SECTION 1

INTRODUCTION AND SUMMARY

TABLE OF CONTENTS

TABLE OF CONTENTS (CONT'D)

TABLE OF CONTENTS (CONT'D)

TABLE OF CONTENTS (CONT'D)

TABLE OF CONTENTS (CONT'D)

* Note: Subsections 1.3.3.3 and 1.3.3.5 do not exist in the FSAR.
Section 1.1.9 deleted by Updated FSAR, Revision 0.

LIST OF TABLES

LIST OF TABLES (CONT'D)

LIST OF FIGURES

SECTION 1

INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Original Final Safety Analysis Report was submitted in support of an application to operate the "Beaver Valley Power Station, Unit No. 1" (BVPS-1) on a site in Shippingport Borough on the Ohio River in Beaver County, Pennsylvania. The United States Nuclear Regulatory Commission, (NRC), subsequently issued Facility Operating License No. DPR-66.

In July 1980, the NRC amended its regulations to require the licensees to periodically revise the Final Safety Analysis Report, (FSAR). The purpose of periodic revisions is to provide an updated reference document to be used in recurring safety analyses performed by the licensees, the NRC, and other interested parties.(3) The Original FSAR, as amended, is still considered to be the licensing basis for the plant. The Original FSAR and the docket file (Docket No. 50-334) is the final authority if a discrepancy exists, although this updated reference document, which will be referred to as the Updated FSAR, will provide the most convenient reference. The Technical Specifications to the Facility Operating License will henceforth reference the Updated FSAR.(1)

The Updated FSAR is therefore submitted in accordance with 10 CFR 50.71(e) to assure that the information submitted to the NRC is the latest material developed. The Updated FSAR contains all the changes necessary to reflect information and analyses submitted to the Commission by BVPS-1 or prepared by BVPS-1 pursuant to the Commissions requirement since the submission of the Original FSAR; or, as appropriate, the latest revision of the Updated FSAR. The Updated FSAR includes the effects of: all changes made in BVPS-1 or programs/procedures as described in the Original FSAR; all safety evaluations performed by BVPS-1, either in support of approved license amendments, or in support of conclusions that changes did not require a license amendment in accordance with 10 CFR 50.59(c)(2); and all analyses of new safety issues performed by or on behalf of BVPS-1 at the Commissions request. These requirements are cited from 10 CFR 50.71(e).

References to the previous owner FirstEnergy (FE) and previous licensee, FirstEnergy Nuclear Operating Company (FENOC), have been retained, where appropriate, for historical purposes.

Table 1.1-1 provides a list of specific guidelines utilized in updating the FSAR to meet the requirements of 10 CFR 50.71(e) as modified by Reference 1.

BVPS-1 includes a pressurized water reactor Nuclear Steam Supply System and turbine generator furnished by Westinghouse Electric Corporation. The balance of the unit was designed and constructed by the Licensee, with the assistance of their agent, Stone & Webster Engineering Corporation (S&W).

The Nuclear Steam Supply System (NSSS) was originally designed for a warranted power output of 2,660 MWt, which was the license application rating, with an equivalent unit net electrical output of ~835 MWe. The NSSS output of 2,660 MWt resulted from a core power (i.e., rated thermal power) of 2,652 MWt and 8 MWt from the reactor coolant pumps. The NSSS design was based upon an expected ultimate output of approximately 2,774 MWt. The NSSS output of 2,774 MWt resulted from a core power of 2,766 MWt and 8 MWt from the reactor

coolant pumps. All safety systems, including containment and engineered safety features, were originally designed and evaluated for operation at the higher power level.

The initial fuel load commenced in February, 1976 and commercial operation was achieved in September, 1976.

The core power level (i.e., rated thermal power) was increased in Fall 2001 to 2,689 MWt, taking advantage of the feedwater flow Measurement Uncertainty Recapture (MUR). The corresponding NSSS thermal power level was 2,697 MWt, which included 8 MWt from the reactor coolant pumps. The licensed core power level was subsequently increased to 2,900 MWt in 2006. The corresponding NSSS thermal power level is 2,910 MWt, which includes 10 MWt of heat from non-reactor sources (primarily reactor coolant pump heat). Analyses and engineering evaluations, as appropriate, at these increased thermal power levels were performed in the areas of thermal-hydraulic and nuclear characteristics of the reactor core, postulated accidents, and plant systems and components. The corresponding uprated gross electrical output is 1014 MWe.

The Updated FSAR consists of 15 sections and two appendices. The contents are briefly described below.

Section 1 of this report summarizes the principal design features and safety criteria of the unit, emphasizing the similarities and differences with respect to other pressurized water nuclear power stations employing essentially the same technology and basic engineering features as BVPS-1. Appendix IA of Section 1 provides a discussion of BVPS-1's degree of conformance to the General Design Criteria (GDC) Published as Appendix A to 10 CFR 50 in July 1971. Section 2 contains a description and evaluation of BVPS-1 site and environs, prepared at the time of Plant Licensing, and supports the suitability of the site for a reactor of the size and type described. Sections 3 and 4 describe the reactor and the reactor coolant system. Section 5 describes the containment structure and related systems. Sections 6 through 11 describe the other auxiliary systems. Sections 5, 6, 7, 8, and 9 include descriptions of the various systems directly related to safety. Section 12 addresses conduct of operations, including organization and personnel training. Section 13 describes the initial tests and operation. Section 14 relates to safety evaluation and summarizes the analyses which demonstrate the adequacy of the reactor protection system, the containment, and engineered safety features. Section 14 demonstrates that the consequences of various postulated accidents are within the guidelines suggested in the Federal Regulation 10 CFR 100 or 10 CFR 50.67, as applicable. Section 15, Technical Specifications and Bases has been deleted from the Updated FSAR since this section has been superceded by the BVPS-1, Technical Specifications, Appendix A to Operating License No. DPR-66. The Technical Specifications, gives safety limits, limiting safety system setting, limiting conditions for operation, surveillance requirements, design features and administrative controls for the station.

Appendix A describes the Quality Assurance program that was followed during all stages of station design and construction. The operations phase quality assurance program is presented in the company Quality Assurance Program Manual. Appendix B describes the seismic design and analysis of structures, systems and components.

The AEC Regulatory Staff Questions and Positions raised during the AEC's review of the FSAR have been incorporated into appropriate sections of the Updated FSAR.

With respect to the numbers, graphs and drawings included within this report, it should be understood that normal tolerance permitted by good engineering practice is intended. Where operating parameters are unusually important, it is acknowledged that such items are included in the "Technical Specifications"; the adoption of which is a condition of the BVPS-1 Operating License. The engineering drawings in the Updated FSAR are not intended to accurately depict other than safety related equipment and systems. In any case, the latest revision of the appropriate approved and released engineering drawings should be consulted for the most current information, as the Updated FSAR is only current as of the date of submittal.

1.1.1 Design Highlights

The design of BVPS-1 was based upon proven concepts which were developed and successfully applied in the construction of other pressurized water reactor systems. In subsequent paragraphs, certain design features of BVPS-1 are indicated which represented slight variation or extrapolations from units which were approved for construction or operation such as the Surry Power Station Units 1 and 2 (Dockets 50-280 and 50-281), North Anna Power Station Units 1 and 2 (Dockets 50-338 and 50-339), Turkey Point Plant Units 3 and 4 (Dockets 50-250 and 50-251), and H. B. Robinson No. 2 (Docket 50-261). The comparison of BVPS-1 with other licensed plants was considered valid at the time of the issuance of the BVPS-1 Operating License.

1.1.2 Power Level

The nominal NSSS power level for BVPS-1 is set at 2,910 MWt. Site and engineered safety evaluation is performed at the power levels specified in Section 14 for each particular accident. The maximum calculated rating of the turbine generator corresponds to the NSSS power level. The NSSS power level is achieved by about a 9.4 percent increase in average reactor heat flux over that established for initial operation.

1.1.3 Reactor Coolant Loops

The reactor coolant system consists of three loops, each loop having components (steam generator, pumps, and piping) similar to those for the Surry Power Station Units 1 and 2, and including two reactor coolant loop stop valves and a bypass valve in each loop.

1.1.4 Peak Specific Power

The reactor core design is based on a maximum steady state peak specific power of 14.3 kw per ft (based on a Heat Flux Hot Channel Factor F(Q) of 2.52) for operation at 2,900 MWt (2,910 MWt NSSS power level) and a corresponding peak power of 22.4 kw per ft for the maximum thermal overpower condition.

1.1.5 Fuel Clad

The fuel rod design for the reactor uses zirconium alloy clad material. Zirconium alloy clad material has proven successful in the CVTR and Saxton reactors and Yankee test assemblies, and is used in most Westinghouse reactors now in operation.(2)

1.1.6 Fuel Assembly Design

The fuel assembly incorporates the rod cluster control concept in a canless 17 x 17 fuel and control rod array using Inconel or zirconium alloy spring clip grids to provide support for the fuel rods as used in the 15 x 15 assembly design. Extensive out-of-pile tests have been performed on this concept, successful in-pile tests have been performed in the Saxton reactor, and operating experience is available from the San Onofre and Connecticut-Yankee plants and other Westinghouse designed NSSS plants.

1.1.7 Moderator Temperature Coefficient of Reactivity

Burnable absorbers are used in the reactor core to ensure the moderator temperature coefficient of reactivity remains within the limits discussed in Section 3.3.2.3.2. As the fuel in the core is depleted and the boron shim concentration is decreased, the moderator temperature coefficient of reactivity becomes more negative.

1.1.8 Containment

The original reactor containment design concept was based upon the use of a reinforced concrete containment structure, similar to that of the Surry Power Station Units 1 and 2 and North Anna Power Station Units 1 and 2. The containment was maintained at subatmospheric pressure during normal operation. Following the postulated loss-of-coolant accident (LOCA), described in Section 14.3, and the Main Steam Line Break (MSLB) accident, described in Section 14.2, the containment peak pressure was reduced to subatmospheric by the use of the containment depressurization system, which contains redundant spray cooling systems, thereby positively terminating outleakage to the environment within 60 minutes after initiation of the accident. In 2006, containment operating conditions were changed from sub-atmospheric to atmospheric. Included in this change to the design basis was the elimination of the design requirement to terminate containment atmosphere leakage within 60 minutes following a LOCA.

The design of the containment structure is based on the following criteria:

1. The peak calculated containment atmosphere pressure shall not exceed the design pressure of 45 psig.

2. The containment pressure shall be reduced to less then 50% of the peak calculated pressure for the design basis loss-of-coolant accident within 24 hours after the postulated accident.

Radioactivity is assumed to leak to the environment at the containment Technical Specification leak rate for the first day, and half that leakage rate for the remaining duration of the accident (i.e., 29 days). No credit is taken for filtering the containment leakage via the safety related ventilation exhaust and filtration system that services the areas contiguous to containment; i.e., the Supplementary Leak Collection System (SLCRS) filters. To ensure bounding values, the atmospheric dispersion factors utilized for the containment release path reflects the worst value between the containment wall release point and the SLCRS release point for each time period.

1.1.9 Xenon Oscillations

This section has been deleted by Revision 0.

1.1.10 Conclusions

The pressurized water reactor unit of the type used at BVPS-1 possesses inherent stability characteristics and safety features. These are typified by the Doppler coefficient effect in the fuel and the barriers against fission product release. Engineered safety features systems, such as the safety injection system and containment depressurization system, complement these characteristics.

The Licensee's submit that the BVPS-1 can be operated without hazard to the general public, because of the engineered safety features described herein and the analyses that ensure adequacy and conformance to NRC criteria and other regulatory requirements.

References to Section 1.1

1. "Periodic Updating of Final Safety Analysis Report (FSAR's)", D. Eisenhut, U.S. Nuclear Regulatory Commission Letter (December 15, 1980).

2. J. Skaritka, J. A. Iorii, "Operational Experience with Westinghouse Cores," WCAP-8183, Revision 10, Westinghouse Electric Corporation (May 1981).

3. "Periodic Updating of Final Safety Analysis Reports", U. S. Nuclear Regulatory Commission, Federal Register, Vol. 45, No. 92, Rules and Regulations, p. 30614 (Friday, May 9, 1980).

1.2 SUMMARY - STATION DESCRIPTION

1.2.1 General

BVPS-1 incorporates a closed-cycle pressurized water Nuclear Steam Supply System (NSSS), a turbine generator and their necessary auxiliaries. A radioactive waste disposal system, a fuel handling system, and all auxiliaries, structure and other onsite facilities required for a complete and operable nuclear power station are also provided. The general arrangement of BVPS-1 and the station arrangement are shown in the site plan, Figure 1.2-1.

Flow diagrams are included with the systems which are described throughout the Updated FSAR. Symbols and abbreviations used in these diagrams are illustrated in Figure 1.2-3.

Some equations in the FSAR are written in a modified FORTRAN type format. A description of this format is given in Table 1.2-1.

1.2.2 Site

The site comprising approximately 453 acres is located on the south bank of the Ohio River, in Beaver County, approximately 25 miles northwest of Pittsburgh. The exclusion radius is 2,000 ft. The nearest continuously occupied residence is located about 2,100 ft from the reactor. The Low Population Zone area distance is 3.6 miles. The population center distance is about five miles. The area is primarily agricultural, with some industrial activity.

1.2.3 Structures

The major structures are the containment structure, cooling tower, intake structure, auxiliary building, fuel building, decontamination building, turbine building, diesel generator building and service building which includes the main control area.

The containment structure is a steel-lined, reinforced concrete cylinder with a hemispherical dome and a flat reinforced concrete foundation mat. The containment which is designed to withstand the internal pressure resulting from the Design Basis Accident (DBA), meets all requirements for leak tightness at this pressure and provides adequate radiation shielding for both normal operation and accident conditions.

The seismic criteria used in the design of the structures and equipment for the station are described in Section 2.7.

1.2.4 Nuclear Steam Supply System

The NSSS consists of a pressurized water reactor, reactor coolant system and associated auxiliary systems. The reactor coolant system is arranged as three closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump, isolation and bypass valves, piping and a steam generator. An electrically heated pressurizer is connected to one of the loops on the reactor side of the loop isolation valves.

The reactor core includes uranium dioxide pellets, enclosed in Zircaloy tubes with welded end plugs, as fuel. The tubes are supported in assemblies by structures of spring clip grids and suitable end pieces for the support of the assembled rods and restraint of abnormal axial movement. The mechanical control rods consist of clusters of stainless steel clad absorber rods which are guided by tubes located within the fuel assembly. The core consists of these fuel assemblies, loaded in three different enrichment regions. New fuel is introduced into the outer region, and is moved inward in a preset pattern as determined by the reactor manufacturer.

The steam generators are vertical U-tube units containing nickel-chromium-iron (NI-Cr-Fe) Alloy 690 tubes. Integral separating equipment reduces the moisture content of the steam at the steam generator outlet to 0.10 percent or less.

The reactor coolant pumps are vertical, single stage, centrifugal pumps equipped with controlled leakage shaft seals.

The reactor coolant loop stop and bypass valves are motor operated gate valves, remotely controlled from the main control room and permit any loop to be isolated from the reactor vessel.

Nuclear auxiliary systems are provided to perform the following functions:

1. Accommodate reactor coolant system water requirements

2. Purify reactor coolant water

3. Introduce chemicals for corrosion inhibition

4. Introduce and remove chemicals for reactivity control

5. Cool system components

6. Remove residual heat during the reactor cooling period, and also when the reactor is shut down

7. Cool the spent fuel pool water

8. Permit sampling of reactor coolant water

9. Provide for emergency safety injection

10. Vent and drain the reactor coolant system and the auxiliary systems

11. Provide emergency containment spray

12. Provide containment ventilation and cooling

13. Dispose of liquid and gaseous wastes, and provide for disposal of solid wastes.

1.2.5 Reactor and Station Controls

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rod assemblies. The control system permits the unit to accept step load increases of 10 percent and ramp load increases of 5 percent per minute over a load range of 15 percent to, but not exceeding, 100 percent power under normal operating conditions subject to xenon limitations.

Control of both the reactor and turbine generator is accomplished from the main control room as supervised by NRC licensed personnel.

1.2.6 Waste Disposal Systems

The waste disposal systems provide all equipment necessary to collect, process and prepare for disposal of radioactive liquid, gaseous and solid wastes produced as a result of station operation.

Radioactive liquid wastes are collected for disposal. After sampling, some liquids are processed through an ion exchanger. The water is analyzed before discharge to the river through the cooling tower blowdown to ensure concentrations are below appropriate limits established by: 10 CFR 20 and the guidelines of Appendix I to 10 CFR 50; the Offsite Dose Calculation Manual; radiation protection procedures; and applicable regulations. Evaporator residues and noncombustible solid wastes are placed in disposable containers, and combustible solid wastes are baled or packaged for shipment to an authorized offsite disposal location.

Nonaerated gaseous wastes are collected, passed through charcoal beds to holdup radioactive isotopes and a fraction stored until radioactivity level permits discharge to the environment, after air dilution, at concentrations below the limits set forth in: 10 CFR 20 and the guidelines of Appendix I to 10 CFR 50; and the Offsite Dose Calculation Manual and radiation protection procedures. Aerated gaseous wastes are filtered in charcoal and HEPA filters.

1.2.7 Fuel Handling Systems

The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for on-site dry storage or for shipment from the site. NOTE: Throughout this document “shipping cask” is used interchangeably with “storage cask.” Underwater transfer of spent fuel enables the use of an optically transparent radiation shield, and provides a reliable source of coolant for removal of residual heat.

1.2.8 Turbine and Auxiliary Systems

The turbine is a tandem-compound, four-flow, 1,800 rpm unit. Four combination moisture separator-reheaters are employed to dry and superheat the steam between the high and low pressure turbines.

The condensate-feedwater cycle is composed of the following equipment:

1. A single-pass, deaerating, surface condenser installed in two sections
2. Two 100 percent design capacity steam jet air ejectors (40 SCFM each)
3. Two 50 percent design capacity condensate pumps (9,700 gpm each)
4. Two 50 percent design capacity steam generator feedpumps (15,200 gpm each)
5. Two 100 percent capacity heater drain pumps (8,360 gpm each)
6. Six stages of feedwater heating gland steam condenser
7. Drain coolers.

Additional auxiliary systems are installed for the reliable operation of BVPS-1 and include compressed air, component cooling water, circulating water, cooling tower, vacuum priming, lubricating oil, secondary water sample, chemical feed, blowdown, hydrogen and $C0_2$, gland steam and water treatment.

1.2.9 Electrical Systems

The main generator is a 1,800 rpm, 22 kv, 3 phase, 60 cycle, hydrogen inner-cooled unit. One main step-up transformer is provided to deliver power to the 345 kv switchyard.

The BVPS-1 service system consists of auxiliary transformers, 4,160 v and 480 v switchgear and buses, 480 v motor control centers, 120 v a-c vital buses, and 125 v d-c batteries and equipment and their distribution systems. The normal source of station service power is obtained from the main generator, the high voltage switchyard, or a combination of both.

Two diesel engine-driven generators are provided to supply emergency power in the event of complete loss of normal and reserve station service power. Each emergency diesel generator has sufficient capacity for operation of all engineered safety features equipment which must be operated in the event of a loss- of-coolant accident.

Adequate independency and physical separation is maintained between redundant power supplies and their distribution systems.

The BVPS-1 electrical system is designed in accordance with General Design Criteria 17, "Electric Power Systems", General Design Criteria 18, "Inspection and Testing of Electric Power Systems", Safety Guide 6 "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems", Safety Guide 9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies" and the Institute of Electrical and Electronic Engineers, IEEE Std. 308,[1] and IEEE Std. 344.[2]

A complete description of the electrical system is presented in Section 8.

1.2.10 Engineered Safety Features Systems

The engineered safety features systems have sufficient redundancy and independence of components and power sources, such that, under the conditions of the DBA, the systems can, even when operating with partial effectiveness, maintain the integrity of the containment and hold the exposure to the public within the criteria provided in 10 CFR 50.67.

The systems provided are summarized below:

1. The steel-lined concrete containment structure provides a highly reliable barrier against the escape of radioactivity when the containment is above atmospheric pressures. The structure and all penetrations, including access openings and ventilation ducts, are of proven design.
2. The emergency core cooling system cools the core by injecting borated water into the reactor coolant loops from the accumulators and the safety injection pumps in major loss-of-coolant accidents.
3. The quench spray and recirculation spray subsystems of the containment depressurization system provide sprays of borated water to the containment atmosphere. Following the DBA, the containment pressure is rapidly reduced by the containment depressurization system, thereby reducing leakage to the atmosphere. Subsequent long-term maintenance of atmospheric conditions is accomplished by the recirculation spray subsystems.
4. The supplementary leak collection and release system (SLCRS) ensures that radioactive leakage from the containment penetrations following a DBA, or radioactive material released in the waste gas storage area is collected, filtered and discharged to the atmosphere via the SLCRS vent. Note that the site boundary and control room dose analyses do not credit the filtration capability of the SLCRS.

1.2.11 Independent Spent Fuel Storage Installation (ISFSI) Facility

BVPS has implemented an on-site ISFSI facility that is used for storage of spent nuclear fuel. The ISFSI site is located within the site Protected Area. The ISFSI utilizes the AREVA/ Transnuclear Spent Fuel Dry Storage system. The ISFSI facility is composed of several components. The main components are the horizontal storage modules (HSM), the HSM concrete support pads, a concrete apron between storage pads, a heavy haul path on which the transporter delivers spent fuel canisters, drainage and electrical systems. The concrete support pads consist of two identical concrete pads that provide storage capacity for a total of 60 HSMs. The HSMs will be arranged in a single row of 30 placed on each pad. The pads are separated by a concrete apron that is part of the heavy haul path. Three foot thick concrete shield walls are placed at the end and rear of each row of HSMs.

References to Section 1.2

1. "IEEE Criteria for Class 1E Power Systems for Nuclear Power Generating Stations", IEEE Std. 308, the Institute of Electrical and Electronic Engineers, Inc.

