RULES OF PRACTICE: MOTIONS FOR RECONSIDERATION

Motions for reconsideration ordinarily must be filed within ten days of a Board decision. Thereafter the Board decision becomes the law of the case, subject to untimely reconsideration only upon demonstration of good cause for late filing.

NRC Control and Overview of the Construction of the Callaway Facility

During the construction phase of the Callaway Plant, the Union Electric Company's responsibility under the Nuclear Regulatory Commission's (NRC's) rules and regulations encompassed control of the principal architectural and engineering criteria required to ensure radiological health and safety. With regard to the steps taken by the NRC to ascertain compliance of Union Electric with its responsibility, the NRC conducted an extensive on-site inspection program during construction which includes a resident inspector as well as inspectors from our regional office in Glen Ellyn, Illinois (Region III). These inspectors are qualified in specialized areas such as electrical and mechanical construction. This regional inspection effort extended from 1976 through 1984 and involved about 50 separate inspectors expending about 21,000 hours during Callaway's construction.

In addition, the NRC conducted two special inspections towards the end of the construction phase at Callaway to determine whether Union Electric had exercised proper construction control. The first of these was a Region III Construction Assessment Team (CAT) inspection conducted in April and May of 1982 with a six-person team over a two-week period. During this review, the Region III team evaluated the quality of the construction itself. The NRC summarized the conclusions of this CAT inspection in a report issued on June 15, 1982. The CAT identified no construction problems and made an overall finding that Union Electric's Quality Assurance program was adequate.

In November and December of 1982, the second of these specialized inspections, an Integrated Design Inspection (IDI), was conducted by a 10-person team from the NRC's Office of Inspection and Enforcement, expending about 1600 hours. The objective of this effort was to expand the NRC's review of quality assurance into the design process, including an examination of the as-built configuration. The team reviewed several hundred design elements. The NRC summarized the conclusions of this IDI inspection in a report issued on April 4, 1983. The basic finding of this team was that the overall design process was well controlled by Union Electric and its contractors and vendors. The NRC considers this design control by Union Electric to be a measure of the overall construction control.

The NRC completed the inspection monitoring of the Callaway construction when Region III sent the Office of Nuclear Reactor Regulation its report on the status of the construction and testing of the facility. In this report, "Status of Plant Readiness for an Operating License" (IE MC 94300), the NRC included the following: (1) a listing of any uncompleted items or enforcement actions; (2) the status of the preoperational testing and an assessment of the operational preparedness; (3) the status of construction; (4) the status of NRC inspections not yet completed; and (5) an estimate of when the preceding items would be completed. Callaway's operating license was not issued until all items identified in this report were satisfactorily addressed.

ENCLOSURE 5

Further construction control at Callaway was provided by Union Electric's quality assurance program and a multitiered set of additional quality assurance programs, including that of the Daniel International Corporation and extending down to the level of each subcontractor on the construction site. This control further extended to the manufacturing facilities of each vendor supplying safety-related equipment. As provided in Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which consists of those quality assurance actions related to the physical characteristics of a material, structure, component, or system that provide a means to control quality. A construction permit holder may delegate to others, such as contractors, agents or consultants, the work of executing the quality assurance program, or any part thereof, but retains responsibility for the program's execution. The NRC reviewed and audited the execution of these quality assurance programs before and during construction of the Callaway facility.

The quality of the construction of Callaway was examined in litigated proceedings before the issuance of an operating license, and it was found that Union Electric properly controlled the quality of construction so as to allow the issuance of an operating license. Union Electric Company (Callaway Plant, Unit 1), LBP-82-109, 16 NRC 1826, 1864 (1982), affirmed, ALAB-740, 18 NRC 343 (1983).

It is the NRC's position that the integrity of construction can be ensured through carefully executed quality assurance programs, including tests and inspections to verify that a nuclear power plant is constructed in accordance with all applicable requirements. In addition, each plant has an intensive preoperational start-up test program that must be successfully completed by the applicant and then reviewed and approved by the NRC before issuance of an operating license. This preoperational test program at Callaway further verified that the plant, as built, would not adversely affect the public's radiological health and safety.