2. "IEEE Recommended Practice for Seismic Qualification of Class 1E Power and Protection Systems", IEEE Std. 344, the Institute of Electrical and Electronic Engineers, Inc.

1.3 SINGLE FAILURE CRITERION, GENERAL DESIGN CRITERIA, AND SAFETY GUIDES

1.3.1 Single Failure Criterion

All features of BVPS-1 related to public safety are designed to the single failure criterion, defined as follows in Appendix A to 10 CFR 50:

> A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions.
>
> Single failures of passive components in electrical systems should be assumed in designing against a single failure.

With respect to fluid systems, the single failure criterion applies only to safety related fluid systems. These fluid systems are designed to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), they are able to accomplish their safety functions assuming a single failure occurs within the system.

The classes of fluid systems to which the single failure criterion applies, as listed in Appendix A to 10 CFR 50, are (with the appropriate 1971 General Design Criterion (GDC) in parentheses):

1.	Residual Heat Removal	(GDC-34)
2.	Emergency Core Cooling	(GDC-35)
3.	Containment Heat Removal	(GDC-38)
4.	Containment Atmosphere Cleanup	(GDC-41)
5.	Cooling Water	(GDC-44)
6.	Containment Isolation	(GDC-55, 56, & 57)

The onsite electrical power sources, including batteries, and the onsite electrical distribution system are also designed to perform their safety functions assuming a single failure. The protection system, as defined in 1971 GDC-20, is designed so that no single failure results in loss of its protection function. Fluid systems other than those listed above may also be subject to the single failure criterion, if they have a safety function.

Each of the above safety related fluid systems or combination of fluid systems are designed to accept only a single active failure in the short term, or to accept a single active or passive failure in the long term following any accident condition. These systems are not necessarily designed for a passive failure in the short term.

The short term is the 24 hours immediately following an accident during which automatic actions are performed, system responses are checked, type of accident is determined and preparation for long term recovery operations are made. This 24 hour criterion applies to each of the above safety related fluid systems or combination of fluid systems.

The long term is the remainder of the recovery period following the short term. The long term period involves bringing the BVPS-1 to a condition where access to the containment or other buildings where the major accident has occurred can be gained and repairs effected.

An active failure is the failure of a powered component, such as a piece of mechanical equipment, component of the electrical supply system or instrumentation and control system, to act on command to perform its design function. Examples include the failure of a valve to move to its correct position, the failure of an electrical breaker or relay to respond and the failure of a pump, fan, or diesel generator to start. Systems are designed to minimize the effect of equipment such as a motor operated valve, moving spuriously from the proper safety features position.

A passive failure is the failure of a static component which limits that component's effectiveness in carrying out its design function. Examples of a passive failure are the rupture of a pipe or a valve body, or the breaking of a cable in an electrical system. A passive failure in a fluid system is considered to occur only in the long term. The failure of a passive component is not considered in the design of fluid systems when it can be demonstrated that the design is accepted on some other defined basis, such as an unusually high quality, high strength or low stress, inspectability, repairability or short term use. Valves are not assumed to fail so as to result in two open pipes.

When a passive failure in an electrical or protection system causes an active failure in a fluid system it is considered an active failure. For example, the breaking of a cable in an electrical system is a passive failure; however, it may result in the failure of a pump to start, which is an active failure.

1.3.2 General Design Criteria

BVPS-1 has been designed and constructed to comply with the "General Design Criteria for Nuclear Power Plant Construction" published in July, 1967 by the AEC. Each criterion applicable to BVPS-1 is followed by a summary discussion of the design and procedures which are intended to meet the design objectives reflected in the criterion. Since the BVPS-1 construction permit was issued in June 1970, the compliance to the AEC General Design Criteria of July, 1967 is addressed. Appendix IA provides a discussion of BVPS-1's degree of conformance to the AEC General Design Criteria published as Appendix A to 10 CFR 50 in July 1971. Modifications made to the plant satisfy the 1967 and 1971 General Design Criteria as discussed in Section 1.3.2 and Appendix 1A, respectively.

1.3.2.1 Overall Station Requirements

Criterion 1 - Quality Standards (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to ensure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Answer

The structures, systems and components of the station are classified according to their importance in the prevention and mitigation of accidents which could cause undue risk to the health and safety of the public. These classifications are described in Appendix A. A discussion of the codes and standards, quality assurance programs, test provisions, etc., applying to each system is included or referenced in that portion of the FSAR describing that system.

Criterion 2 - Performance Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect:

1. Appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area

2. An appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Answer

The structures, systems and components designated Seismic Category I are designed to withstand, without loss of capability to protect the public, the most severe environmental phenomena ever experienced at the site with appropriate margins included in the design for uncertainties in historical data. Potential environmental hazards are discussed and analyzed in Sections 2 and 14 of the FSAR and the influence of these hazards on various aspects of the station design is discussed in the sections covering the specific systems and components concerned. An outline of the design philosophy for Seismic Category I structures, systems, and components is included in Appendix B.

Criterion 3 - Fire Protection

The reactor facility shall be designed to:

1. Minimize the probability of events such as fires and explosions

2. Minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Answer

Through the use of noncombustible and fire resistant materials wherever practical in the facility and the limitation of combustible supplies (e.g., logs, records, manuals, etc.) in such areas as the main control room to amounts required for operation, the probability of such events as fire and explosion and the effects of such events should they occur are minimized. Fire protection criteria and specific means of meeting these criteria are described in Sections 7.8 and 9.10.

Criterion 4 - Sharing of Systems

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Answer

Systems to be shared between BVPS-1 and BVPS-2 are provided in Section 1.7 of the Updated FSAR.

Criterion 5 - Records Requirements

Records of the design, fabrication, and construction of essential components of the station shall be maintained by the reactor operator or under its control throughout the life of the reactor.

Answer

BVPS-1 intends to maintain in its possession or under its control, a complete set of records of the design, fabrication, construction and testing of major Seismic Category I components throughout the life of BVPS-1. Section 12 of the Updated FSAR presents records requirements for: station operation, maintenance, and modification; and review of procedures.

1.3.2.2 Protection by Multiple Fission Product Barriers

Criterion 6 - Reactor Core Design

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine-generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Answer

The reactor core with its related control and protection system is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater and loss of all offsite power.

The reactor control and protection instrumentation is designed to actuate a reactor trip for any anticipated combination of plant conditions when necessary to ensure a minimum Departure from Nucleate Boiling Ratio (DNBR) equal to or greater than the design limit and fuel center temperatures below the melting point of UO_2.

Section 3 discusses the design bases and design evaluation of reactor components. The details of the control and protection systems instrumentation design and logic are discussed in Section 7. This information supports the safety analysis presented in Section 14.

Criterion 7 - Suppression of Power Oscillations

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

Answer

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and limitations on the moderator temperature coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the radial, diametral and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and limitations on the moderator temperature coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided as a result of reactor trip functions using the measured axial power imbalance as an input.

Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone, are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

The stability of the core against xenon-induced power oscillations and the functional requirements of instrumentation for monitoring and measuring core power distribution are discussed in Section 3. Details of the instrumentation design and logic are discussed in Section 7.

Criterion 8 - Overall Power Coefficient

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Answer

Prompt compensatory reactivity feedback effects are ensured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the operational limits on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is ensured by the inherent design using low-enrichment fuel. The limitations on moderator temperature coefficient of reactivity are ensured by administratively controlling either the dissolved neutron absorber concentration, burnable poisons, and/or rod withdrawal limits.

These reactivity coefficients are discussed in Section 3.

Criterion 9 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Answer

The reactor coolant system boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. Section 4 has additional details.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture as discussed in Appendix B.

The system is protected from overpressure by means of pressure relieving devices as required by applicable codes.

Means are provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary with indication in the control room as discussed in Section 4.

The reactor coolant system boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leaktight integrity. The details are given in Section 4.

For the reactor vessel, a material surveillance program conforming to applicable codes is provided. Section 4 has additional details.

Criterion 10 - Containment

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Answer

The design of the containment structure and associated auxiliary systems is described in Section 5. Other engineered safety features required to limit pressure inside the containment are described in Section 6. Section 14 demonstrates the adequacy of the containment and engineered safety features under various accident conditions, including a rupture of the largest reactor coolant pipe.

1.3.2.3 Nuclear and Radiation Controls

Criterion 11 - Control Room

The facility shall be provided with a control room from which actions to maintain safe operational status of the station can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Answer

A main control room is provided which contains all controls and instrumentation necessary for operation of the reactor, turbine- generator and auxiliary and emergency systems under normal or accident conditions.

The main control room is designed and equipped to minimize the possibility of events which might preclude occupancy. In addition, provisions are made for bringing BVPS-1 to and maintaining it in a hot shutdown condition for an extended period of time from an auxiliary shutdown panel located outside the main control room.

Materials used in the construction of the main control room equipment, and furnishings, are discussed in Section 7.8. Section 9.13 discusses the main control room ventilation system. Section 11.3 discusses main control room shielding.

Criterion 12 - Instrumentation and Control Systems

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Answer

Plant instrumentation and control systems are provided to monitor significant variables in the reactor core, coolant systems, and reactor containment building over their anticipated range for all conditions as appropriate to assure adequate safety. The installed instrumentation provides continuous monitoring, warning and initiation of safety functions.

The following processes are controlled to maintain key variables within their normal ranges:

1. Reactor power level (controlled manually or automatically by controlling thermal load)

2. Reactor coolant temperature (controlled manually or automatically by rod cluster control assembly (RCCA) motion, in sequential groups)

3. Reactor coolant pressure (controlled manually or automatically by heaters and spray in the pressurizer)

4. Reactor coolant water inventory, as indicated by the water level in the pressurizer (controlled manually or automatically by charging flow)

5. Reactor coolant system boron concentration (controlled manually or automatically by make-up of charging flow)

6. Steam generator water inventory on secondary side (controlled manually or automatically by feedpump flow through feedwater control valves).

The reactor control system is designed to automatically maintain a programmed average temperature in the reactor coolant system during steady state operation and to ensure that plant conditions do not reach reactor trip settings as the result of a transient caused by a design load change.

The reactor protection system trip setpoints are selected so that anticipated transients do not cause a DNBR of less than the design limit.

Proper positioning of the control rods is monitored in the control room by bank arrangements of individual meters for each RCCA. A rod deviation alarm alerts the operator of deviation of one RCCA from its bank position. There are also insertion limit monitors with visual and audible annunciation to avoid loss of shutdown margin. Each RCCA is provided with a sensor to detect positioning at the bottom of its travel. This condition is also alarmed in the control room. Four excore ion chambers also detect asymmetrical flux distribution indicative of rod misalignment. Movable in-core flux detectors and fixed in-core thermocouples are provided as operational aids to the operator.

Overall reactivity control is achieved by the combination of soluble boron and RCCA. Long term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short term reactivity control for power changes is accomplished by the reactor control system which automatically moves RCCA. This system uses input signals including neutron flux, coolant temperature, and turbine load.

Containment pressure, pressurizer water level, pressurizer pressure and steam line pressure are monitored to sense accident conditions. Section 7 contains further details of instrumentation and controls.

Criterion 13 - Fission Process Monitors and Controls

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Answer

The nuclear instrumentation system described in Section 7 is provided to monitor the reactor power from source range through the intermediate range and power range up to 120 percent of full power. The system provides indication, control and alarm signals for reactor operation and protection.

The operational status of the reactor is monitored from the control room. When the reactor is subcritical, the relative reactivity status is continuously monitored and indicated by proportional counters located in instrument wells in the neutron shield tank adjacent to the reactor vessel. Two source range detector channels are provided for supplying information on multiplication while the reactor is subcritical. A reactor trip is actuated from either channel if the neutron flux level becomes excessive.

When the reactor is critical, means for showing the relative reactivity status of the reactor is provided by control bank positions displayed in the control room. The position of the control banks is directly related to the reactivity status of the reactor when at power and any unexpected change in the position of the control banks under automatic control or change in the coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. Periodic evaluation of boric acid concentration provides a long term means of following reactivity status.

Criterion 14 - Core Protection Systems

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Answer

The operational limits for the core protection system are defined by analyses of all plant operation and fault conditions requiring rapid rod insertion to prevent or limit core damage. With respect to acceptable fuel design limits, the protection system design bases for all anticipated transients or faults are:

1. Minimum departure from nucleate boiling ratio (DNBR) shall not be less than 1.21 (design limit).

2. Clad strain on the fuel element shall not exceed 1 percent

3. No center melt shall occur in the fuel elements.

A region of permissible core operation may be defined in terms of power, axial power distribution, and coolant flow and temperature. The protection system monitors these process variables (as well as many other process and plant variables). If the region limits are approached during operation, the protection system will automatically actuate alarms, initiate load cutback, prevent control rod withdrawal and/or trip the reactor.

Operation within the permissible region and complete core protection is ensured by the Overtemperature ΔT and Overpower ΔT reactor trips in the system pressure range defined by the Pressurizer High Pressure and Pressurizer Low Pressure reactor trips, provided that the transient is slow with respect to piping delays from the core to the temperature sensors. High Nuclear Flux and Low Coolant Flow reactor trips provide core protection in the event that a transient which is faster than the ΔT responses occurs. Finally, thermal transients are anticipated and avoided by reactor trips actuated by turbine trip and primary coolant pump motor low frequency or low voltage on 2 out of 3 buses.

The protection system trips the reactor by interrupting power to the rod control power supply. All full length control and shutdown rods insert by gravity as a result. The Westinghouse protection system meets the requirements of IEEE Std. 279-1971.[2]

The protection system measures a wide spectrum of process variables and plant conditions. All analog channels which actuate reactor trip, rod stop and permissive functions are indicated or recorded.

In addition, visual and/or audible alarms are actuated for reactor trip, partial reactor trip: (any input channel), and any control variable exceeding its setpoint, (any input channel). These measurements and indications provide information upon which corrective actions to prevent the development of abnormal conditions can be based. Automatic actuation of reactor trips will be performed if abnormal conditions reach the plant operational limits.

Another important safety function of the reactor protection system is that of processing signals used for engineered safety features actuation and generation of the actuation demand. The conditions leading to engineered safety features actuation are:

1. Low pressurizer pressure
2. High containment pressure
3. High-High containment pressure
4. Low steamline pressure
5. Manual.

Engineered safety features are discussed further in Section 6 and Section 7.

Criterion 15 - Engineered Safety Feature Protection Systems

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Answer

An important safety function of the reactor protection systems is that of processing signals used for engineered safety features actuation and generation of the actuation demand. The conditions leading to engineered safety features actuation are:

1. Low pressurizer pressure
2. High containment pressure
3. High-high containment pressure
4. Low steamline pressure
5. Manual

Engineered safety features are discussed in Sections 6 and 7.

Criterion 16 - Monitoring Reactor Coolant Pressure Boundary

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Answer

All reactor coolant system components are designed, fabricated, inspected, and tested in conformance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Section 4).

Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature monitored leak-off between double gaskets.

Leakage into the reactor containment is drained to the containment sump where the level is monitored.

Leakage is also detected by measuring the airborne activity and activity of the condensate drained from the containment air recirculation units.

Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank, and primary drain transfer tanks makes available an accurate indication of integrated leakage.

These leakage detection methods are described in detail in Section 4.

Criterion 17 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Answer

BVPS-1 contains means for monitoring the containment atmosphere, effluent discharge paths and the environs for radioactivity which could be released under any conditions. The details of the effluent discharge paths and containment monitoring methods are contained in Sections 5 and 11, while the environmental radiation monitoring program is described in Section 2.8.

Criterion 18 - Monitoring Fuel and Waste Storage

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Answer

Sufficient monitoring and alarm instrumentation is provided in waste and fuel storage areas to detect conditions which might contribute to loss of cooling for decay heat removal or abnormal radiation releases. Details of the monitoring systems are included in Sections 9 and 11.

1.3.2.4 Reliability and Testability of Protection Systems

Criterion 19 - Protection Systems Reliability

Protection systems shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed.

Answer

The protection system is designed for high functional reliability and inservice testability. The design employs redundant logic trains and measurement and equipment diversity. Sufficient redundancy is provided to enable individual end-to-end channel tests with the reactor at power. Built in semi-automatic testers provide means to test the majority of system components very rapidly. The protection system is described in Section 7.

Criterion 20 - Protection Systems Redundancy and Independence

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

Answer

As detailed in Section 7, sufficient redundancy and independence is designed into the protection systems to ensure that no single failure nor removal from service of any component or channel results in loss of the protection function. In addition, IEEE Std. 279-1971, was employed in the detailed design of the protection systems.

Criterion 21 - Single Failure Definition

Multiple failures resulting from a single event shall be treated as a single failure.

Answer

When evaluating the protection systems, the engineered safety features and their support systems, multiple failures resulting from a single event are treated as a single failure. The ability of each system to perform its function with a single failure is discussed in those sections describing the individual systems. Additional details are found in Section 7.

Criterion 22 - Separation of Protection and Control Instrumentation Systems

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Answer

The protection system is designed in accordance with IEEE Std. 279-1971, thereby meeting all requirements of this standard.

The protection system is separate and distinct from the control system. The control system is dependent on the protection system in that control signals are derived from protection system measurements where applicable. These signals are transferred to the control system by isolation amplifiers which are classified protection system components. The adequacy of system isolation has been verified by testing under conditions of all postulated credible faults. The failure or removal of any single control instrumentation system component or channel or of those common to the control instrumentation and protection circuitry leaves intact a system which satisfies the requirements of the protective system. The protection system and control systems are discussed in Section 7.

Criterion 23 - Protection Against Multiple Disability for Protection Systems

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Answer

Physical separation and electrical isolation of redundant channels and subsystems, functional diversity of subsystems and safe failure modes are employed in the design as defenses against functional failure through exposure to common causative factors. The redundant logic trains, reactor trip breakers and engineered safety features are physically separated and electrically isolated. Physically separate channel trays, conduit and penetrations are maintained upstream from the logic elements of each train. The protection system is discussed in Section 7.

Criterion 24 - Emergency Power for Protection Systems

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Answer

BVPS-1 is supplied with normal, reserve and emergency power to provide for the required functioning of the protection systems.

In the event of a loss of auxiliary power, emergency power is supplied by two emergency diesel generators as described in Section 8.

The instrumentation and controls portions of the protection systems will be supplied from the BVPS-1 batteries during the diesel start-up period (detailed in Section 8).

Criterion 25 - Demonstration of Function Operability of Protection Systems

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Answer

Each protection channel in service at power is capable of being calibrated and tripped independently by test signals to verify its operation. Individual semiautomatic testers are built in each logic train. These testers provide a very rapid means of testing the majority of protection system components with the reactor at power. Protection systems detailed are found in Section 7.

Criterion 26 - Protection Systems Fail-Safe Design

The protections systems shall be designed to fail to a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Answer

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the de-energized to trip principle so loss of power, disconnection, open channel faults and the majority of channel short circuit faults cause the channel to go into its tripped mode. Additional defenses against loss of function are discussed under Criterion 23. Additional details can be found in Section 7.

1.3.2.5 Reactivity Control

Criterion 27 - Redundancy of Reactivity Control

At least two independent reactivity control systems, preferably of different principles, shall be provided.

Answer

Two reactivity control systems are provided. These are control rods and chemical shim (boration).

The rod system can compensate for the reactivity effects of fuel/water temperature changes accompanying power level changes over the full range from full-load to no-load at the design maximum load change rate. Automatic control by the rods is, however, limited to the range of 15 percent to 100 percent of power rating for reasons unrelated to reactivity or reactor safety. The rod system can also compensate for xenon burnout reactivity transients over the allowed range of rod travel.

The boron system will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for all xenon burnout transients.

Details of the construction of the rod cluster control assembly are included in Section 3, with the operation discussed in Section 7. The means of controlling the boric acid concentration is described in Section 9.

Criterion 28 - Reactivity Hot Shutdown Capability

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Answer

The rod cluster control system is capable of making and holding the core subcritical from all operating and hot shutdown conditions sufficiently fast to prevent exceeding acceptable fuel damage limits as described in Section 14. The chemical shim control is also capable of making and holding the core subcritical, but at a slower rate, and is not employed as a means of compensating for rapid activity transients. The rod cluster control system is, therefore, used in protecting the core from such transients. Details of the operation and effectiveness of these systems are included in Sections 3 and 9.

Criterion 29 - Reactivity Shutdown Capability

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Answer

BVPS-1 is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Sections 3 and 9. Combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long term xenon decay and plant cooldown. In addition, the reactor can be made subcritical by the rod cluster control system alone for any anticipated operational transient so that fuel damage limits are not exceeded. The single highest worth control cluster is assumed to be stuck full-out upon trip for these determinations.

In the event of a loss-of-coolant accident, the safety injection system is actuated and concentrated boric acid is injected into the cold legs of the reactor coolant system. This is in addition to the boric acid content of the accumulators which is passively injected due to a decrease in system pressure. See Section 6 for further details.

Criterion 30 - Reactivity Holddown Capability

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Answer

BVPS-1 is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Sections 3 and 9. Normal reactivity shutdown capability is provided within two seconds following a trip signal by control rods. The chemical shim control system permits the necessary shutdown margin to be maintained during longterm xenon decay and plant cooldown.

Criterion 31 - Reactivity Control Systems Malfunction

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Answer

Reactor shutdown by full length rod insertion is completely independent of the normal control function, since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. The protection system is designed to limit reactivity transients so that DNBR will not be less than the design limit for any single malfunction in either reactor control system.

The analysis presented in Section 14 shows that for postulated dilution during refueling, startup, manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate reboration before the shutdown margin is lost. BVPS-1 reactivity control systems are discussed further in Section 7 and an analyses of the effects of the other possible malfunctions are discussed in Section 14. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

Criterion 32 - Maximum Reactivity Worth of Control Rods

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot:

1. Rupture the reactor coolant pressure boundary or

2. Disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Answer

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the reactor coolant system boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the withdrawal of rod cluster control assemblies (RCCA) and the dilution of the boric acid

in the reactor coolant systems are specified in the Technical Specifications for BVPS-1. The specification includes appropriate graphs that show the permissible mutual withdrawal limits and overlap of functions of the several RCCA banks as a function of power. These data on reactivity insertion rates, dilution and withdrawal limits are also discussed in Section 3. The relationship of the reactivity insertion rates to plant safety is discussed in Section 14.

Assurance of core cooling capability following accidents, such as rod ejection, steam line break, etc., is given by keeping the reactor coolant pressure boundary stresses within faulted condition limits as specified by applicable ASME Boiler and Pressure Vessel Codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of needed safety features.

1.3.2.6 Reactor Coolant Pressure Boundary

Criterion 33 - Reactor Coolant Pressure Boundary Capability

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Answer

The reactor coolant boundary is shown in Section 14 to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as rod ejection.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling group are inserted in the core at power, and these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given to:

1. The notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve

2. The state of stress of materials under static and transient loadings

3. The quality control specified for materials and component fabrication to limit flaw sizes

4. The provisions for control over service temperature and irradiation effects which may require operational restrictions.

Answer

As detailed in Section 4, the reactor coolant pressure boundary is designed to minimize the probability of rapidly propagating type failures. To fulfill these requirements, the selection of materials for the systems and the fabrication of components are closely controlled and inspected. The details of the material selection and inspection procedures are contained in Section 4.

Criterion 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120 F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60 F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Answer

Sufficient testing and analysis of materials employed in reactor coolant system components will be performed to ensure that the required NDT temperature limits specified in the criterion are met. Removable test capsules will be installed in the reactor vessel and removed and tested at various times in the plant lifetime to determine the effects of operation on system materials. Details of the testing and analysis programs are included in Section 4.

Criterion 36 - Reactor Coolant Pressure Boundary Surveillance

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM E-185-1966.[3]

Answer

Provision has been made in the reactor coolant system design for adequate inspection testing and surveillance during the station's service lifetime. The vessel inspection program will conform to ASTM E-185-82. These provisions are discussed in detail in Section 4.

1.3.2.7 Engineered Safety Features

Criterion 37 - Engineered Safety Features Basis for Design

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Answer

The containment structure, the containment isolation system, the emergency core cooling system and the containment depressurization system comprise the engineered safety features for BVPS-1. These systems and their supporting systems (primary plant component cooling water system and river water systems) are designed to cope with any size reactor coolant pressure boundary break up to and including rupture of the largest reactor coolant pipe. The design bases for each system are included in the appropriate portions of Sections 5, 6, 9 and 10. An analysis of the performance of the engineered safety features is presented in Section 14.

Criterion 38 - Reliability and Testability of Engineered Safety Features

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the station.

Answer

All engineered safety features components are tested in the manufacturer's shop and after installation at BVPS-1 to demonstrate their reliability. Provision has also been made in the system design for periodic testing of engineered safety features during the station lifetime. Details of the tests to be performed and the basis for the determination of system reliability are included in appropriate portions of Sections 5, 6, 9 and 10.

Criterion 39 - Emergency Power for Engineered Safety Features

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity and testability to permit the functioning required of the engineered safety features. As a minimum the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Answer

Reliability of electric power supply is ensured through independent connections to the system grid and a redundant source of emergency power from two diesel generators installed in the station. Power to the engineered safety features is ensured even with the failure of a single active component. The station electrical systems, including network interconnections and the emergency power system, are described in Section 8.

Criterion 40 - Missile Protection

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from station equipment failure.

Answer

Engineered safety features are protected against dynamic effects and missiles resulting from equipment failures. The means of accomplishing this protection are described in Sections 5, 6 and 14.

Criterion 41 - Engineered Safety Features Performance Capability

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Answer

Sufficient redundancy and duplication is incorporated into the design of the engineered safety features to ensure that they could perform their function adequately even with the loss of a single active component. Details of the capability of these systems under normal and component malfunction conditions are included in Sections 6 and 9. An analysis of the adequacy of these systems to perform their functions is included in Section 14.

Criterion 42 - Engineered Safety Features Components Capability

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Answer

The design of the engineered safety features, the materials selected for fabrication of these systems and the layout of the various portions of the systems combine to ensure that the performance of the engineered safety features is not impaired by the effects of a loss-of-coolant accident. Details of the design and construction of the engineered safety features are included in Sections 5, 6 and 9. The ability of these features to perform their functions is analyzed in Section 14.

Criterion 43 - Accident Aggravation Prevention

Engineered safety features will be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Answer

The operation of the engineered safety features will not accentuate the after effects of a loss-of-coolant accident. These considerations are detailed in Sections 1, 4, 6 and 14.

Criterion 44 - Emergency Core Cooling Systems Capability

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that:

1. The capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation.

2. Failure of the shared feature or component does not initiate a loss-of-coolant accident.

3. Capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Answer

By combining the use of passive accumulators with two independent high pressure pumping systems and two independent low pressure pumping systems abundant emergency core cooling is provided even if there should be a failure of any component in any system. The emergency core cooling system employs a passive system of accumulators which do not require any external signals or source of power for their operation to cope with the short term cooling requirements of large reactor coolant pipe breaks. Two independent pumping systems, each capable of the required emergency cooling, are provided for small break protection and to keep the core submerged after the accumulators have discharged following a large break. These systems are arranged so that the single failure of any active component does not interfere with meeting the short term cooling requirements.

Two independent pump systems are provided, each capable of fulfilling long term cooling requirements. The failure of any single active component or the development of excessive leakage during the long term cooling period does not interfere with the ability to meet necessary long term cooling objectives with one of the systems.

The primary function of the emergency core cooling system is to deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel-clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical circumferential rupture of a reactor coolant loop

2. A loss-of-coolant associated with a rod ejection accident.

The basis criteria for loss-of-coolant accident evaluations are: no clad melting, Zircaloy-water reactions will be limited to an insignificant amount and the core geometry is to remain essentially in place and intact so that effective cooling of the core will not be impaired. The Zircaloy-water reactions will be limited to an insignificant amount so that the accident:

1. Does not interfere with the emergency core cooling function to limit clad temperatures

2. Does not produce hydrogen in an amount that when burned would cause the containment pressure to exceed the design value.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the emergency core cooling system adds shutdown reactivity so that with stuck rod, no off-site power and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact. With no stuck rod, off-site power available and all equipment operating at design capacity, there is insignificant cladding rupture. The emergency core cooling system is described in Section 6.

Criterion 45 - Inspection of Emergency Core Cooling Systems

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Answer

Design provisions facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes and valves for visual inspection and for nondestructive inspection where such techniques are desirable and appropriate.

The components outside the containment are accessible for leak- tightness inspection during operation of the reactor.

Details of the inspection program for the reactor vessel internals are included in Section 4. Inspection of the emergency core cooling system is discussed in Section 6.

Criterion 46 - Testing of Emergency Core Cooling System Components

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Answer

The emergency core cooling system design permits periodic testing of active components for operability and required functional performance. The test procedures are described in Section 6.

Criterion 47 - Testing of Emergency Core Cooling Systems

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Answer

By recirculation to the refueling water storage tank, the emergency core cooling system delivery capability can be tested periodically. The system can be so tested to the last valve before the piping enters the reactor coolant piping. Details of the system tested are included in Section 6.

Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling Systems

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Answer

Provision has been made in the emergency core cooling system design for testing the sequence of operation including transfer to alternate power sources. The details of these tests are included in Section 6 and the switching sequence from normal to emergency power is described in Section 8.

Criterion 49 - Containment Design Basis

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions, that could occur as a consequence of failure of emergency core cooling systems.

Answer

The containment structure and its heat removal system (containment depressurization system) are designed to accommodate the pressures and temperatures associated with a loss-of-coolant accident without exceeding the design leak rate. A considerable margin for unidentified energy sources has been included in the design. The loadings and energy sources considered in the design and the stress and loading criteria are described in Section 5. An analysis of the performance of the containment during a loss-of-coolant accident is included in Section 14. The heat removal systems are described in Section 6. Design of the concrete structure is discussed Section 5.

Criterion 50 - Nil Ductility Transition (NDT) Temperature Requirement for Containment Material

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30 F above NDT temperature.

Answer

All containment structure ferritic materials are selected to ensure that the NDT temperature for these materials is at least 30 F below normal operating and testing temperatures.

Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Answer

The reactor coolant pressure boundary is defined as those piping systems and components which contain reactor coolant at design pressure and temperature. With the exception of the reactor coolant sampling lines, the entire reactor coolant pressure boundary, as defined above, is located entirely within the containment structure. All sampling lines are provided with remotely operated valves for isolation in the event of a failure. These valves also close automatically on a containment isolation signal. Sampling lines are only used during infrequent sampling and can be readily isolated.

All other piping and components which may contain reactor coolant are low pressure, low temperature systems which would yield minimal environmental doses in the event of failure.

The sampling system and low pressure systems are described in Section 9. An analysis of malfunctions in these systems is included in Section 14.

Criterion 52 - Containment Heat Removal Systems

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Answer

Two full capacity quench spray subsystems and four 50 percent capacity recirculation spray subsystems are included in the BVPS-1 design as part of the containment depressurization system. These systems are described in Section 6. The performance of the containment depressurization system during a loss-of-coolant accident is described in Section 14.

Criterion 53 - Containment Isolation Valves

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Answer

At least two barriers are provided between the atmosphere outside the containment and the containment atmosphere, the reactor coolant system, or closed systems which are assumed vulnerable to accident forces. The valving installed on the various systems penetrating the containment and the other barriers employed in the design are described in Section 5.

Criterion 54 - Containment Leakage Rate Testing

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Answer

Provision is included in the containment vessel design for integrated leak rate testing after completion of construction. The test procedure is described in Section 5 and the Technical Specifications and is formulated to demonstrate that leakage is below the design value of 0.1 percent per day.

Criterion 55 - Containment Periodic Leakage Rate Testing

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during station lifetime.

Answer

Provision for full integrated leak rate testing of the containment is incorporated in the design. The testing procedures are discussed in Section 5 and the Technical Specifications.

Criterion 56 - Provisions for Testing of Penetrations

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

Answer

Each containment penetration includes a means to test its leaktightness. These means are described in Section 5.

Criterion 57 - Provisions for Testing of Isolation Valves

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Answer

Class B and C tests are performed on containment isolation valves to verify their sealing capability and leaktightness. The tests are discussed in Section 5.6.

Criterion 58 - Inspection of Containment Pressure-Reducing Systems

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus and sumps.

Answer

The design of the containment depressurization system includes provision for physical inspection of vital components. The inspectability of the systems is discussed in Section 6.

Criterion 59 - Testing of Containment Pressure-Reducing Systems Components

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Answer

The various pumps and valves in the containment depressurization system can be periodically tested. Component testing of the containment depressurization system is discussed in detail in Section 6 and the Technical Specifications.

Criterion 60 - Testing of Containment Spray Systems

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Answer

Provision is made to permit testing of the containment depressurization system throughout the life of the station to ensure that this is operational. Details of the testing of this system are discussed in Section 6 and the Technical Specifications.

Criterion 61 - Testing of Operational Sequence of Containment Pressure-Reducing Systems

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Answer

Capability for testing of the operational sequence of the containment depressurization system is incorporated into the system design. Details of the containment depressurization system are included in Section 6. The switching sequence from normal to emergency power is described in Section 8.

Criterion 62 - Inspection of Air Cleanup Systems

Design provisions shall be made to facilitate physical inspection of all critical parts of the containment air cleanup systems, such as ducts, filters, fans and dampers.

Answer

The air cleanup system is the containment depressurization system. The containment depressurization system sprays a sodium tetraborate decahydrate solution into the containment following a LOCA so as to clean up iodine in the containment atmosphere. Design provisions have been made so that critical portions of this system can be inspected.

Criterion 63 - Testing of Air Cleanup Systems Components

Design provisions shall be made so that active components of the air cleanup systems, such as fans and damper, can be tested periodically for operability and required functional performance.

Answer

Provisions have been made so that active components of the containment depressurization system can be tested periodically. Testing of this system is discussed in detail in Section 6 and the Technical Specifications.

Criterion 64 - Testing of Air Cleanup Systems

A capability shall be provided for insitu periodic testing and surveillance of the air cleanup systems to ensure:

1. Filter bypass paths have not developed

2. Filter and trapping materials have not deteriorated beyond acceptable limits.

Answer

The containment depressurization system does not have any cleanup filter or trapping materials.

Criterion 65 - Testing of Operational Sequence of Air Cleanup Systems

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

Answer

Provisions have been incorporated so that the operational sequence of the containment depressurization system can be tested. A discussion of the operational sequence testing of the system is included in Sections 6 and 8.

Criterion 66 - Prevention of Fuel Storage Criticality

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Answer

Criticality in new and spent fuel storage areas is prevented both by physical separation of new and spent fuel elements and the presence of borated water in the spent fuel pool. Criticality prevention is discussed in detail in Section 9.

Criterion 67 - Fuel And Waste Storage Decay Heat

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to station operating areas or the public environs.

Answer

The fuel pool cooling system provides decay heat removal for the spent fuel pool. Details of the fuel pool cooling system and fuel handling facilities are described in Section 9.

Criterion 68 - Fuel Waste Storage Radiation Shielding

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10CFR20.

Answer

Shielding is provided for fuel handling and waste storage areas to reduce radiation doses to levels below the limits specified in 10CFR20. Shielding for these areas and other station shielding requirements and criteria are included in Section 11.3.

Criterion 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Answer

All fuel storage and waste storage facilities are designed to prevent the release of undue radioactivity to the public. Fuel storage facilities are described in Section 9, waste storage facilities are described in Section 11 and analysis of potential accidents in these systems is included in Section 14.

Criterion 70 - Control of Releases of Radioactivity to the Environment

The facility design shall include those means necessary to maintain control over the station radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified on:

1. The basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur

2. The basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Answer

Provision is included in the station design for storage and processing of radioactive waste and the release of such wastes under controls adequate to prevent exceeding the limits of 10 CFR 20. BVPS-1 also includes provision to prevent radioactivity releases during accidents from exceeding the limits of 10 CFR 100 or 10 CFR 50.67, as applicable. Descriptions of the radioactive waste disposal systems are included in Section 11. The effects of potential accidents, including a loss-of-coolant accident, are analyzed in Section 14.

1.3.3 Safety Guides

The AEC Safety Guides applicable to BVPS-1 design and construction are provided below. Following each Safety Guide is a summary discussion of BVPS-1's method of satisfying each regulatory position.

1.3.3.1 Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)

BVPS-1 complies with the intent but not the letter of Safety Guide 1.

The regulatory position expressed in Safety Guide 1 is similar to part of the design bases as presented in Sections 6.3 and 6.4.

The operation of the emergency core cooling system is not dependent on containment pressure until after the refueling water storage tank is empty. By the time recirculation safety injection is required, the net positive suction head (NPSH) available to the recirculation pumps is sufficient to ensure satisfactory performance under all conditions. The recirculation spray pumps start following a CIB signal in conjunction with a RWST low level signal and are only required if an increase in containment pressure occurs. Consequently, sufficient NPSH exists throughout their operating range. The original design of the recirculation spray subsystems for the BVPS-1 was similar to that for the Surry Power Station Units 1 and 2 and the North Anna Power Station Units 1 and 2. The conformance to Safety Guide 1 was discussed in the design of all these units by specific answers to AEC questions. The basis for achieving sufficient NPSH was presented to the NRC in Reference 1 and shown to be acceptable by issuance of Amendment No. 28 to Facility Operating License No. DPR-66 for BVPS-1.

1.3.3.2 Thermal Shock to Reactor Pressure Vessels (Safety Guide 2)

Compliance with Safety Guide 2 for Thermal Shock to Reactor Pressure Vessels is discussed in detail in Section 4 and the Technical Specifications.

1.3.3.4 Assumption Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors (Safety Guide 4)

BVPS-1 has implemented alternative source term methodology in evaluating the potential radiological consequences of a loss of coolant accident. Regulatory Guide 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an alternative source term.

The guidance of Regulatory Guide 1.183 supersedes corresponding radiological analysis assumptions provided in other regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67. Refer to the discussion of Regulatory Guide 1.183 in Section 1.3.4.1 of the UFSAR.

Accident meteorology is discussed in Section 2.2 and Appendix 2A and the offsite dose calculations for the DBA are discussed in Section 14.3.

1.3.3.6 Independence Between Redundant Standby (Onsite) Power Sources and Between their Distribution Systems (Safety Guide 6)

Adequate redundancy and independence exists between standby (onsite) power sources and between their distribution systems in accordance with the AEC regulatory position outlined in Safety Guide 6.

The electrical power loads for engineered safety features are separated into redundant load groups fed from separate buses such that loss of one group will not prevent operation of minimum safety functions.

The redundant power loads are each connected to buses which may have power fed from an offsite power source or an onsite power source (a diesel generator).

Four 125 v d-c systems, each complete with batteries, chargers, switchgear, and distribution equipment, are provided for engineered safety features equipment. These systems are not tied together.

A standby source of power for one redundant load cannot be automatically paralleled with the standby source of power for the other redundant load.

Each redundant 120 a-c engineered safety features load is supplied with power from a separate emergency diesel generator.

Figures 8.5-1 and 8.5-2 illustrate the physical arrangement of the emergency switchgear, 125 v d-c batteries, battery chargers, 125 v d-c distribution panels and 120 v a-c vital bus system equipment.

1.3.3.7 Deleted

1.3.3.8 Personnel Selection and Training (Safety Guide 8)

Application of Regulatory Guide 1.8 during the operations phase of BVPS-1 is described in Section 1.3.4.

The organization of BVPS-1 is described in Section 12.1. The training program for the various job classifications is listed in Section 12.2.

1.3.3.9 Selection of Diesel Generator Set Capacity for Standby Power Supplies (Safety Guide 9)

Each emergency generator set is rated as follows:

2,600 kW	8,760 hr/yr
2,850 kW	2,000 hr/yr
2,950 kW	168 hr/yr
3,050 kW	0.5 hr/yr

The basis for this rating is analyzed by emergency diesel generator steady state analysis. Each diesel generator is sized to ensure that the total loads of the engineered safety features required to be powered at any one time does not exceed 90 percent of the 0.5 hour per year rating.

Sequence starting of the motors associated with the safety features is provided to reduce the instantaneous load on the diesel generator. This ensures that adequate power is available to start and accelerate to rated speed all engineered safety features and emergency shutdown loads.

Each emergency diesel generator is up to speed and capable of accepting load within 10 seconds and energizes designated loads in a stepped sequence operation within an additional 60 seconds.

The speed and voltage variations of each emergency diesel generator are within the limits as set forth in Safety Guide 9.

During the preoperational test, the predicted engineered safeguard loads were verified by tests.

The selection of diesel generator capacity for standby power supply is in accordance with the intent of AEC Safety Guide 9.

1.3.3.10 Mechanical Cadweld Splices in Reinforcing Bars of Concrete Containments (Safety Guide 10)

Testing and sampling of mechanical splices in reinforcing bars used in the containment and other structures are in compliance with Safety Guide 10. Crew qualifications used differed from those required by Safety Guide 10. During the initial stages of construction, each member of the Cadweld crew was required to make one qualification splice for each 200 Cadwelds made. Later, this qualification requirement was increased to require two qualification splices, one in a vertical, and one in a horizontal position. Qualification requirements for Cadweld crew members, and testing and sampling the mechanical Cadweld splices in reinforcing bars are discussed in Section 5.2.5.3.

During the operations phase, splicing reinforcing bars shall be performed in accordance with individual project specifications. Project specifications shall include or reference manufacturer's instructions and comply with the applicable requirements of ANSI N45.2.5-1974. The company Quality Assurance Program Manual (QAPM) identifies specific subarticles of ASME Section III Division 2-1995 edition that will be used in lieu of the corresponding requirements in ANSI N45.2.5-1974.

1.3.3.11 Instrument Line Penetrating Primary Reactor Containment (Safety Guide 11)

There are seven instrument sensing lines that penetrate the reactor containment as follows:

1. Four penetrations for the engineered safety features containment pressure

2. Two penetrations for the particulate and gas activity monitor

3. One penetration for the pressurizer dead weight calibrator.

The instrument sensing lines that are part of the protection system, Group 1 above, are redundant and independent and provide for testing of the protection system. These lines are provided with restriction orifices inside the containment to limit the inflow of air caused by an external failure of a line, valve body or instrument. This is in accordance with Safety Guide 11.

The sensing lines for Group 2 above are equipped with automatic containment isolation valves in accordance with Safety Guide 11.

The sensing line in Group 3 is discussed in Section 5.3.

1.3.3.12 Instrumentation for Earthquakes (Safety Guide 12)

Instrumentation for earthquake monitoring for BVPS-1 is provided in accordance with Safety Guide 12. This instrumentation is described in Section 5.2.8.1.

1.3.3.13 Fuel Storage Facility Design Basis (Safety Guide 13)

The fuel handling and storage facilities are designed to meet the following objectives:

1. Prevent loss of water from the fuel pool which could uncover spent fuel

2. Provide a safe, effective means of handling fuel, and protect it from mechanical damage

3. Provide a means of limiting any potential radioactive release to the environment in the event of a fuel handling accident.

Detailed explanations of how these objectives are met are outlined in the sections provided below. As explained in these sections, the fuel handling and storage facilities are in compliance with the regulatory positions set forth in Safety Guide 13. One exception is to regulatory position C.3 concerning interlocks to prevent cranes from passing over stored fuel when fuel handling is not in progress. However, the fuel handling movable platform is the only hoist operable over the spent fuel storage area. When not in use, administrative procedures require that the fuel handling movable platform have no loads suspended over stored fuel. In addition, the movable platform is designed as a Seismic Category I component.

The fuel cask crane is operable over the spent fuel cask laydown area of the spent fuel pool. This crane meets the requirements for single failure proof cranes (NUREG-0554).

The following table cross references the Regulatory Positions stated in Safety Guide 13 and related Updated FSAR sections:

Regulatory Position Number	Related FSAR Section
C1	2.5 - Seismology
C2	2.7 - Site Design Data
C3	9.12.2 - Fuel Handling System
C4	11.3.3.6 - Fuel Building Ventilation Monitor
C5	9.12.2 - Fuel Handling System
C6	9.5.1 - Fuel Pool Cooling and Purification System
C7	11.3.4 - Area Radiation Monitoring System
C8	9.5.2 - Fuel Pool Cooling and Purification System

1.3.3.14 Reactor Coolant Pump Flywheel Integrity (Safety Guide 14)

The design conforms with the intent of Safety Guide 14. The shaft and the bearings supporting the flywheel are capable of withstanding any combination of the normal operating loads, anticipated transients, the design basis LOCA and the Design Basis Earthquake loads.

The flywheel integrity is described in Section 4.2.2.5.

1.3.3.15 Testing of Reinforcing Bars for Concrete Structures (Safety Guide 15)

Testing and inspecting of reinforcing bars are in compliance with provisions stated below.

In compliance with the Safety Guide 15, a full diameter specimen for all special chemistry (Grade 50) reinforcing steel was tested in accordance with ASTM A-370-1968.[5] One such test was made for each heat. Acceptance standards for all rebars are in accordance with ASTM A-615-1968[6], as further described in Section 5.2.3.3.

During the operations phase, splicing reinforcing bars shall be performed in accordance with individual project specifications. Project specifications shall include or reference manufacturer's instructions and comply with the applicable requirements of ANSI N45.2.5-1974. The company Quality Assurance Program Manual (QAPM) identifies specific subarticles of ASME Section III Division 2-1995 edition that will be used in lieu of the corresponding requirements in ANSI N45.2.5-1974.

1.3.3.16 Reporting of Operating Information (Safety Guide 16)

Operating reports for the BVPS-1 are submitted to the NRC in accordance with the Technical Specifications.

1.3.3.17 Protection Against Industrial Sabotage (Safety Guide 17)

The probability and effects of industrial sabotage are reduced at the Beaver Valley Power Station by:

1. Control of access of personnel and material to station within the security fence:
2. Control of movements of personnel within the station
3. Careful selection and review of station personnel
4. The monitoring of vital station equipment
5. Design and arrangement of station features.

The methods used comply with the intent of AEC Safety Guide 17. The Security Plan is discussed in Section 12.7.

1.3.3.18 Structural Acceptance Test for Concrete Primary Reactor Containments (Safety Guide 18)

The structural acceptance test for the containment structure will equal or exceed the requirements of Safety Guide 18. Changes will be made in the number and location of points for measuring deflections at the largest opening with a thickened ring beam. The points will be located in areas of highest stress.

For the largest opening with a thickened ring edge beam, radial deflections will be measured at twelve points, three located at each of 4 azimuthal positions (3, 6, 9, and 12 o'clock).

The change in diameter of the thickened ring will be measured on the inside and outside edges, on the horizontal and vertical diameters of the clear opening.

The procedures for acceptance test for the containment structure are described under Section 5.6.

1.3.3.19 Nondestructive Examination of Primary Containment Liner Seam Welds (Safety Guide 19)

The nondestructive examination of containment liner seal welds was done in the following steps.

Every liner seam weld was dye-penetrant tested. After the channels were welded, they were air pressurized up to containment design pressure and soap suds applied to the welds; next the channels were air vacuum and halogen pressurized up to containment design pressure for leak detection.

On the floor and lower portion of the cylinder (where backing strips are used) the channel to the liner weld was dye-penetrant tested before being air pressurized. Soap bubble and halogen detection were done from one side only.

On the rest of cylinder and dome the seam welds were dye-penetrant tested on both sides. The soap bubble test was done on the outside and the halogen test was done on the inside.

The deviations from the Safety Guide 19, published after the tests were completed, ensure a more severe test. For example, using pressurized channels to a greater pressure difference than is possible with a vacuum box. By using halogen leak detection instead of pressure drop detection (C.1.d), a more sensitive test was accomplished.

Containment liner seam welds on repaired sections of the liner are also nondestructively examined. Visual examination, magnetic particle, and ultrasonic examination of the entire repair weld are utilized. A local leak rate test is performed with a zero leakage acceptance criteria.

1.3.3.20 Vibration Measurements on Reactor Internals (Safety Guide 20)

Westinghouse complied with the requirements of Safety Guide 20 (now Regulatory Guide 1.20) for vibration measurements on reactor internals (Section 3.2.2.6).

For each prototype reactor internals design, a program of vibration analysis, measurement and inspection was developed and reviewed by the AEC prior to the performance of the scheduled preoperational functional test. Westinghouse has prepared the vibrational analysis and test programs for prototype 2, 3 and 4-loop plants. The status of these programs is given in Table 1.3-1.

This subject is discussed in Section 3.3.3.

1.3.3.21 Measuring and Reporting of Effluents from Nuclear Power Plants (Safety Guide 21)

BVPS-1 radioactive effluents are reported to the NRC in accordance with the recommendations in Regulatory Guide 1.21, Revision 1 issued June 1974, except for population dose. Population dose as discussed in Regulatory Guide 1.21 Revision 1 will not be calculated on an annual basis. The radiological impact to man will be calculated using the individual receptor as described in the Offsite Dose Calculation Manual. All normal and potential paths for release of radioactive materials during normal reactor operation are continuously monitored and recorded as detailed in Section 11.3. The sampling system is discussed in Section 9.6. The frequency of sampling and reporting of effluents is discussed in the Offsite Dose Calculation Manual.

1.3.3.22 Periodic Testing of Protection System Actuation Functions (Safety Guide 22)

The protection system is in accordance with IEEE Std. 279-1971. Safety actuation circuitry is provided with a capability for testing with the reactor at power. The design of the protection system, including the Engineered Safety Features Test Cabinet, complies with Safety Guide 22. Under the present design, there are protection functions which are not tested at power. These are:

1. Generation of a reactor trip by tripping the turbine
2. Generation of a reactor trip by use of the manual trip switch
3. Generation of a reactor trip by manually actuating the safety injection system
4. Generation of safety injection signal by use of the manual safety injection switch
5. Generation of containment spray signal by use of the manual spray actuation switch.
6. Testing of the automatic transfer from safety injection phase to recirculation phase. The final actuators for this feature will be tested during plant shutdown.

Exception to testing the devices listed above is taken, as allowed by Safety Guide 22, where it has been determined that:

1. "There is no practicable system design that would permit operation of the equipment without adversely affecting the safety or operability of the plant."

   The present position is that it is not a "practicable system design" to provide equipment to bypass a device such as a main steam line stop valve solely to test the device. In the case of manual initiation switches, the design for test capability would require that switches be provided on a train or sequential basis. This increases the operation action required to manually actuate the function.

2. "The probability that the protection system will fail to initiate the operation of the equipment is, and can be maintained, acceptably low without testing the equipment during reactor operation."

   Probabilities have been established by the use of general failure data based on continuous operation. Specific probability analyses will be provided on a plant basis at the request of the commission.

3. "The equipment can routinely be tested when the reactor is shut down."

In all the cases discussed above, it is only the device function that is not tested. The logic associated with the devices has the capability for testing at power.

Refer to Sections 7.2 and 7.3 for further discussion.

1.3.3.23 Onsite Meteorological Programs (Safety Guide 23)

The BVPS-1 onsite meteorological program complies with Regulatory Guide 1.23 as described in Section 2.2.3.

1.3.3.24 Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)

The assumptions used for evaluating the potential radiological consequences of radioactive gas storage tank ruptures are provided in Section 11.2.3.4 and Table 11.5-8.

1.3.3.25 Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)

An alternative radiological source term was used in evaluating the fuel handling accident. Therefore, the assumptions used for evaluating the potential radiological consequences of a fuel handling accident follow the guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 1, 2000. The fraction of core inventory in the fuel rod gap is based on Draft Regulatory Guide DG-1199.

1.3.3.26 Quality Group Classifications and Standards (Safety Guide 26)

Components for BVPS-1 were classified by quality assurance categories, as discussed in Appendix A.1.

Compliance with GDC 1 is discussed in Appendix 1A.1. Design and fabrication criteria for the Engineered Safety Feature (ESF) equipment is covered in Section 6.2. The codes and standards applicable to other systems and components are discussed within the respective sections.

1.3.3.27 Ultimate Heat Sink (Safety Guide 27)

The ultimate heat sink of BVPS-1 is the Ohio River. The river is the water source of the cooling water system that removes residual heat after reactor shutdown and following an accident.

Although the ultimate heat sink consists of a single source, design consideration has been given to the most credible minimum and maximum river levels in compliance with Safety Guide 27.

A discussion of how the Regulatory Positions set forth in Safety Guide 27 were implemented are presented below. Identify each exception taken and supply the basis for each exception.

Regulatory Position C.1

An analysis of how position C.1 is implemented is provided in Section 2.3.4.

Regulatory Position C.2

The Ohio River is capable of withstanding the effects of the natural phenomena discussed in this position and a single failure of man-made structural features without loss of the capability specified in position C.1. Supporting information is found in Sections 2.3.1.1, 2.3.3 and 2.3.4.

Regulatory Position C.3

The sections referenced in Position C.2 clearly demonstrate that there is an extremely low probability of losing the capability of the Ohio River as the ultimate heat sink. As such a second source of cooling water was not considered.

Regulatory Position C.4

As discussed in Section 2.3.11, the minimum water surface elevation of 648.6 ft provides sufficient submergence margin for the safety related pumps located in the intake structure to support normal shutdown and long term cooling. Further, as discussed in Section 9.9.3, a minimum operability level of 654 ft. assures sufficient submergence margin for these pumps to meet design basis accident cooling requirements. Technical Specification limitations have been imposed to require the Unit to shutdown to cold shutdown if the Ohio River water level decreases below an elevation of 654 ft.

Sections 2.3.3 and 2.3.4 (and associated documentation from the U.S. Army Corps of Engineers) discuss the probability and consequences of upstream or downstream dam failures and the resulting maximum and minimum river water levels expected. These sections show that even with a single failure of one dam concurrent with high or low river water flow sufficient cooling water is always available from the Ohio River for residual heat removal. Diversion of the river due to natural or accidental phenomena is not considered credible.

Therefore, the following conclusions are presented:

1. Severe natural phenomena, such as earthquake, tornado, hurricane, etc., could not cause diversion of the river or loss of the water supply even if these cause a dam single failure.

2. Site related events or accidental phenomena could not cause diversion or loss of water for the same reasons as discussed in Section 2.1.7.

1.3.3.28 Quality Assurance Program Requirements (Design and Construction) (Safety Guide 28)

The Duquesne Light Company Quality Assurance Program during the design and construction phase is described in Appendix A.2.1. This program was intended to fulfill the intent of Appendix B to 10 CFR 50.

The Stone & Webster Quality Assurance Program is described in Appendix A.3. This program implemented the intent of 10 CFR 50, Appendix B.

The Westinghouse Quality Assurance Plan described in Appendix A.4 for safety related NSSS equipment complied with the requirements of ANSI N45.2.(8) The requirements provided therein apply to the design and fabrication of safety related equipment, and therefore, satisfies Safety Guide 28.

1.3.3.29 Seismic Design Classification (Safety Guide 29)

The seismic design of the BVPS-1 structures and components is discussed in Appendix B. Seismic Category I components, systems and structures are listed in Table B.1-1. The NSSS fluid systems component seismic category list is given in Table B.3-1.

The terminology Operational Basis Earthquake (OBE) and Design Basis Earthquake (DBE) is considered comparable to the terms 1/2 Safe Shutdown Earthquake (1/2 SSE) and Safe Shutdown Earthquake (SSE).

1.3.3.30 Quality Assurance Requirements for the Installation Inspection and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

Quality assurance requirements for the installation, inspection and testing of instrumentation and electric equipment was, to the greatest extent possible, in accordance with Safety Guide 30. The quality assurance program is in Appendix A.

Application of Regulatory Guide 1.30 during the operations phase of BVPS-1 is described in Section 1.3.4.

1.3.3.31 Control of Stainless Steel Welding (Safety Guide 31)

Stone & Webster Position

"Based on the control of ferrite in the purchased electrode materials and the controls during procedure qualification, it is Stone & Webster's position that the intent of Safety Guide 31 is being met and that no ferrite measurements of deposited weld metal are necessary. The specifications for ferrite control at BVPS-1 required that the ferrite content purchased in electrodes, as determined by Severn Gage and Schaeffler Diagram, be 5 to 15 percent.

Welding procedures in use at BVPS-1 have been qualified using electrodes with 5 to 15 percent ferrite. Weld samples were examined for microfissuring. No unsatisfactory conditions were noted. Heat input is controlled by restricting range of arc voltage and amperes allowed for welding."

Westinghouse Position

"Safety Guide 31 states that weld deposits should contain between 5 and 12 to 15 percent delta ferrite. It is not practical to specify absolute minimum or even maximum delta ferrite limits as a basis for acceptance or rejection of otherwise acceptable austenitic stainless steel welds.

The Westinghouse PWR criteria placed control on the actual wire analysis for inert gas welding processes and on the final weld deposit for the fluxing weld process as follows.

In the case of the bare wire when used with inert gas processes, although the wire may contain 5 percent ferrite, only about 1 or 2 percent ferrite will be developed in the resultant weld deposits. This is not the case in fluxing processes, such as when using coated electrodes or submerged arc since the flux is enriched with additional ferrite formers resulting in higher ferrite contents in the resultant weld deposits. Similarly, the amount of ferrite that may exist in any given weld will vary across the width of the weld deposit depending on the base materials being joined. For example, when fully austenitic wrought product is welded, the interface regions will be practically zero percent ferrite because of the resultant base metal dilution, but it will progressively increase toward the weld centerline. Conversely, when a two phase (austenite + ferrite) case product that normally contains over 15 percent ferrite is welded, the interface region will be high in delta ferrite content depending on the amount of delta ferrite available and diluted from the casting base material.

The ferrite distribution in a weld will also vary depending on the weld position. This is, in areas of the downhand and horizontal position, weld deposit ferrite will be the highest; whereas, in the vertical and overhead position, weld deposit ferrite will be the lowest in a given weld because of different weld or manipulations necessary to overcome effects of gravity.

In addition, Type 310 and 330 weld materials are always fully austenitic; yet sound welds are being made every day with these alloys using fine-tuned welding procedures. Also, welds are being made without the use of weld metal such as, electron beam welds and autogenous gas shielded tungsten arc welds.

Furthermore, the limits as set are arbitrary because various methods used to measure the percentage of delta ferrite yield widely differing results. The Welding Research Council has recognized this situation and has an organized approach that may result in an acceptable solution.

The basis for classifying the low, medium, and high energy input ranges is not given in the guide. Using our conservative welding procedure parameters, the following energy inputs are being applied everyday in producing high quality welds. They are (for American Welding Society designations):

1. SMAW 15.4 to 95 kJ/in. using 1/16 to 3/16 diameter electrodes

2. GTAW 2.16 to 32.5 kJ/in. using .03 to 1/8 diameter wires

3. GMAW 46 to 55 kJ/in. using .03 to 1/16 diameter wires

4. SAW 74 to 79 kJ/in. using .09 to 1/8 diameter wires

We have a large amount of evidence showing that the above energy input ranges produce fissure-free weldments in shop and on site welding.

Westinghouse PWR does not require in process delta ferrite determination. When the welding material is tested in accordance with the requirements of Section III, to the ASME Code (ASME-III) NB2430, and includes delta ferrite determination that sound welds displaying more than one percent average delta ferrite content by any agreed method of determination will be considered unquestionable. All other sound welds that display less than one percent average delta ferrite will be considered acceptable providing that there is no evidence of malpractice or deviation from procedure parameters. If evidence of the latter prevails, sampling will be required to determine the acceptability of the welds. The sample size shall be 10 percent of the welds in the system or component. If any of these weld samples are defective, that is, fail to pass bend tests as prescribed by ASME Code, Section IX, all remaining welds shall be sampled and all defective welds shall be removed and replaced."

1.3.3.32 Use of IEEE STD-308-1971 "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations" (Safety Guide 32)

Class IE electric systems, to the greatest extent possible, comply with Safety Guide 32.

Availability of offsite power is discussed in Appendix 1A.17.

The capacity of each battery charger supply is based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery to the fully charged state, irrespective of the status of the plant during which these demands occur.

1.3.3.33 Quality Assurance Program Requirements (Operation) (Safety Guide 33)

BVPS-1 has formed a Quality Assurance Department. This department is responsible for the administration of the operational quality assurance program.

The BVPS-1 Quality Assurance Manual has been revised to incorporate quality assurance for operations. This program complies with AEC Safety Guide 33. ANSI N45.2 and ANSI N18.7[9] (previously ANS 3.2) requirements are referenced within Safety Guide 33.

BVPS-1 Quality Control is responsible for the preparation of the quality control procedures necessary to comply with Safety Guide 33.

Application of Regulatory Guide 1.33 during the operations phase of BVPS-1 is described in Section 1.3.4.

1.3.4 <u>Guidelines Used for the Operations Phase</u>

1.3.4.1 Regulatory Guides

REGULATORY GUIDE 1.8, PERSONNEL SELECTION AND TRAINING

See the company Quality Assurance Program Manual, Regulatory Guide 1.8.

REGULATORY GUIDE 1.30, QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT

See the company Quality Assurance Program Manual, Regulatory Guide 1.30.

REGULATORY GUIDE 1.33: QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATIONS)

See the company Quality Assurance Program Manual, Regulatory Guide 1.33.

REGULATORY GUIDE 1.37: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.37.

REGULATORY GUIDE 1.38: QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.38.

REGULATORY GUIDE 1.39: HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.39.

REGULATORY GUIDE 1.52, REVISION 2: DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR POST ACCIDENT ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (MARCH 1978)

Testing criteria for the BVPS Unit 1 Control Room emergency outside air pressurization system meets the intent of Positions C.5 and C.6 of this regulatory guide with the following alternatives:

Paragraph C.5.b

The airflow capacity and flow distribution test procedure will be developed based on ANSI N510-1980, "Testing of Nuclear Air Cleaning Systems," Section 8, except that:

1. To avoid damage to system components, an artificial resistance may be used in lieu of the recommendations of Paragraph 8.3.1.1;

2. The Paragraph 8.3.1.6 airflow capacity test will be performed with the filter bank at 100 percent of design dirty pressure drop, since the system design and surveillances preclude inadvertent operation of the filter banks with the pressure or flow rate outside of the allowable limits;

3. The Paragraph 8.3.1.7 airflow capacity test will be performed at a point between the design dirty and design clean pressure drops; and

4. The airflow distribution through the HEPA filters will not be tested since the design employs a single HEPA filter located downstream of the charcoal adsorber.

Paragraph C.5.c

The air-aerosol mixing uniformity test in Section 9 of ANSI N510-1980 will not be performed since the Unit 1 emergency ventilation subsystem contains only one HEPA filter.

The in-place leak test of the HEPA filters will be performed in accordance with Section 10 of ANSI N510-1980 to confirm a penetration of less than 0.05% at a flow rate of 800 to 1,000 cfm.

Paragraph C.5.d

The in-place leak test of the carbon adsorber will be performed in accordance with Section 12 of ANSI N510-1980 to ensure that bypass leakage is less than 0.5% at a flow rate of 800 to 1,000 cfm.

Paragraph C.6.b

The laboratory testing for a carbon adsorbent will be performed at the frequency described in Technical Specifications. The carbon samples not obtained from test canisters will be prepared in accordance with Technical Specification requirements. Laboratory test conditions will be in accordance with ASTM D3803-1989 as described in Technical Specifications.

REGULATORY GUIDE 1.54, JUNE, 1973: QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS

BVPS-1 follows the guidance of Regulatory Guide 1.54. Procedures and/or specifications were developed prior to, and implemented concurrent with the start of the operations phase.

REGULATORY GUIDE 1.58: QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION AND TESTING PERSONNEL

See the company Quality Assurance Program Manual, Regulatory Guide 1.58.

REGULATORY GUIDE 1.64: QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.64.

REGULATORY GUIDE 1.74, QUALITY ASSURANCE TERMS AND DEFINITIONS

See the company Quality Assurance Program Manual, Regulatory Guide 1.74.

REGULATORY GUIDE 1.88: COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

See the company Quality Assurance Program Manual, Regulatory Guide 1.88.

REGULATORY GUIDE 1.94, QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.94.

REGULATORY GUIDE 1.116, QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS

See the company Quality Assurance Program Manual, Regulatory Guide 1.116.

REGULATORY GUIDE 1.123, QUALITY ASSURANCE REQUIREMENTS FOR THE CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.123.

REGULATORY GUIDE 1.144: AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.144.

REGULATORY GUIDE 1.146, QUALIFICATION OF QUALITY ASSURANCE PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS

See the company Quality Assurance Program Manual, Regulatory Guide 1.146.

REGULATORY GUIDE 1.155, JUNE 1988: STATION BLACKOUT

The utilization of BVPS emergency diesel generators as alternate AC (AAC) power sources for coping with station blackout, and the reliability program for these generators follow the guidance of Regulatory Guide 1.155 (June 1988).(10,11)

REGULATORY GUIDE 1.163, SEPTEMBER 1995: PERFORMANCE-BASED CONTAINMENT LEAK TEST PROGRAM

This regulatory guide provides guidance on an acceptable performance based leak test program, leakage rate test methods, procedures, and analyses that may be used to comply with the performance based Option B in Appendix J of 10 CFR 50. With the issuance of License Amendment 293, BVPS Unit 1 now complies with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the Code of Federal Regulations] Part 50 Appendix J," instead of Regulatory Guide 1.163, "Performance Based Containment Leak Test Program." Refer to UFSAR Section 5.6 for additional discussion of containment leakage rate tests.

REGULATORY GUIDE 1.183, JULY 2000: ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

This regulatory guide provides assumptions, methods and acceptance criteria that are acceptable to the NRC staff for performing design basis radiological analyses using an alternate source term. This regulatory guide is utilized to evaluate the potential radiological consequences of all BVPS-1 design basis accidents with the following exception. The radiological dose consequences of a Waste Gas System Rupture are addressed in UFSAR Section 11.2.3.4 and do not utilize the Regulatory Guide 1.183 methodology.

Draft Regulatory Guide DG-1199, October 2009: Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

<u>Regulatory position C.3.2</u>

The gap fractions provided in Table 3 of draft Regulatory Guide DG-1199 are used for all non-LOCA events other than reactivity initiated accidents, where only the fuel clad is postulated to be breached. (The Control Rod Ejection Accident, CREA, is considered to be reactivity initiated.) The gap fractions used to assess the dose consequences of the Locked Rotor Accident (LRA) and the Fuel Handling Accident (FHA) are as shown in UFSAR Section 14B.2. Table 3 in Regulatory Guide 1.183 specifies the fraction of fission product inventory assumed to be in the fuel rod gap to be used for the Loss of Coolant Accident (LOCA) and the CREA.

1.3.4.2 American National Standards Institute (ANSI) Standards

N45.2.5: "SUPPLEMENTARY QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS"

See the company Quality Assurance Program Manual, Regulatory Guide 1.94.

N45.2.8: "SUPPLEMENTARY QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS FOR THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS"

See the company Quality Assurance Program Manual, Regulatory Guide 1.116.

N45.2.13: "QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS"

See the company Quality Assurance Program Manual, Regulatory Guide 1.123.

References for Section 1.3

1. "Analysis and System Modification for Recirculation Spray and Low Head Safety Injection Pumps Net Positive Suction Head, Final Report," Beaver Valley Power Station - Unit No. 1, Docket No. 50-334, License No. DPR-66, Stone & Webster Engineering Corporation (November 17, 1977).

2. "IEEE Criteria for Protection Systems for Nuclear Power Generation," IEEE Std. 279-1971, The Institute of Electrical and Electronic Engineers, Inc.

3. "ASTM Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," ASTM E-185-1966, The American Society for Testing Materials.

4. Deleted

5. "ASTM Mechanical Testing of Steel Products Methods and Definitions," ASTM A-370-1968, The American Society for Testing Materials.

6. "ASTM Deformed and Plain Billet Steel Bars for Concrete Reinforcement," ASTM A-615-1968, The American Society for Testing Materials.

7. "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654/EEMA-REP-1, Rev. 1, U.S. Nuclear Regulatory Commission and Federal Emergency Management Agency (November 1980).

8. "ANSI Quality Assurance Program Requirements for Nuclear Power Plants," ANSI Std. N45.2, The American National Standards Institute.

9. Deleted

10. Letter from A. W. De Agazio (Nuclear Regulatory Commission) to J. D. Sieber (Duquesne Light Company), Subject: Safety Evaluation Related to Station Blackout, dated November 23, 1990.

11. Letter from J. D. Sieber (Duquesne Light Company) to the Nuclear Regulatory Commission, Subject: Response to Safety Evaluation Report for Station Blackout, dated December 20, 1990.

1.4 COMPARISON WITH OTHER STATIONS

The comparisons with other stations provided herein reflects the status of BVPS-1 at the time of the issuance of the Operating License. This information is being retained for historical perspectives. Submission of new material in this section is not required since design changes are incorporated in the text throughout the Updated FSAR.

Table 1.4-1 presents a summary of the design and operating parameters for the BVPS-1 NSSS. The table compares these data with data available from the Final Safety Analysis Reports of Surry Units 1 and 2, Turkey Point Units 3 and 4, and H. B. Robinson Unit 2. These units are all three loop PWRs (similar reactor power levels) related technically to BVPS-1.

Surry, Turkey Point and H. B. Robinson are currently in operation.

1.4.1 Design Developments Since Receipt of the Construction Permit

BVPS-1 has been built essentially in conformance with the PSAR and amendments and supplements thereof. However, during the development of the detailed design of the station, the following features have been modified:

1. ELECTRICAL AND INSTRUMENTATION:

    a. The logic for the containment high and high-high pressure signals has been changed from a 3-out-of-4 matrix to a 2-out-of-4 matrix for the high-high pressure signal and 2-out-of-3 matrix for the high pressure signal.

    b. The number of station batteries has been increased from two to five which provides for a separate battery for each vital bus and a fifth battery for non-essential loads.

    c. The number of inverters supplying the vital buses has been increased from two to four.

2. NUCLEAR:

    a. Fuel density has been increased from 94, 92 and 91 percent of theoretical to 95 percent of theoretical.

    b. Neutron absorbing material in control rods has been changed from B_4C to 5 percent Cd-15 percent In-80 percent Ag.

    c. The number of full length control rods has been increased from 45 to 48.

    d. Fuel assemblies have been changed from 15 X 15 to 17 x 17 arrangement.

3. ENGINEERED SAFETY FEATURES:

a. A jet deflector baffle has been added around the recirculation spray coolers to protect these coolers in the event of a pressurizer surge line failure.

b. The starting signals for the containment depressurization system have been modified to allow a delay of up to five minutes in actuation of the system. This feature is in response to the interim criteria for emergency core cooling in that it prevents rapid depressurization of the containment during the blowdown phase of the loss-of-coolant accident.

4. AUXILIARY SYSTEMS:

a. A wall has been added across the fuel storage pool to prevent total loss of water from the pool in the event the spent fuel cask is dropped. This wall separates the spent fuel storage area from the cask pit where fuel is loaded into the spent fuel casks.

b. A number of trip valves have been added to the component cooling water system to isolate portions of the system which are not required to maintain the station in a safe condition.

c. Auxiliary river water cooled coils have been added to the main control room ventilation system so that the control room air conditioning compressors are not required in order to maintain an acceptable temperature in control room.

d. The circulating water system for the condenser has been changed from an open cycle using Ohio River water to a closed cycle using a natural draft cooling tower. This change has resulted in a major modification to the secondary portion of the station but only minor modifications to the primary portion of the station. The addition of the cooling tower has resulted in a revision to the intake structure. The new intake structure is described in Section 9.9.

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

The Research and Development programs provided herein reflects the status of BVPS-1 related research and development at the time of the issuance of the operating license. This information is being retained for historical perspectives.

Safety Related Research and Development for Westinghouse Pressurized Water Reactors[1] presents descriptions of the safety related Research and Development Programs which are being carried out for, or by, or in conjunction with, Westinghouse Nuclear Energy Systems and which are applicable to Westinghouse pressurized water reactors.

Each safety related program is first introduced, followed, where appropriate, by background information. There is then, a description of the program which relates the program objectives to the problem and presents pertinent recent results. Finally, a backup position may be given for programs, generally experimental rather than analytical, which have not yet reached a stage where it is reasonably certain that the results confirm the expectation. The backup position is one that might be used if the results are unfavorable; it is not necessarily the only course that might be taken.

The term "research and development" as used in this section is the same as that used by the Commission in Section 50.2 of 10CFR50 as follows:

> (n) Research and development means (1) theoretical analysis exploration or experimentation; or (2) the extension of investigative findings and theories of a scientific nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, materials and processes.

The research and development discussed in the FSAR is to confirm the engineering and design values normally used to complete equipment and system designs. It does not involve the creation of new concepts or ideas.

The technical information generated by these research and development programs will be used either to demonstrate the safety of the design and more sharply define margins of conservatism or to lead to design improvements.

Progress in these development programs will be reported semiannually. New safety related research and development programs, which include existing programs which become safety related, will also be described. Included in the overall Research and Development effort are the programs below which are applicable to the 17 x 17 fuel assembly. The test programs are scheduled for completion during 1974 in order to support the initial loading of the first 17 x 17 fuel scheduled for early 1976.

1.5.1 Programs Required for Issuance of Operating License

There are no remaining research and development programs required for station operation. Previously, two programs had been identified as required for station operation. These are the core stability evaluation and fuel rod burst programs. The research and development has been completed for both programs.

The only remaining programs are those that will be used to demonstrate the margin of conservatism of the design. These are the 17 x 17 fuel assembly verification programs and the LOCA heat transfer tests (17x17) which are summarized in Sections 1.5.4 and 1.5.5 below.

Core stability Evaluation

The purpose of this program was to establish means for the detection and control of potential xenon oscillations and for the shaping of the axial power distribution for improved core performance.

The research and development portions of this program have been completed, as discussed below. The development program for power distribution control is divided into four general areas, namely:

1. Confirmation of the capability of the out-of-core detector system to indicate axial and diametral gross core power distributions sufficiently to permit control of xenon oscillations within specified operating limits.

2. Development of a control system utilizing the out-of-core detector system for axial power shaping with part-length control rods (such a system is used in the Robert Emmett Ginna, Indian Point Unit 2, and all subsequent Westinghouse reactors). Note: The part length control rods have been removed from the reactor control system. Reference to the part length control rods for axial power shaping has been retained for historical perspectives of the original design.

3. Verification that the control system discussed in Item 2 above can control the core power distribution during startup tests at other Westinghouse reactors.

4. Verification that adequate margins exist to operate BVPS-1 at the license power rating by measurements taken during the operation of other Westinghouse reactors.

The research and development phases of this program (Items 1 and 2 above) have been completed and the results have been summarized in Reference 13. The remaining work (Items 3 and 4) will be evaluated on a continuing basis for Westinghouse reactors going into operation prior to the BVPS-1. These include Donald C. Cook No. 1 and 2 (Docket Nos. 50-315 and 316), Zion No. 1 and 2 (Docket Nos. 50-295 and 304), Diablo Canyon No. 1 (Docket No. 50-275), and the H. B. Robinson Unit No. 2.

Safe operation at the design power level depends upon experimental demonstration, at the time of the BVPS-1 startup, that the actual power shapes at full power are no worse than those used in the calculation of core integrity. Further, the analytical model used to predict these power shapes will have been justified by these and earlier measurements so that a calculation of margin to design limits in a transient or accident situation can be made conservatively. However, it is clear that very similar conditions will exist on earlier plants and that very little, if any, extrapolation will be required.

In the unlikely event that the verification phases described above do not show that margins for operation at the proposed power levels are adequate, the margins designed for the BVPS-1 could be achieved by systems modifications or restrictions on operation.

Fuel Rod Burst Program (Item 2 in Reference 1, p.2)

Fuel rod burst program, a study of the performance of Zircaloy cladding under simulated LOCA conditions, has been completed for the 15 x 15 fuel assembly geometry. It has supplied empirical data from which the effect of the geometric distortion on the ability of the ECCS to meet the LOCA design criteria has been determined using present analytical design techniques. The program included burst and quench tests on single rods and burst tests on rod bundles. As a result of single rod tests, specific design limits have been established on peak clad temperature and allowable maximum metal-water reaction to ensure effective core cooling. The multi-rod burst tests demonstrated that even when rod-to-rod contact does occur after burst, the remaining flow area is always sufficient to ensure adequate core cooling.

The program is complete and results are satisfactory. Additional single rod burst test verification testing for the 17 x 17 fuel assembly geometry is discussed below in Section 1.5.4.3.

1.5.2 Other Areas of Research and Development Not Required For Issuance of Operating License

Inpile Fuel Densification

Recent operating experience with uranium dioxide fuel has indicated that the fuel may densify under irradiation to a greater density than that to which it was manufactured. This densification can lead to shorter active fuel stack lengths, increased initial rod-to-clad radial gaps and pellet-to-pellet axial gaps. The shorter fuel stack lengths gives rise to a small increase in overall average linear power density (kW/ft). Increased radial gap dimensions result in reduced gap conductance and lead to higher pellet temperatures. Axial gaps give rise to local power peaking due to decreased neutron absorption.

Westinghouse fuel densification research is directed toward producing fuel with a structure which minimizes inpile densification (hereafter called stable fuel). The objective of the program is to define material characteristics and manufacturing processes which lead to stable fuel. Any residual effects of densification will be evaluated on an appropriate model developed in this program. A more detailed description of the program and results and all other areas of research and development, not required for issuance of the operating license are presented in Reference 1.

1.5.3 Other Areas Specified by the AEC Staff, ACRS, and ASLB

The following research and development activities outside Westinghouse scope have been identified in the issuance of the BVPS-1 construction permit.

Control of Hydrogen Generation

Hydrogen accumulation in the containment atmosphere following the design basis accident is the result of production from several sources. The potential sources of hydrogen that have been identified are the zirconium-water reaction, corrosion of materials of construction and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis, is appraised from two aspects, core solution radiolysis and sump solution radiolysis. Collectively, this analysis of the various hydrogen sources provides a conservative prediction of hydrogen production following the Design Basis Accident.

The quantity of zirconium which reacts with the core cooling solution will, of course, depend on the functioning of the emergency core cooling system. System analysis has shown that the core cooling initiation is sufficiently rapid to limit the zirconium - water reaction to a maximum of less than one percent. For the reference case, however, five percent of the fuel cladding has been conservatively assumed to react with the core cooling solution.

Hydrogen recombination is presently regarded as a sufficient means of controlling hydrogen concentrations that might accumulate following a possible loss-of-coolant accident in BVPS-1. The analysis of Section 14 demonstrates that, for assumptions that are regarded as reasonably conservative, the concentration could be maintained below the lower flammability limit with hydrogen recombination. Furthermore, the analysis presents a method of using both recombiners and containment venting.

The design flow basis for the post DBA hydrogen control system is presented in Section 14. The system description is given in Section 6.5. Influent DBA operating conditions (saturated, hydrogen-air mixture containing traces of radiolytic halogens) have been simulated and evaluated by the hydrogen recombiner vendor and independent consultants. (References to other sections in this paragraph are in a historical context. Hydrogen recombiners and purge blowers are no longer required. Refer to current Section 6.5.1.)

<u>Testing of Emergency Diesel Generator</u>

Tests have been conducted at the diesel generator manufacturers plant to confirm the adequacy of the emergency diesel generator design with respect to starting times on simulated loading sequences. Part of the preoperation and startup programs for BVPS-1, as discussed in Section 8, is to demonstrate the ability of the emergency diesel generators to assume their necessary loads in the prescribed time interval in a sequential manner. The Technical Specifications list the inservice testing programs for the emergency diesel generators.

1.5.4 Verification Tests (17 x 17)

The design of the reactor described herein uses a 17 x 17 square array of fuel rods and thimbles in a fuel assembly and is conceptually similar to, but geometrically different from the 15 x 15 array used in previous designs. The 17 x 17 design is considered to be a relatively small extrapolation of the 15 x 15 design. Comprehensive testing has been planned, however, to verify that the extrapolation is sufficiently conservative. Preliminary evaluation of the data obtained to date has not revealed any anomalies.

Design changes, if necessary, will be made to the reference 17 x 17 hardware in the unlikely event that any of the experimental results fall outside the conservative design values used in analysis.

Westinghouse maintains that no plant need be designated a prototype and instrumented to verify the 17 x 17 fuel design. The change in flow induced vibration response of the internals from a 15 x 15 to a 17 x 17 fuel design will be minimal for the following reasons:

1. The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod drive line.

2. The guide tube is rigidly attached at the upper core support plate only. The upper core plate serves only to align the guide tubes. Because of this type of support arrangement the guide tube has a minimal contribution to the vibrational response of the core barrel and other internals components.

3. The effective flow area of the 17 x 17 guide tube is essentially the same as that of the 15 x 15 and therefore there are no significant differences in the flow distribution in the upper plenum.

4. The differences in mass and spring rate between the 15 x 15 and 17 x 17 fuel assemblies are very small (approximately three percent). This ensures that the effects of the fuel on the vibrational response of the reactor internals will remain essentially unchanged. The preoperational hot functional flow testing presented in Chapter 13 is considered the most conservative test condition since higher flow rates exist.

5. More adequate and meaningful tests to verify the change from 15 x 15 to 17 x 17 would be to test the new guide tube and fuel assembly designs individually in a special test facility such as the loop test facilities at the Westinghouse Forest Hills site. This type of program is in fact being conducted and is discussed below.

Some of the verification work described herein was conducted using 17 x 17 assemblies of seven grid design whereas, the selected 17 x 17 assembly design has eight grids. Table 1.5-1 provides the 17 x 17 tests which utilized a seven grid geometry and the effect of adding an eighth grid.

Table 1.5-1 shows that: (1) additional design changes are not required (e.g. no new fuel assembly holddown spring) due to the addition of a grid and (2) seven grid test information can be used to assess the adequacy of the eight grid design. Additional testing to specifically investigate the eight grid assembly is not required.

1.5.4.1 Rod Cluster Control Spider Tests

Test Purpose and Parameters

The 17 x 17 Rod Cluster Control (RCC) spider is conceptually similar to, but geometrically different from the 15 x 15 spider. The 17 x 17 spider supports 24 rodlets (the 15 x 15 design support 20) with no vane supporting more than two rodlets (same as the 15 x 15 design). The RCC spider tests verified the structural adequacy of the design.

The spider vane to hub joint was tested for structural adequacy by: (1) vertical static load test to failure and (2) vertical fatigue test to approximately three million steps. The static load test was performed by applying tensile and compressive loads to the spider. The load was applied parallel to the spider hub and reacted between the spider hub and fingers. The spider fingers shared the load equally. The number of cycles for the fatigue test was determined from the expected number of steps a control rod drive mechanism would experience during 20 years in a load follow reactor ($1.5x10^6$ steps). The test met the recommended cyclic test requirements of the ASME Boiler and Pressure Vessel Code Section III, Appendix II, paragraph 1520.

The spring pack within the spider hub was tested to determine the spring load-deflection characteristic as a function of the loading cycles seen by the spring. The test was terminated after one thousand cycles compared to a 400 cycle (rod drop) design value. The test loads were equal to and greater than that predicted to result in spring material yielding. These loads were in excess of the design values. The test acceptance criterion was for the spring to retain adequate preload after the repeated cycling.

Facility

The 17 x 17 spider tests were performed at the Westinghouse Engineering Mechanics Laboratory (Section 1.5.6.2.15).

Status

Spider tests have been completed. A vertical static load test approximately seven times the design dynamic load did not result in spider vane to hub joint failure. A spider was tested to 2.8 x 10^6 steps without failure. The spider loading was 110 percent of the design value for 1.8 x 10^6 cycles and 220 percent of the design loading for lx10^6 cycles. Design load is 3600 pounds compression and 1800 pounds tension. The spring test resulted in negligible preload loss.

1.5.4.2 Grid Tests

Test Purpose and arameters

The 17 x 17 grid is conceptually similar but geometrically different from the 15 x 15 "R" grid. The purpose of the grid tests is to verify the structural adequacy of the grid design.

Load-deflection tests have been made on the grid spring and dimple. Grid spring radial (normal) stiffness and the grid dimple radial and tangential stiffness were obtained. This information was used to verify that the fuel rod clad wear evaluation has been based on conservative values of these parameters. The fuel rod wear evaluation is conservative as shown by the flow test results presented in Reference 10.

The grid buckling strength has been determined from tests. The grid test specimens had short sections of fuel tubing inserted in the cell in place of fuel rods. These tests are used to verify that grid buckling during a postulated seismic occurrence does not interfere with control rod insertion.

The grid buckling strength is defined as the maximum load that can be applied without failure. In the case of static tests, the applied load, which is deflection controlled, results in an elastic buckling failure since no permanent deformation is experienced on removing the load. The static test established the lower limit for grid failure. The grid dynamic buckling strength is also defined as the maximum load resulting from an impact, however, some localized permanent deformation occurs before the maximum load is attained.

The grids were tested under both static and dynamic loads. The loads were applied uniformly to the face of the outside strap, transmitted directly through the grid and reacted at the grid face opposite the input. Descriptions of the grid impact test and the analytical use of the test parameters are also given in Reference 9.

Facility

The grid tests were conducted in the Westinghouse Forest Hills Engineering Mechanics aboratory (Section 1.5.6.2.15).

The grid tests have been completed. Test results are in agreement with pretest design values. The test results, along with fuel assembly structural test results, are being factored into the seismic analysis.[9]

1.5.4.3 Fuel Assembly Structural Tests

Test Purpose and Parameters

The 17 x 17 fuel assembly tests were performed to determine mechanical strength and properties. The fuel assembly parameters obtained were as follows:

1. Lateral and axial stiffness

2. Impact and internal structural damping coefficients

3. Vibrational characteristics and

4. The lateral and axial impact response for postulated accident loads.

The parameters obtained from the lateral dynamic tests are used for seismic analysis, while those obtained from the axial tests are incorporated on the loss-of-coolant (blowdown) accident analysis. The remaining tests are primarily to demonstrate that the assembly has sufficient mechanical strength to preclude damage during shipment, normal handling and normal operation.

The fuel assembly is subjected to both lateral and axial loads to obtain the respective static axial and lateral stiffnesses. The information obtained from these tests are used to establish parameters primarily for accident analysis since these conditions appear limiting. The axially applied loads, which were well in excess of shipment, normal handling and normal operational design loads, did not result in any fuel assembly permanent deformation or damage.

Lateral tests were accomplished with both nozzles fixed in place and forces applied to various grids. The lateral stiffness is found by incrementally increasing and decreasing the static load.

The fuel assembly was tested in a vertical position using core pins to simulate reactor support conditions. An electrodynamic shaker was attached to the center fourth grid to provide excitation. The fuel assembly mode shapes and corresponding natural frequencies were obtained from displacement transducers. A comparison of analytical and experimental results is given in Reference 9. Experimental vibrational studies of individual fuel rods were also performed. The rods were tested under simulated fuel assembly support condition and as assembled in a prototype fuel assembly. The information obtained from these tests included the fundamental frequencies and mode shapes. A general test description and a summary of the results is presented in Reference 10.

The fuel assembly axial stiffness was found by incrementally increasing the static load (compressive) and then incrementally decreasing the static load.

Lateral impact tests were performed by displacing the center of the assembly with the nozzles fixed in place. The assembly was released and allowed to impact on lateral restraints at each of the five center grid locations.

The axial impact response and damping were found by dropping the fuel assembly from various heights. The axial impact test was performed with the fuel assembly in the upright position.

The relevant parameters measured during the lateral and axial impact tests are as follows:

1. The impact duration versus impact load
2. Impact force versus drop height or initial displacement
3. Impact damping or restitution as a function of impact force.

A general description of the test procedure, including a description of use of the parameters as related to accident analysis is presented in Reference 9.

There is a general axial test buckling criterion which does not allow local buckling of components which could preclude control rod insertion during an accident. The fuel assembly overall buckling and component local buckling is checked during the axial static and dynamic tests. The lateral displacement associated with the fuel assembly overall (beam type) buckling is constrained by the reactor internals and therefore does not reduce the fuel assembly ultimate strength. Local component buckling was not experienced during either the static or dynamic tests for loads well in excess of the design values. The general acceptance criteria was not violated.

Facility

These tests were conducted at the Westinghouse Engineering Mechanics Laboratory (Section 1.5.6.2.15).

Status

The fuel assembly structural tests have been completed. The fuel assembly structural test results are factored into the seismic and blowdown analyses.[9]

1.5.4.4 Guide Tube Tests

Test Purpose and Parameters

A new rod cluster control guide tube is being designed to accommodate the 24 rodlet pattern adopted for 17 x 17 cores and which is sufficiently strong to assure that the support column function is not impaired and to provide increased margins of safety over present guide tubes. A high degree of interchangeability of parts has been designed into a new guide tube design. The main features of the new design are full length enclosures and cylindrical upper guide tubes.

The 17 x 17 rodlet pattern reduced the central area available for drive line passage significantly, thus necessitating a generally tighter design of the rod guidance elements.

To verify the structural adequacy of the new guide tubes, an extensive series of tests will be conducted to determine guide tube deflection with simulated blowdown forces comparable to those expected during a loss-of-coolant accident and to determine the maximum acceptable deflection which assures insertion of a control rod by free fall. Additional tests will be conducted to determine fatigue strength, displacement as a function of strain and the natural frequencies of the guide tubes for use in dynamic analysis. The following guide tube tests are considered as engineering tests. These tests, which are used as design tools and are not specifically required for demonstration of plant safety, are described below:

1. Upper Internals Scale Model Flow Test

2. 17 x 17 Guide Tube Dynamics Characteristics Test

3. Full Scale Flow Distribution Test

4. Guide Tube Drop and Deflection Test

5. 17 x 17 Guide Tube Fatigue Tests (Full Scale).

Upper Internals Scale Model Flow Test

In this series of tests, a one-seventh scale model of the upper internals is employed to determine the radial and lateral induced vibration and flow forces on the guide tubes and support columns. This test will be conducted in the Westinghouse Test Engineering Laboratory H-Loop (Section 1.5.6.2.5).

17 x 17 Guide Tube Dynamics Characteristics Test

This test will determine the strain versus displacement characteristic and the natural frequency in both air and water. This test is conducted in the Westinghouse Test Engineering Laboratory Autoclave Pit (Section 1.5.6.2.11).

17 x 17 Guide Tube Fatigue Tests

The fatigue strength will be determined for the new guide tube design. Tests are being conducted in the Westinghouse Test Engineering Laboratory Autoclave Pit (Section 1.5.6.2.11).

Full Scale Flow Distribution Test

Using one full scale 17 x 17 guide tube (96 inch style) and support column, tests will be conducted to determine the flow distribution and pressure balance between the guide tube and support column during normal operation and during blowdown. The normal operation tests will be conducted in the H-Loop and the blowdown tests in the E-Loop, of the Westinghouse Test Engineering Laboratory Facility (Sections 1.5.6.2.5 and 1.5.6.2.3 respectively).

Guide Tube Drop and Deflection Test

In a simulated blowdown condition, tests will be conducted to determine the guide tube deflection and control rod drop time. In addition, the guide tube deflection which prevents control rod drop will be determined. These tests are to be conducted at the Westinghouse Test Engineering Laboratory Facility.

Status

Upper Internals 1/7 Scale Model Flow Test

Support column and guide tube calibration tests have been completed, the model assembled and data collection has begun. Approximately 50 percent of the data required to complete the initial phase of testing has been acquired.

Preliminary data analysis shows that none of the measured forces exceed the maximum calculated forces.

17 x 17 Guide Tube Dynamics Characteristics Test

Dynamic tests to determine mode shapes and natural frequencies have been completed. The first beam mode natural frequency is in good agreement with the calculated value.

17 x 17 Guide Tube Fatigue Tests

A fatigue test is in progress. The guide tube has been vibrated for 10^7 cycles at the initial fatigue test vibratory amplitude and for 10^6 cycles at twice the initial test amplitude with no damage detected. Testing will be continued at higher amplitudes until damage is detected.

Full Scale Flow Distribution Tests

The flow distribution tests are in progress. Preliminary analysis shows that the present hardware provides satisfactory flow distribution to the top of the core.

Guide Tube Drop and Deflection Tests

Static load tests to measure strains and deflections for several support conditions were completed.

1.5.4.5 Prototype Assembly Tests

The purpose of these tests is to demonstrate that the 17 x 17 fuel assembly and control rod hardware designs will perform as predicted. Two prototype assemblies will be sequentially tested in order to obtain the required experimental data. A single set of control rod hardware, including driveline, will be used in the tests. The fuel assemblies will be subjected to flow and system conditions covering those most likely to occur in a plant during normal operation as well as during a pump overspeed transient. Seismic testing is not included in the test sequence.

These tests will be used to verify the integrated fuel assembly and RCC performance in several areas. Data to be obtained included pressures and pressure drops throughout the system, hydraulic loadings on the fuel assembly and drive line, control rod drop time and stall velocity, fuel rod vibration and control rod, drive-line, guide tube and guide thimble wear during a lifetime of operation.

Specifically, two full-size 17 x 17 fuel assemblies, one control rod, drive shaft and control rod drive mechanism will be installed and tested in the 24 inch ID by 40 foot high D-Loop at the Westinghouse Test Engineering Laboratory Facility.

Fuel Assembly Life Test (Phase I)

The first fuel assembly was subjected to the maximum expected control rod travel during fuel irradiation. The nominal test conditions were a flow velocity based on the design flow rate, temperature of 585°F and pressure of 2000 psig. These conditions represent an extreme set of conditions.

Using a fully instrumented 17 x 17 prototype fuel assembly, guide tube and RCC drive assembly, tests were conducted in the D-Loop to obtain information on the following:

1. Evaluate mechanical integrity and performance
2. Determine drop time
3. Fuel rod vibration
4. Control rod velocity
5. Hydraulic lift force
6. Guide thimble dashpot pressure.

Following this, the prototype fuel assembly underwent a complete post test evaluation and the guide tubes and drive line were inspected for any abnormal wear conditions. The purpose of this test was basically to determine the effect of the 17 x 17 fuel assembly and control rod configuration on Items 1 through 6 in Phase I (Fuel Assembly Life Test) and Items 1 through 5 in Phase II (Guide Tube and RCC Life Test). The effect on control rod drop due to a seismic disturbance is evaluated analytically for each specific plant.

The test procedures, conditions and results for Phase I are described in Reference 10.

Guide Tube and RCC Life Test (Phase II)

The second fuel assembly will then be installed to continue the test at the same flow and temperature until 3,000,000 total steps of the drive mechanism are accumulated. For Phase II, testing will be run at temperatures between 100°F and 585°F and at flow rates from 50 percent to 180 percent of the design flow rate.

The test includes a program of control rod drops and mechanism stepping that approximates the drive-line duty for the design lifetime of an operating plant. Approximately, 750,000 mechanism steps and 110 control rod drops will be accumulated. The components will then be inspected. Following inspection, testing will be continued until a total of about 3 x 10^6 mechanism steps and about 430 control rod drops have been accumulated over all of the Phase II testing.

These tests will be directed toward:

1. Life wear evaluation
2. Drop time
3. Stepping forces in drive line
4. Rod stall
5. Guide tube strains.

When about one-third of the life test is completed, the test assembly will be inspected to determine guide tube and drive line wear characteristics. This inspection will be repeated at the end of the test.

Facility

The above testing is conducted in the Westinghouse Test Engineering Laboratory Facility (Section 1.5.6).

Status

Phase I of the D-loop testing has been completed. The results of the testing are given in Reference 10.

1.5.4.6 Departure from Nucleate Boiling (DNB)

Purpose and Parameters

The effect of the 17 x 17 fuel assembly geometry on the DNB heat flux has been determined experimentally and has been incorporated in a modified spacer factor for use with the W-3 correlation. The effect of cold wall thimble cells in the 17 x 17 geometry has also been quantified.

A similar program was conducted to quantify the DNB performance of the R-type mixing vane grid as developed for the 15 x 15 fuel assembly design(2)(3). The results of that program were used to develop a modified spacer factor which quantifies the power capability associated with the use of the R-type mixing vane grid as well as the change in power capability due to the axial spacing of the grids. The modified spacer factor, along with the W-3 correlation with the cold-wall factor, was shown to be applicable to cold-wall thimble cells in the 15 x 15 geometry(3).

The experimental program consisted of three test series employing rod bundles which are representative of the 17 x 17 fuel assembly geometry. Two of the tests employed all heated rods; one test section being eight feet long and the other being fourteen feet long. The third test had one simulated cold-wall thimble tube. All three tests employed a uniform axial heat flux. The applicability of DNB data obtained using a uniform heat flux to a non-uniform heat flux has been well established by use of an axial flux shape factor. Tong(5) first developed the form of the factor. This same form with some minor change in the empirical constants has been confirmed by Wilson.(6) This method of analysis has proven correct for non-uniform rod bundle data as shown by Rosal(7), Motley(2) and Wilson(6).

The concern over comparison of uniform and non-uniform axial heat flux in long bundles is addressed in Reference 11. This compares the 0.422 inch rod diameter uniform axial heat flux data to previous non-uniform axial heat flux data. This should provide a suitable basis for 17 x 17 DNB evaluation for all axial heat fluxes.

However, in order to accumulate additional data for non-uniform axial power distributions, an additional test series is planned. This test program will consist of two test series employing a typical cell and a thimble cold-wall cell test section, respectively. The test section and range of test conditions will be the same as those investigated in the 5 x 5, 0.374 inch OD 14 ft uniformly heated length tests of Reference 11. The main difference will be that the 14 ft uniform axial heat flux heater rods will be replaced with non-uniform axial heat flux rods with a chopped cosine distribution. The axial grid spacing will probably reflect the addition of another grid bringing the spacing to approximately 20 inches compared to 26 inches in Reference 11.

This test series is scheduled for completion in the 4th quarter of 1974.

Facility

These tests were conducted in the high temperature and high pressure loop that was constructed by Westinghouse at the Columbia University Heat Transfer Laboratories. Table 1.5-2 provides the loop characteristics of this facility.

The 17 x 17 DNB tests have been performed parametrically for various combinations of inlet temperature and flow rate by increasing the bundle power incrementally until DNB occurs.

Status

The original DNB test program is completed and the results are reported in Reference 11.

1.5.4.7 Incore Flow Mixing

Test Purpose and Parameters

In the thermal-hydraulic design of a reactor core, the effect of mixing or turbulent energy transfer within the hot assembly is evaluated using the THINC code. The rate of turbulent energy transfer is formulated in the THINC analysis in terms of a thermal diffusion coefficient (TDC).

A program[4] to determine the proper value to TDC for the R-type grid vane, as used in the 15 x 15 fuel assembly design, has been completed and showed that a design value of 0.038 (for 26 inch spacing) can be used for TDC. These results also showed that TDC was independent of Reynold's number, mass velocity, pressure, and quality over the ranges tested.

A new TDC experimental program employed a geometry typical of the 17 x 17 fuel assembly to determine the effects of the geometry on mixing and to determine an appropriate value for TDC. A uniform axial heat flux was used. There is no analytical reason to expect that the mixing coefficient would be affected by a non-uniform axial heat flux. The THINC computer code considers the mixing in each increment along the heated length and within that increment the heat flux is considered uniform. The tests reported by Cadek[8] indicate that there was no difference, within experimental accuracy, between a test section with a uniform flux (Pitt) and one half of a cosine flux (Columbia). The heat flux varied between the simulated fuel rods in the test section to create a thermal gradient in the radial direction. Using different flow rates and inlet temperatures, the TDC for the 17 x 17 geometry was determined.

Facility

These tests were conducted at the Columbia University Heat Transfer Laboratories in the equipment described above.

Status

The TDC tests are completed and the results are reported in Reference 12.

1.5.5 LOCA Heat Transfer Tests (17 x 17)

Extensive experimental programs have been completed or are in progress to determine the thermal hydraulic characteristics of 15 x 15 fuel assemblies, and to obtain experimental heat transfer data under simulated loss-of-coolant accident (LOCA) conditions.

Complementary experimental programs will be performed with a simulated 17 x 17 assembly to determine its behavior under similar LOCA conditions. The 17 x 17 test will be conducted in a new test loop which is presently being activated. Results from the 17 x 17 programs will be compared with data from the 15 x 15 assembly test programs and will be used to confirm predictions made by correlations and codes based on the 15 x 15 test results.

1.5.5.1 Facility Description

The 17 x 17 test facility will provide experimental measurements on the reflooding behavior of a 17 x 17 rod array following a LOCA. The test assembly consists of an array of 336 electrically heated rods and 25 guide tube thimbles arranged in a 17 x 17 array. The heater rod diameter, the active heated length, and pitch spacing is identical to that used in the 17 x 17 fuel. There will be eight Westinghouse production mixing vane grids in the bundle which will also improve the simulation.

The reflood test series which will be run will be similar to the FLECHT reflooding tests. There is a high temperature forced flow injection system as shown in Figure 1.5-1 which will be used for constant flooding rate tests and variable flooding rate tests. Table 1.5-3 provides the range of initial conditions which will be examined.

The bundle has a 1.66 axial cosine power shape and will use a simulated radial power shape which is representative of the hottest assembly in the core.

1.5.5.2 D^2NB Test

Test Purpose and Parameters

The "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors" was issued in Section 50.46 of 10CFR50 on December 28, 1973. It defines the basis and conservation assumptions to be used in the evaluation of the performance of Emergency Core Cooling Systems (ECCS). Westinghouse believes that some of the conservatism of the criteria is associated with the manner in which transient DNB phenomena are treated in the evaluation models. Transient critical heat flux data presented at the 1972 specialists meeting of the Committee on Reactor Safety Technology (CREST) indicated that the time to DNB can be delayed by several seconds. To demonstrate the conservatism of the ECCS evaluation models, Westinghouse has initiated an accelerated program to experimentally simulate the blowdown phase of a loss-of-coolant accident. The objective of the D^2NB test is to determine the time that DNB occurs under LOCA conditions.

The D^2NB tests, which are part of this LOCA program, will be used to confirm the predictions made with the new Westinghouse transient DNB correlation. The motivation for conducting the D^2NB test is independent of the change over to the 17 x 17 fuel and are being conducted with a 15 x 15 geometry. However, the results of the steady state DNB program, described in Section 1.5.4.6, will be used to assure that the minimal geometric difference between the 17 x 17 and 15 x 15 arrays can be correctly treated in transient correlations.

The program is divided into two phases. Phase I will provide data directly applicable to the PWR to permit definition of the time delay associated with onset of DNB. Tests in this phase will cover a range of cold and hot leg breaks, with particular emphasis on the large double-ended guillotine cold leg break. All tests in Phase I will be started upon establishment of typical steady state operating conditions. The fluid transient would then be initiated, and the rod power decayed in such a manner as to simulate the actual heat input of fuel rods. The transient will be required to follow a predetermined behavior as predicted by Westinghouse computer codes and the as-designed system hydraulics. The test would be terminated when the heater rod temperatures reach a predetermined limit (dependent on power level). Table 1.5-4 provides the parameters to be studied under Phase I testing.

The Phase II test, which will provide separate effects data to permit heat transfer correlation development, will also start from steady state conditions, with sufficient power to maintain nucleate boiling throughout the bundle. Controlled ramps of decreasing test section pressure or flow will initiate DNB. By applying a series of controlled conditions, investigation of the DNB will be studied over a range of qualities and flows, and at pressures relevant to a PWR blowdown. To obtain qualities higher than in the Phase I tests, a steam generator will be added to the unbroken loop prior to the Phase II tests. Table 1.5-5 provides the parameters to be studied under Phase II testing.

The experiments in the D^2NB facility will result in cladding temperature and fluid properties measured as a function of time throughout the blowdown range from 0 to 20 seconds.

Facility

The experimental program is being conducted in the J-Loop at the Westinghouse Test Engineering Laboratory Facility with a full length 5 x 5 rod bundle simulating a section of a 15 x 15 assembly to determine DNB occurrence under loss-of-coolant accident conditions. The schematic for the D^2NB facility is shown in Figure 1.5-2 (Section 1.5.6.2.6).

The heater rod bundles used in this program are assembled using internally-heated rods. The proprietary heater rods are designed for high reliability, long life and high power density. The maximum power is 18.8 KW/ft, and the total power is 136 KW for extended periods over the 12-foot heated length of the rod. Heat is generated internally by means of a varying cross-section, rugged, tubular resistor which approximates a U^2 cos U power distribution, skewed to the bottom. Each rod is adequately instrumented with a total of 20 thermocouples (8 inside resistor, 12 sheath thermocouples).

Status

The construction of the D^2NB facility is complete. Shakedown testing has been completed and all instruments have been calibrated. Phase I testing is presently starting and will continue to the end of 1974.

1.5.5.3 Single Rod Burst Test (SRBT)

Test Purpose and Parameters

The SRBT results are used to quantify the maximum assembly flow blockage which is assumed in LOCA analyses.

Previously, single rod and multi-rod burst test (MRBT) have been completed on 15 x 15 fuel assembly rods under conditions which exist during the loss-of-coolant accident. The conclusion of these tests were that fuel rods burst in a staggered manner so that maximum average assembly wise flow area blockage is 55 percent during blowdown and 65 percent during reflood based on the characteristics of the pressurized PWR fuel rod and the conservative peak clad temperature predicted during the LOCA transient.

The SRBT program for the 17 x 17 fuel assembly rods consists of testing specimens at two internal pressures and three heating rates in a steam atmosphere. The specific test parameters are provided in Table 1.5-6.

All specimens were then heated 5°F per second from 1940°F to about 2300°F, held for a short time and then cooled 5°F per second to 1200°F.

Metallography is done on specimens to determine the degree of wall thinning and the extent of oxygen embrittlement.

In addition, tests were run on 15 x 15 fuel assembly rods to insure reproducibility of the 1972 single rod burst test results.

Facility

The SRBTs are conducted in the Westinghouse Engineering Mechanics Laboratory in an electrically heated furnace.

Status

The SRBTs are in progress. Results of initial tests showed that the LOCA behavior of 17 x 17 clad in comparison to that of 15 x15 clad exhibited no significant differences in failure ductility.(11) Because of the result and the geometric scaling, the flow blockage (in percent) as determined by 15 x 15 MRBT simulation can be used for 17 x 17 fuel geometry.

1.5.5.4 Power-Flow Mismatch

Design emphasis has been placed on reliable and effective control and protection systems for the reactor core and engineered safety features to ensure adequate margins within which incidents can be terminated before the onset of fuel failure. For this reason, investigations into the mechanisms of fuel failure and its propagation and phenomena of molten fuel-coolant interaction have been limited.

Obtaining complete answers to the question related to fuel rod failure will require extensive testing because of the multiplicity of parameters involved. Such an extensive inpile test program has been proposed by the Aerojet Nuclear Corporation under AEC sponsorship for the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory.

The proposed PBF program is expected to simulate conditions appropriate to major accidents postulated for reactor systems, i.e., loss-of-flow, loss-of-coolant and reactivity-initiated accidents. Phase I of the proposed PBF program is expected to be completed approximately two years after facility checkout or by the end of 1975. Phase I is expected to concentrate upon establishing the thresholds for, and the consequences of, fuel failure and utilizes fuel rod clusters of various sizes to determine the extent of failure propagation. This program would broaden the experimental basis for evaluating reactor safety, but should not be considered essential for the design and safe operation.

Westinghouse will closely follow any such experimental program or analytical studies that may become available. Until such information is available, clearly demonstrating that local fuel melting is an acceptable condition, the emphasis in design will continue to be placed on providing adequate margins to minimize the probability of fuel melting. The margins incorporated in the design within which incidents can be terminated before the onset of fuel failure, provide a sound basis for safe operation.

1.5.6 Westinghouse Test Engineering Laboratory Facility

1.5.6.1 Introduction

The Test Engineering Laboratory at Forest Hills, Pennsylvania, has long been the major Westinghouse center for nuclear research and development. The Test Engineering Laboratory is totally involved with the design and implementation of facilities and programs to prove the reliability of Westinghouse PWR concepts and components.

The Test Engineering Laboratory has full in-house capabilities to design and construct pressurized water reactor loops for both hydraulic and heat transfer testing programs. The most vital current project is Emergency Core Cooling Systems (ECCS), which involves scale-model tests, run on three separate facilities.

The G-Loop consists of a test vessel, which presently contains a 480 heater rod bundle, the largest such test facility in the world. It also has a steam supply to provide the proper environment during system blowdown, and the capability to test high-pressure and low-pressure ECCS. G-Loop operates at pressures up to 2000 psi and temperatures up to 650°F. It is designed to start operation at eight seconds after a loss-of-coolant accident (LOCA), and is presently capable of investigating the current ECCS, Upper-Head Injection and other spray systems.

J-Loop consists of a test vessel, which contains a 25 heater rod array, a broken loop simulation and an unbroken loop simulation. The loop is designed to operate at 2500 psi and 650°F, and is capable of simulating the first 20 seconds of a LOCA with primary emphasis on D^2NB.

FLECHT-SET consists of a test vessel, which contains a 100 heater rod and thimble array, which is used to investigate the reflood phase of the current ECCS with plant system effects measured with scaled piping and two scale-model steam generators. The facility is designed to operate at up to 100 psia.

Five general purpose hydraulic loops are also involved in the development of improved water reactor components, as well as the reliability testing of current and prototype PWR components.

Historically the Test Engineering Laboratory has been in a state of transition, depending upon the current need for its services. Today's great need is for ECCS data and the verification of many new PWR system components. Past needs and accomplishments have included the development of the supercritical heat transfer once through loops, rod cluster control drive mechanisms, fuel assemblies, underwater handling tools and fuel assembly grid design among many other earlier projects. Testing has included air filter tests, water chemistry tests, in-pile testing for fuel rods, single fuel rod burst tests, hydraulic studies on fuel assemblies and corrosion testing of Zircaloy and other PWR components and materials with and without heat transfer.

The Test Engineering Laboratory is a very flexible installation, one which will continue to expand and develop as future needs for its services arise. Its staff varies according to requirements.

There are currently more than 100 persons involved in Laboratory projects, including 12 electrical and mechanical engineers, more than 75 highly skilled technicians and some 30 specialists from other divisions of Westinghouse. The Test Engineering Laboratory has the option of obtaining personnel from the entire Corporation, depending upon the need for specific skills, knowledge and experience.

Ongoing research performed at the Test Engineering Laboratory continues to demonstrate the reliability of Westinghouse PWR plant components and greatly facilitates the development of improved reactor system components. As the test center for Westinghouse Nuclear Energy Systems, the Test Engineering Laboratory is totally committed to the advancement of the nuclear energy industry.

1.5.6.2 Test Loops and Equipment

This section contains a brief description of the major test loops and test equipment at the Westinghouse Test Engineering Laboratory Facility.

1.5.6.2.1 A & B-Loops: Low-Flow/High-Pressure Hydraulic Facilities

These loops are small, high pressure, stainless steel facilities used for testing small components and individual parts of larger components under normal working conditions. A canned motor pump circulates water in both A-Loop and B-Loop at 150 gpm. Operating temperatures are obtained from the conversion of the pumping power into heat, as well as from external heaters. Typical tests run in these loops are: 1) full-scale gate and check valves, 2) material corrosion-erosion, with variable water chemistry, and 3) corrosion product release and transport properties of crud. The characteristics of the A & B-Loops are provided in Table 1.5-7.

1.5.6.2.2 D-Loop: Medium-Flow/High-Pressure Hydraulic Facility

The D-Loop is a flexible test facility used for demonstrating the interplay of reactor subsystems and evaluating component design concepts. It contains a canned motor pump, which produces a 290 ft head at 3000 gpm. All piping (10 inch Schedule 160) in contact with the primary water is stainless steel. Loop pressure is established and maintained by an air driven charging pump operating in conjunction with a gas loaded back pressure valve. Most of the power required to establish and maintain loop temperature is derived from the circulating pump operation, and 75 kW of heat is available from electric strip heaters.

The D-Loop services a 24 inch ID by 40 ft long test vessel, which accommodates full-scale models of large PWR core components for operational studies. The characteristics of the D-Loop are provided in Table 1.5-8.

1.5.6.2.3 E-Loop: Low-Flow/Low-Pressure Hydraulic Facility

The E-Loop is a low-pressure, six inch, stainless steel loop, with two circulating pumps. These pumps may be connected in parallel, giving 2000 gpm at 130 ft head, or in series, giving 1000 gpm at 260 ft head. Flow and vibration studies are conducted with this loop, and, because of its low pressure, plastic models for visual observation or photography may be used. In addition, a 4 inch Rockwell water meter in a branch line permits the calibration of flow meters up to 800 gpm. The characteristics of the E-Loop are provided in Table 1.5-9.

1.5.6.2.4 G-Loop: Emergency Core Cooling System Facility

Loop is a high-pressure, Emergency Core Cooling System (ECCS) test facility designed and fabricated for 2000 psi and 650 F in accordance with the ASME Boiler and Pressure Vessel Code Section 1. It consists of a main test section and vessel, exhaust system, piping, separators and muffler, flash chamber steam supply system, and high-pressure/low-pressure cooling system.

This loop is basically designed to obtain test data for analysis of LOCA, for breaks up to and including double-ended pipe breaks for Pressurized Water Reactors. Tests are initiated at simulated conditions existing eight seconds after the start of a LOCA. A typical run consists of constant power and pressure, followed by pressure blowdown, power decay and Emergency Core Cooling. G-Loop is capable of performing the following methods of Emergency Core Cooling: Upper-Head Injection (UHI) and other core spray systems. It may also be used for constant temperature/pressure small leg break tests (core uncovering tests). These consist of boiling off water at a constant bundle power input until the rods can no longer be cooled.

The G-Loop test bundle consists of 480 electrically heated rods, 16 grid support thimbles, and 33 spray thimbles bounded by an octagonal stainless steel baffle and arranged as per a 4 loop 15 x 15 rod bundle configuration. The loop is controlled (fully automated during transients) through a PDP-II-DEC-16K computer with a 600 point Computer Products A-D Converter operating at a sweep rate of 40,000 points/second for data acquisition. The characteristics of the G-Loop are provided in Table 1.5-10.

1.5.6.2.5 H-Loop: High-Flow Hydraulic Facility

The H-Loop constitutes a versatile hydraulic facility, capable of supplying 14,000 gpm of water at a developed head of 600 ft and at temperatures as high as 200°F. This 4 loop system can simultaneously handle either full-scale prototype test assemblies or one large-scale reactor model. The major purpose of H-Loop is to permit the use of one-seventh scale reactor models and full-scale fuel assemblies for conducting mixing studies, flow distribution studies, and similar low-temperature/low-pressure hydraulic tests. The characteristics of the H-Loop are provided in Table 1.5-11.

1.5.6.2.6 J-Loop. Delayed Departure From Nucleate Boiling Heat Transfer Facility

The J-Loop is a completely instrumented pressurized water test facility for verifying D^2NB phenomena during a LOCA, and for conducting steady state heat transfer studies. This test loop is a full-size, single-loop simulation of a typical four loop reactor system; it will accept a full-length 5 x 5 bundle of internally heated "fuel rods." J-Loop is designed to operate at 2500 psia at 650°F, and at variable flow rates of up to 450 gpm. During LOCA tests, fluid input to the "reactor vessel" is closely controlled by two servo-controlled mixers which inject a two phase water/steam mixture into the test vessel, to simulate flow from the unbroken loops. The characteristics of the J-Loop are provided in Table 1.5-12.

1.5.6.2.7 K-Loop: Boron Thermal Regeneration Test

The K-Loop, Boron Thermal Regeneration System (BTRS) test facility, is used to study the performance and to verify the component sizing of both the currently available THERM I and the improved THERM II BTR systems. The function of this system is to process boron-containing effluents from the Reactor Coolant System (RCS) to yield a high-boron concentration fraction, which can be used to borate the RCS. A relatively boron-free fraction is also processed, which can be used to dilute the RCS, such as that required in load-follow operations. The characteristics of the K-Loop are provided in Table 1.5-13.

1.5.6.2.8 FLECHT-SET: Emergency Core Cooling System Facility

The FLECHT-SET is a low-pressure facility, designed to provide experimental data on the influence of system effects on ECCS during the reflood phase of a loss-of-coolant accident (LOCA).

The facility consists of a once-through system, including an electrically heated test section ("fuel rods" and housing), accumulator, steam generator simulators, pressurizer, catch vessels, instrumentation and the necessary piping to simulate the reactor primary coolant loop. Data acquisition is accomplished through a PDP-II-DEC-16K Computer with a 256 point Computer Products A-D Converter operating at a sweep rate of 200 points/sec. The characteristics of the FLECHT-SET are provided in Table 1.5-14.

1.5.6.2.9 Single Rod Test Loop: Heater Rod Development Facility

The single-rod test loop is used to evaluate prototype heater rods and for in-depth study of isting rods in pressurized water systems. The test section of the loop is easily replaced to facilitate the installation of various length and diameter heater rods. The single-rod loop is electrically controlled and operated by one person. Steady state and blowdown at various conditions can be simulated in the loop. The main test section can be replaced with a quartz tube, and DNB phenomenon can be observed on a single rod with a remotely operated camera. The characteristics of the single-rod test loop are provided in Table 1.5-15.

1.5.6.2.10 Hydraulic Model Testing

Miscellaneous hydraulic tests on mock-ups of reactor system parts and components are routinely performed at the Test Engineering Laboratory. Typical of this type of testing are the two discussed below, which were recently completed:

<u>Emergency Core Cooling Flow Distribution</u>: As shown above, a 10 x 10 rod bundle was installed in a plastic housing with a water supply at the top. A grid-collection unit at the bottom of the bundle collected the water as it flowed through the model and diverted it to the measuring tubes at the base. Knowledge of the flow distributed in the bundle was obtained in this manner.

<u>Sample System Mixing Test</u>: This test used one thermocouple to measure the temperature of water from four locations in a reactor. The purpose of the procedure was to determine whether the indication from the single thermocouple was representative of the average temperature of the four water supplies. A mock-up of the mixing chamber was constructed so that hot or cold water, at closely controlled pressure, could be supplied to any of the four inlets. By running combinations of hot and cold inlets and making simultaneous recordings of the various temperatures, highly useful information was obtained.

1.5.6.2.11 Autoclave Testing

The Test Engineering Laboratory is equipped with autoclaves ranging in size from one-half gallon to 100 gallons. These devices are in constant use to determine the effect of various water chemistries on core components, as well as to perform corrosion tests. The units have also been used as boilers to provide steam for miscellaneous development tests, including acoustic leak detection.

1.5.6.2.12 Mechanical Component and Vibration Tests

Full-scale mechanical and vibration tests are performed at the Test Engineering Laboratory on plant and reactor components to prove the reliability of equipment design. Vibration testing of reactor components is also performed in this Laboratory, using electronically excited shaker heads. Three sizes are available (2 lbs, 50 lbs and 150 lbs) for regular scale model testing for frequencies from 5 Hz to 50 Hz.

1.5.6.2.13 Electronic Component Assembly

Highly skilled technicians are available at the Test Engineering Laboratory for constructing complex control and instrumentation systems. Work is initiated with engineering ideas and sketches and includes mounting of process controllers, recorders, meters, relay logic, protection circuits, switches and indicators.

Point-to-point wiring or PCBs are used, as required. Final "as- built" drawings are prepared, inspection and thorough electrical checkout is performed before installation in a facility.

1.5.6.2.14 Surveillance System Development

Surveillance Systems provide on-line monitoring of pressure vessels for flaws. Electronic components are being developed at the Test Engineering Laboratory for an acoustic emission monitoring system for in-service inspection of operating plant vessels and piping. This system is designed to detect and locate initiation and propagation of cracks at various locations, such as welds and stress risers. Vessel flaw growth and rupture data have been obtained through joint programs at the Idaho National Engineering Laboratory (formerly the National Reactor Testing Station) in Idaho, and at the Oak Ridge National (Laboratories). Pipe rupture data have been obtained from AEC sponsored tests, and hydrostatic test data, operational noise and attenuation characteristics have been measured at various Westinghouse operating plants.

1.5.6.2.15 Engineering Mechanics Laboratory

Bench tests are performed in fixtures designed for the particular test using standard test equipment and techniques.

References for Section 1.5

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3. F. E. Motley and F. F. Cadek, "DNB Test Results for R Grids with Thimble Cold Wall Cells": WCAP-7695-L Addendum 1, (Westinghouse Proprietary) Westinghouse Electric Corporation (October 1972), and WCAC-7958 Addendum 1, Westinghouse Electric Corporation (October 1972).

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11. K. W. Hill, F. E. Motley, F. F. Cadek, A. H. Wenzel, "Effect of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8297, Westinghouse Electric Corporation (March 1974).

12. F. E. Motley, A. H. Wenzel, and F. F. Cadek, "The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing," WCAP-8299, Westinghouse Electric Corporation (March 1974).

13. "Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Fall 1971 Spring 1972." WCAP-7856, Westinghouse Electric Corporation (May 1972).

1.6 IDENTIFICATION OF CONTRACTORS

No changes have been made to this section based on the fact that all Contractors currently performing work on BVPS-1 are controlled by the company Quality Assurance Program Manual. The list of Contractors described below were performing work on BVPS-1 prior to the issuance of the Operating License. This section is being retained for history purposes.

1.6.1 Stone & Webster Engineering Corporation

Stone & Webster Engineering Corporation (S&W) is an engineering construction firm serving the electric utility industry in the design and construction of all types of power stations. Stone & Webster has and is providing engineering services in connection with generating capacity in excess of 40,000,000 kW.

Stone & Webster began its early association with the nuclear industry in 1942 with the Metallurgical Laboratory at the University of Chicago. The Company has performed varied design and construction services. Stone & Webster was retained on the Shippingport Atomic Power Project to provide engineering design for the nuclear portion of the plant, including containment, shielding, waste disposal system, fuel handling system and to provide inspection services during plant construction.

Subsequently, S&W has had major responsibility for engineering and construction on the Yankee Atomic Electric Plant, completed in 1960; the Carolinas Virginia Nuclear Power Associates Plant, completed in 1963; the Connecticut Yankee Atomic Power Plant, completed in 1967; the Maine Yankee Atomic Power Plant, completed in 1972; the Surry Power Station Units 1 and 2, scheduled for completion in 1972 and 1973, respectively; the Shoreham Nuclear Power Station scheduled for completion in 1983; the James A. FitzPatrick Nuclear Power Plant completed 1975; and the North Anna Power Station Units 1 and 2, completed in 1978 and 1980, respectively. In addition, S&W had construction management responsibility for the Nine Mile Point Nuclear Station, completed in 1968.

1.6.2 Westinghouse Electric Corporation

Westinghouse Electric Corporation is thoroughly qualified to perform the development, design and manufacturing necessary to fulfill its contractual obligations with Duquesne Light Company in the construction of BVPS-1. Westinghouse has accumulated considerable experience in the design and construction of closed-cycle water reactors. The Westinghouse technical and manufacturing organization has been responsible for projects which require specialized knowledge and positive overall coordination. Its atomic activity dates back to 1936, three years before the discovery of atomic fission, when Westinghouse began its program of research in nuclear physics.

1.6.3 Hansen, Holley, and Biggs

This firm is composed of Professors R. J. Hansen, M. J. Holley, Jr. and J. M. Biggs. All are actively associated with the Massachusetts Institute of Technology (MIT). The firm has acted as structural design and analyses consultant on nuclear power plant work for S&W and others. The S&W nuclear power plants on which it was consulted include Yankee Atomic, Connecticut Yankee, Malibu and Shoreham.

1.6.4 NUS Corporation

NUS Corporation is retained to provide general consulting services, to perform surveys, and to prepare sections of the safety analysis report on site meteorology, area population, ecology and radiological monitoring. This corporation is a consulting engineering firm headquartered in Rockville, Maryland that serves utilities, industry and the government in the fields of nuclear engineering, environmental engineering, systems analysis and operations research, plant water technology, nuclear personnel training and manpower planning services. It numbers over eighty utilities among its clients and has provided these clients with a broad spectrum of nuclear related services including reactor safeguards analysis, reactor siting and reactor design services.

1.6.5 Weston Geophysical Research, Inc.

Weston Geophysical Research, Inc., has been retained on the Beaver Valley project to provide consulting services on seismicity. Rev. Daniel Linehan, Director of Weston Observatory, is a consultant to Weston Geophysical Research, Inc. and participated in these studies. Father Linehan is also a consultant to the U.S. Coast and Geodetic Survey and to numerous reactor projects, including those of Northeast Utilities Service Company, Connecticut Yankee Atomic Power Company, Virginia Electric and Power Company and Boston Edison Company.

Weston Geophysical engineers are pioneers in the development of shallow refraction survey techniques, especially shear wave velocity determinations, which are necessary for establishing dynamic soil moduli for use in analysis of structural response to earthquakes.

1.6.6. Whitman and Rand

Doctor R. V. Whitman, who is associated with MIT, is an outstanding authority in the field of soil dynamics and has published many papers on the subject. His studies have included significant work on amplification of earthquake motion within the overburden.

Mr. J. R. Rand was formerly Chief State Geologist for the State of Maine and assists in geology and ground water hydrology studies.

1.7 COMMON FACILITIES

The following identifies and discusses all structures, systems, subsystems and components of BVPS-1 that may be shared with BVPS-2. The discussion identifies those items that are essential in attaining and maintaining a safe shutdown as well as the considerations and protective measures taken to prevent essential systems from functionally being disabled by the failure of other essential and nonessential components. Each of the essential structures, systems and components are discussed in the appropriate Sections of the BVPS-1 and BVPS-2 Updated FSAR.

1.7.1 Identification of Shared Systems, Structures and Components

Structures, systems, subsystems and components and electrical systems to be shared between BVPS-1 and BVPS-2 that are considered nonessential because they are not required for attaining or maintaining a safe shutdown are provided in Tables 1.7-1, 1.7-2 and 1.7-3 respectively.

The sharing of emergency diesel generators between BVPS-1 and BVPS-2 during a station blackout event is discussed in Section 8.4.6.

Common structures, systems and components between BVPS-1 and BVPS-2 considered essential for attaining and maintaining a safe shutdown are discussed below.

Intake Structure

The Seismic Category I intake structure is a structure common to both BVPS-1 and BVPS-2. The BVPS-1 river water pumps and the BVPS-2 service water pumps housed in this structure, are considered essential systems and are so designed. Refer to Section 9.9 of the BVPS-1 Updated FSAR and Section 9.2 of the BVPS-2 Updated FSAR. Both the river water and service water systems are operated completely independent of each other and are designed to meet the single failure criterion. A cross-connect is provided between one of the two river water and one of the two service water discharge headers. This cross-connect is usually inoperable and is isolated from the two headers by two isolation valves. Catastrophic failure of one river water or service water pump can disable the other pump located in the same bay. However, since three 100 percent pumps are provided for the river water and the service water system and there is a cross-connect that can be used, there is no credible way that failure of one system can disable the other. The possibility of other essential and nonessential equipment failure damaging the essential river water and service water piping and pumps is discussed in Section 5.2.6 and the current high-energy pipe heat study and such a failure mode is not considered credible.

Main Control Area

The control areas for BVPS-1 and BVPS-2 are located in the same Seismic Category I missile-protected structure. However, the control boards for the individual units are physically and functionally separated within the main control area.

1.8 SPECIAL CONSIDERATIONS

1.8.1 Austenitic Stainless Steel Material Control

Austenitic stainless steel of the 3xx series are used for components that are part of the reactor coolant pressure boundary, systems required for reactor shutdown, systems required for emergency core cooling, reactor vessel internals required for emergency core cooling, and reactor vessel internals relied on to permit adequate core cooling.

This material must be suitably cleaned and protected against contaminants capable of causing stress corrosion cracking throughout the fabrication, shipment, storage, construction, testing and operation of components and systems.

It was required that all austenitic stainless steel materials used in the fabrication, installation and testing of nuclear steam supply components and systems be handled, protected, stored and cleaned according to recognized and accepted methods and techniques. The rules covering these controls were stipulated in Westinghouse Electric Corporation process specifications. These process specifications supplemented the equipment specification and purchase order requirements of every individual austenitic stainless steel component or system which Westinghouse procures for a nuclear steam supply system, regardless of the ASME Boiler and Pressure Vessel Code classification.

Every possible effort was made to assure that Manufacturers and Installers adhere to the rules in these specifications. This was accomplished through actual surveillance of operations by Westinghouse personnel either in residence at the Manufacturer's plant and the Installer's construction site or, when residency was not practical, during periodic engineering and quality assurance visitations and audits at these locations.

The discovery of any deviation from these rules, whether it be during the "act" or as the result of a subsequent "material-rejection," required corrective measures to eliminate the condition or replacement of the material and/or component.

The process specifications which establish these rules and which were in compliance with the American National Standards Institute, ANSI N-45 Committee specifications are as follows:

1. Requirements for Pressure Sensitive Tapes for Use on Austenitic Stainless Steels

2. Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment

3. Requirements for Marking of Reactor Plant Components and Piping

4. Site Receiving, Inspection and Storage Requirements for Systems, Material and Equipment

5. Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Material

6. Packaging and Preparing Nuclear Components for Shipment and Storage

7. Cleaning and Packaging Requirements of Equipment for Use in the NSSS

8. Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures

9. Cleanliness Requirements During Storage, Construction, Erection, and Startup Activities of Nuclear Power Systems.

Controls were specified during fabrication, construction and operation to minimize exposure of austenitic stainless steel surfaces to contaminants that could lead to stress corrosion cracking. Halogen-bearing compounds were avoided or halogen concentration restrictions imposed for shop and field cleaning procedures, packaging and handling. Precautions were specified for keeping components protected and dry during shipment and storage. Water chemistry controls were required for flushing, testing and operations to minimize the possibility of stress corrosion cracking. Either stainless steel insulation or non-metallic insulation which had been tested in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel", was used on austenitic stainless steel to minimize the possibility of stress corrosion cracking.

Special cleanliness requirements for Nuclear Steam Supply System electrical components, instrumentation, core materials and reactor vessel internals were incorporated into the process specifications.

A specification for cleaning and maintaining cleanliness during construction was prepared for the BVPS-1. The specification defines cleaning requirements and criteria for components and systems.

Cleanliness requirements were expressed for each of three grades (for the balance of plant). The requirements which must be met to comply with a cleanliness classification were to be specified for a particular component or surface in the cleaning specification.

It was the purpose of the cleaning and cleanliness control provisions to minimize post-assembly cleaning of parts and components and minimize preoperational cleaning of components and systems after installation. This required that components, parts and subassemblies be finished cleaned while still in simple geometric form, i.e., without deep internal crevices, undrainable or unflushed spaces or uninspectable surfaces and required that they be carefully protected to prevent recontamination until they were properly packaged, shipped, stored and are ready for installation, and that contamination prior to the final cleaning after field fabrication was minimized.

The three grades of cleanliness are:

1. Grade A was the final preoperational cleanliness of stainless steel primary components and systems required to have an exceptionally high degree of cleanliness and a very low level of chloride contamination. Grade "A" surfaces shall be free of dust, scale, organic film, soil, grease, oil, rust, preservative, sand, grit, or any other contaminant.

2. Grade B clean was the final preoperational cleanliness of non-corrosion and corrosion resistant primary and secondary components and systems required to have a good degree of cleanliness with a low level of chloride contamination. Surfaces shall be clean as in Grade "A" except that light rusting of carbon steel or non-corrosion resistant material and tarnishing of copper base materials, was allowable. Light dust and tap water residue were also acceptable contaminants.

3. Grade C clean was the final preoperational cleanliness of surfaces and components required to be free of all loose particulates, grease, oil, flux, and other contaminants that could adversely affect system operation. Acceptable contaminants were adherent films of rust, scale, tarnish, dust and tap water residues.

Table 1.8-4 summarizes the water requirements for the above cleanliness grades:

Heat Treatment

All of the austenitic stainless steels listed in Tables 1.8-1, 1.8-2 and 1.8-3 were procured from raw material producers in the final heat treated condition required by the respective ASME Boiler and Pressure Vessel Code Section II material specification for the particular type or grade of alloy.

Material Inspection Program

All of the wrought austenitic stainless steel alloy materials which required corrosion testing in the final heat mill treatment were tested in accordance with ASTM A-262(1) (previously ASTM A-393) using material test specimens obtained from specimens selected for mechanical testing. These materials were obtained in the solution annealed condition.

Welding and Sensitization

The unstabilized austenitic stainless steels used for core structural load bearing members and component parts of the reactor coolant pressure boundary were processed and fabricated using the most practicable and conservative methods and techniques to avoid partial or local severe sensitization.

After the material was heat treated during the final mill heat treatment, the material was not heated above 800 F during subsequent fabrication other than instantaneously and locally by welding operations.

Methods and material techniques that were used to avoid partial or local severe sensitization are as follows:

1. Nozzle Safe Ends

   Use of a stainless steel weld deposit containing more than five percent ferrite.

2. All welding is conducted using those procedures that have been approved by the ASME Boiler and Pressure Vessel Code Sections III and IX.

3. All welding procedures have been qualified by non-destructive and destructive testing according to the ASME Boiler and Pressure Vessel Code Sections III and IX.

4. Table 1.8-5 provides a listing of welding methods that have been tested individually and in multiprocess combinations as outlined in the Welding and Sensitation Section, Item number 3, using prudent energy input ranges for the respective method.

5. The interpass temperature of all welding methods is limited to 350°F maximum.

6. All full penetration welds require inspection in accordance with the ASME Boiler and Pressure Vessel Code Section III, Article NB5000. Welding materials are required to conform and are controlled in accordance with ASME Boiler and Pressure Vessel Code Section III, Article NB2400.

When these welding procedure tests were being performed on test welds that were made from base metal and weld metal materials which were from the same lot(s) of materials used in the fabrication of components, additional testing was frequently required to determine the metallurgical, chemical, physical, corrosion, etc., characteristics of the weldment. The additional tests that were conducted on a technical case basis are as follows:

1. Light and electron microscopy

2. Elevated temperature mechanical properties

3. Chemical check analysis

4. Fatigue tests

5. Intergranular corrosion tests using ASTM A-262 "Susceptibility to Intergranular Attack in Stainless Steels, Rec. Practices for Detecting", Practice E

6. Static and dynamic corrosion tests within reactor water chemistry limitations.

<u>Post Weld Heat Treatment and Chemistry Control</u>

The unstabilized austenitic stainless steel material specifications was used for the:

1. Reactor coolant pressure boundary

2. Systems required for reactor shutdown

3. Systems required for emergency core cooling are listed in Tables 1.8-1 and 1.8-2.

The unstabilized austenitic stainless steel material specifications used for the reactor vessel internals which are required for emergency core cooling for any mode of normal operation or under postulated accident conditions, and for core structural load bearing members are listed in Table 1.8-3.

The water chemistry control is described in Section 4.2.8. These chemistry controls coupled with the satisfactory experience with components and internals using unstabilized austenitic steel materials which had been post weld heat treated, show compatibility of these heat treatments for stainless steel in a PWR chemistry environment.[(2)] Actual observation of post weld heat treated austenitic stainless steel after actual operation, indicate no effects of such treatments. Internals that were heat treated above 800°F and with subsequent service in the following plants have been examined and show acceptable material condition:

Robert Emmet Ginna

Jose Cabrera

Connecticut Yankee

San Onofre

Beznau, Unit 1

Yankee Rowe

Trino Vercellese (Vercelli)

For reactor vessel internals the austenitic stainless steel was given a stress relieving treatment above 800°F, using a high temperature stabilizing procedure. This was performed in the temperature range of 1,600°F to 1,900°F, with holding times sufficient to achieve chromium diffusion to the grain boundary regions to limit the effects of sensitization of Cr-carbide precipitation in the grain boundary. The stainless nozzles on the pressurizer were given a post weld treatment associated with fabrication of the head. No intergranular tests are planned because of satisfactory service experience as noted above.

1.8.2 Material Equivalency

Materials listed in the UFSAR that are qualified with an "or equivalent" statement may be replaced with an evaluated alternative material. Prior to the replacement of an existing material for a component part, an engineering technical evaluation is performed to determine suitability and acceptability for the application. These technical evaluations are performed utilizing approved procedures which meet the design control requirements of the BVPS Appendix B design control program. The site design control program under which acceptability of alternate materials is evaluated addresses many material properties and their interactions with the system environment. In a typical evaluation, material properties such as tensile strength, yield strength, ductility, fracture toughness, corrosion resistance, surface conditions, hardness, thermal conductivity, heat treatment, and electro-chemical potential are evaluated as appropriate for the application under evaluation. The term "equivalent," however, does not apply to materials specified in the UFSAR that are required by a legally binding commitment described in the license or NRC Safety Evaluation Report (SER).

1.8.3 Historical Information

UFSAR sections listed below have been determined to be historical based on Regulatory Guide 1.81 (References 1 and 2). This information is not intended or expected to be updated for the life of the plant.

While information provided in these sections has been designated as historical, associated design bases themselves should not. The original design bases continue to be part of the overall design bases for the facility, and new information may warrant their update.

Section	Title
1.4	Comparison with Other Stations
1.5	Research and Development Requirements
1.6	Identification of Contractors
2	Site
11A	Estimated Radioactive Nuclide Concentrations in Waste Disposal Systems and in Discharge to the Environment
11B	Evaluation Of The Doses From Radiation Exposure Due To Normal Operation Of Unit 1 of the Beaver Valley Power Station
13	Initial Tests And Operation
14B-5	Tritium Production Within a Light Water Reactor
14E	Generic Sensitivity Study Results For A 3-Loop Plant With 17 x 17 Fuel
A.1	Quality Assurance Program (Note: The Operations Phase Quality Assurance Program is described in the company Quality Assurance Program Manual.)
A.2	Duquesne Light Company Design and Construction Quality Assurance Program
A.3	Stone & Webster Engineering Corporation Quality Assurance Program (Design and Construction Phase)
A.4	Westinghouse Pressurized Water Reactor Systems Division Quality Assurance Plan (Design and Construction Phase)
A.5	Westinghouse Nuclear Fuel Division Reliability and Quality Assurance Program

<u>References for Section 1.8</u>

1. "ASTM Recommended Practice for Detecting Susceptibility to Intergranular Attack in Stainless Steels," ASTM A-262, The American Society for Testing Materials.

2. W. S. Hazelton, "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7735, Westinghouse Electric Corporation (August 1971).

3. "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," Nuclear Regulatory Commission Regulatory Guide 1.81 (September 1999).

4. "Guidelines for Updating Final Safety Analysis Reports," Nuclear Energy Institute guidance document NEI 98-03 (Revision 1, June 1999).

TABLES FOR SECTION 1

Table 1.1-1
UPDATED FSAR GUIDELINES

1. The Updated FSAR includes changes made to BVPS-1 up to six months prior to the filing date.

2. The Updated FSAR includes a physical description of changes made to BVPS-1 that have been approved for use and are operable. By convention, for future Updated FSAR Revisions, the Operational Acceptance Date associated with each design change, determines when each design change is considered approved for use and is operable, and must therefore be included, if applicable, in the Updated FSAR.

3. The Updated FSAR maintains the same level of detail as was provided in the Original FSAR.

4. Minor differences between actual and projected population figures or other such changes in the site environment are not reported unless the conclusions of safety analyses relative to public health and safety are affected and BVPS-1 prepared a new analyses as a result of NRC requirements.

5. The Updated FSAR includes where applicable, the latest information on specialized studies that has been developed in response to NRC requirements. New analyses which were required during consideration of facility or procedure changes, Technical Specification changes, or other licensing questions, may be incorporated as appendices or otherwise appropriately inserted within the Updated FSAR. No analyses are required other than those already prepared or submitted pursuant to NRC requirements, either originally with the application, or as part of the operating license review process, or those required by 10 CFR 50.59 or other NRC Requirement, or those prepared to support license amendments.

6. The Updated FSAR uses the same format as the Original FSAR.

7. Comparisons to other plants in the Updated FSAR were considered valid at the time the BVPS-1 Operating License was issued. This information is being maintained for historical perspectives.

8. The Updated FSAR references the Security and Emergency Plans that are currently in effect.

9. Responses to original AEC Questions are appropriately incorporated into the Updated FSAR.

10. The Updated FSAR has maintained information pertaining to programs described in the Original FSAR even though new information is not required to be submitted as part of the initial Updated FSAR. This information is being maintained for completeness and to locate previously submitted information in one document. This is noted in the text where applicable.

Table 1.2-1
DESCRIPTION OF MODIFIED FORTRAN TYPE FORMAT USED FOR WRITING EQUATIONS

SYMBOLS

Equals	=
Greater Than	>
Less Than	<
Addition	+
Subtraction	-
Multiplication	*
Division	/
Exponentiation (Represents the operation "to raise to a power")	**
Summation, i.e., $\sum_{N=1}^{10}$	SUM, N, 1, 10 ( )
Differentiation, i.e., dy/dx	dy/dx
Integration, i.e. $\int_{1}^{10} s\,dx$	INT, 1, 10 (S) dx
Parentheses	( )
Brackets	[ ]
Square Root	SQRT ( )
Sine of an Angle	SIN ( )
Cosine of an Angle	COS ( )
Tangent of an Angle	TAN ( )
Arctangent	ATAN ( )
Exponential - $e^{(\ )}$	EXP ( )
Natural Logarithm	LOG ( )

TABLE 1.2-1 (CONT'D)

DESCRIPTION OF MODIFIED FORTRAN TYPE FORMAT
USED FOR WRITING EQUATIONS

Absolute Value	ABS ( )
Greek Letters	Are Written Out, i.e., Alpha, Beta

RULES

1. All operators (+, -, *, /, **) must be explicitly stated i.e., A * C must be written, rather than AC.

2. No two operators can be juxtaposed, i.e., A/(-2) must be written, rather than A/-2.

3. The following is the hierarchy of operators:

   All exponentiations are performed first

   All multiplications and divisions are performed second

   All additions and subtractions are performed last

4. If an expression consists entirely of operators of the same priority, the expression is evaluated in order from left to right, except for exponentiations which are performed from right to left.

5. The presence of parentheses overrides the natural order of priority given under Items 3 and 4.

6. The following symbols are available in ATS as superscripts: numbers, +, -, † (cross), parentheses, >, and <. Superscripts using other symbols should not be used.

7. Subscripts should not be used, i.e., A_1 = A1, B_5 = B5

EXAMPLE

The equation

$$\frac{a^{.8} \log F}{Y} + .5\ X_1 = 10$$

is written as

(ALPHA ** .8) * LOG(F)/Y + .5 * X1 = 10

Table 1.3-1

PROTOTYPE REACTORS INTERNALS ASSURANCE PROGRAM STATUS**

PROTOTYPE REACTORS		TOPICAL REPORTS			
No. of Loops	Plant (Operating Utility)	Title	WCAP No.	Class	Status
2	Robert Emmett Ginna (Rochester Gas & Electric Corporation)	"Westinghouse PWR Internals Vibration Summary, 2-Loop	7845	2	AEC Accepted
		Internals Assurance"	7718	3	AEC Accepted
3	H. B. Robinson No. 2 (Carolina Power and Light Company)	"Westinghouse PWR Internals Vibration Summary, 3-Loop	7765-L	2	AEC Accepted with additional information
		Internals Assurance"	7765	3	AEC Accepted with additional information
			7765-L-AR* 7765-AR*	2 3	To be completed by December 1, 1972 & will constitute the approved AEC documents.
4	Indian Point No. 2 (Consolidated Edison Company of New York)	"Four-Loop PWR Internals Assurance and Test Program"	7879	2	Submitted to AEC Note: No notice of acceptance received) To be completed June, 1973.

*AR = Acceptance Review, notation used to designate report with additional information reviewed and accepted by the AEC and now this information is incorporated into the report and the report reissued.

** The information provided herein was considered valid and up-to-date at the time the Operating License was issued.

Table 1.4-1

COMPARISON OF DESIGN PARAMETERS* (Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
Thermal and Hydraulic Design Parameters					
1	Total Core Heat Output, Mwt	2,652	2,441	2,200	2,200
2	Total Core Heat Output, Btu per hr	9,051 x 10^6	8,331 x 10^6	7,479 x 10^6	7,479 x 10^6
3	Heat Generated in Fuel, Percent	97.4	97.4	97.4	97.4
4	Maximum Thermal Overpower, Percent	118	112	112	112
5	System Operating Pressure, Nominal psia	2,250	2,250	2,250	2,250
6	System Operating Pressure, Minimum Steady State, psia	2,220	2,220	2,220	2,220
7	Hot Channel Factors: Heat Flux, Fq	2.50	2.80	3.23	3.23
8	Hot Channel Factors: Enthalpy Rise, F_H	1.55	1.60	1.77	1.77
9	DNBR at Nominal Initial Rating Conditions	1.86	1.97	1.81	1.81
10	Minimum DNBR for Design Transients	1.30	1.30	1.30	1.30

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
Coolant Flow					
11	Total Thermal Flow Rate, lb per hr	100.9×10^6	100.7×10^6	101.5×10^6	101.5×10^6
12	Effective Flow Rate for Heat Transfer, lb per hr	96.3×10^6	97.0×10^6	97.0×10^6	97.0×10^6
13	Effective Flow Area for Heat Transfer, ft^2	41.5	41.8	41.8	41.8
14	Average Velocity along Fuel Rods, ft per sec	14.4	14.2	14.3	14.3
15	Average Mass Velocity, lb per hr - ft^2	2.32×10^6	2.31×10^6	2.32×10^6	2.32×10^6
Coolant Temperatures, °F (at 100 Percent Full Power)					
16	Nominal Inlet	542.5	543	546.2	546.2
17	Maximum Inlet Due to Instrumentation Error and Deadband	546.5	547	550.2	550.2
18	Average Rise in Vessel, F°	67.4	62.6	55.9	55.9
19	Average Rise in Core, F°	70.3	65.5	58.3	58.3

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
Coolant Temperatures, F (at 100 Percent Full Power) (Cont'd)					
20	Average in Core, F	579.3	577	575.4	575.4
21	Average in Vessel, F	576.2	574	574.2	574.2
Heat Transfer at 100 Percent Power					
25	Active Heat Transfer Surface Area, ft^2	48,600	42,460	42,460	42,460
26	Average Heat Flux, Btu per hr - ft^2	181,400	191,100	171,600	171,600
27	Maximum Heat Flux, Btu per hr - ft^2	453,500	534,100	554,200	554,200
28	Average Thermal Output, kW per ft	5.2	6.2	5.5	5.5
29	Maximum Thermal Output, kW per ft	13.0	17.3	17.9	17.9
Fuel Central Temperature, (BOL) F					
31	Maximum at 100 Percent Power	3,400	4,050	4,030	4,030
32	Maximum at Overpower	4,400	4,300	4,300	4,300

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
Fuel Central Temperature, (BOL) F (Cont'd)					
33	Thermal Output, kW per ft at Maximum Overpower	21.1	20.6	20.0	20.0
CORE MECHANICAL DESIGN PARAMETERS					
	Fuel Assemblies				
34	Design	Canless 17 x 17	Canless 15 x 15	Canless 15 x 15	Canless 15 x 15
35	Rod Pitch, inches	0.496	0.563	0.563	0.563
36	Overall Dimensions, inches	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426
37	Fuel Weight (as UO_2), lb	181,205	176,200	176,200	176,200
39	Number of Grids per Assembly	7	7	7	7
	Fuel Rods				
40	Number	41,448	32,028	32,028	32,028
41	Outside Diameter, inches	0.374	0.422	0.422	0.422
42	Diametral Gap, inches	0.0065,	0.0075, 0.0075, 0.0085	0.0065	0.0065

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
	Fuel Rods (Cont'd)				
43	Clad Thickness, inches	0.0225	0.0243	0.0243	0.0243
44	Clad Material	Zircaloy-4	Zircaloy-4	Zircaloy	Zircaloy
	Fuel Pellets				
45	Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered
46	Density (Percent of Theoretical)	95	94, 93, 92	94, 92, 91	94, 92, 91
47	Diameter, inches	0.3225	0.3659, 0.3659, 0.3649	0.3669	0.3669
48	Length, inches	0.530	0.6000	0.6000	0.6000
	Control Rod Assemblies				
49	Neutron Absorber	5 percent Cd- 15 percent In 80 percent Ag	5 percent Cd- 15 percent In 80 percent Ag	5 percent Cd- 15 percent In 80 percent Ag	5 percent Cd- 15 percent In 80 percent Ag
50	Cladding Material	Type 304 SS Cold Worked	Type 304 SS Cold Worked	Type 304 SS Cold Worked	Type 304 SS Cold Worked
51	Clad Thickness, inches	0.0188	0.019	0.019	0.019

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
	Control Rod Assemblies (Cont'd)				
52	Number of Control Rod Assemblies (Full Length/Part Length)	48/5	48/5	45/8	45/8
53	Number of Rods per Assembly	24	20	20	20
	Total Rod Worth (%$\Delta\rho$)	See Table 3.3-3	See Table 3.3-3	See Table 3.2.1-3	See Table 3.2.1-3
	Core Structure				
54	Core Barrel ID/OD, inches	133.875/137.875	133.875/137.875	133.875/137.875	133.875/137.875
55	Thermal Shield ID/OD, inches	142.625/148.000	142.625/148.000	142.625/148.0	142.625/148.0
NUCLEAR DESIGN DATA					
	Structural Characteristics				
56	Fuel Weight (as UO_2), lb	181,205	176,200	176,200	176,200
57	Clad Weight, lb	38,230	36,300	36,300	36,300
58	Core Diameter, inches (Equivalent)	119.7	119.5	119.5	119.5
59	Core Height, inches (Active Fuel)	143.7	144	144	144

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
	Reflector Thickness and Composition				
60	Top - Water plus Steel	10 in.	10 in.	10 in.	10 in.
61	Bottom - Water plus Steel	10 in.	10 in.	10 in.	10 in.
62	Side - Water plus Steel	15 in.	15 in.	15 in.	15 in.
63	H_2O/U, Cold Molecular Ratio (Lattice)	3.43	4.18	4.18	4.18
64	Number of Fuel Assemblies	157	157	157	157
65	UO_2 Rods per Assembly	264	204	204	204
	Performance Characteristics				
66	Loading Technique	3 region, nonuniform	3 region, nonuniform	3 region, nonuniform	3 region, nonuniform
	Fuel Burnup, MWD per MTU				
67	Average First Cycle	13,700	12,600	13,000	13,000
68	First Core Average	33,000	22,300	24,500	24,500

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
	Feed Enrichments, w/o				
69	Region 1	2.10	1.85	1.85	1.85
70	Region 2	2.60	2.55	2.55	2.55
71	Region 3	3.10	3.10	3.10	3.10
	Control Characteristics				
	Effective Multiplication (Beginning of Life), k_{eff}				
72	Cold, No Power, Clean	1.25	1.176	1.180	1.180
	Boron Concentrations				
75	To Shut Reactor Down With No Control Rod Assemblies Inserted, Clean (keff = 0.99) Cold	1,443	1,250 ppm	1,250 ppm	1,250 ppm
	Hot	1,418	1,240 ppm	1,210 ppm	1,210 ppm
76	To Control at Power With No Control Rod Assemblies Inserted, Clean/Equilibrium Xenon and Samarium	1197/904	700 ppm	1,000 ppm/ 670 ppm	1,000 ppm/ 920 ppm

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
	Boron Concentrations (Cont'd)				
77	Burnable Poison Worth, Hot	7.0	6.9 percent $\Delta k/k$	7.3 $\Delta k/k$	7.3 $\Delta k/k$
78	Burnable Poison Worth, Cold	5.5	5.3 percent $\Delta k/k$	5.6 $\Delta k/k$	5.6 $\Delta k/k$
	Kinetic Characteristics				
79	Moderator Temperature Coefficient	0.0 to -0.40 pcm/F	$+.3 \times 10^{-4}$ to -3.5×10^{-4} $\Delta k/k$ per F	$+0.3 \times 10^{-4}$ -3.5×10^{-4} $\Delta k/k$ per F	0.3×10^{-4} to -3.5×10^{-4} $\Delta k/k$ per F
82	Doppler Coefficient	See Figure 3.3-27 and 3.3-28	-1×10^{-5} to -1.6×10^{-5} Δ/k per F	-1×10^{-5} to -1.6×10^{-5} $\Delta k/k$ per F	-1×10^{-5} to -1.6×10^{-5} Δ/k per F
	REACTOR COOLANT SYSTEM-CODE REQUIREMENTS				
	Component				
83	Reactor Vessel	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
	Steam Generator				
84	Tube Side	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
85	Shell Side	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
Steam Generator (Cont'd)					
86	Pressurizer	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
87	Pressurizer Relief Tank	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C
88	Pressurizer Safety Valves	ASME III	ASME III	ASME III	ASME III
89	Reactor Coolant Piping	USAS B31.1	USAS B31.1	USAS B31.1	USAS B31.1
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM (100 PERCENT POWER)					
90	Reactor Core Heat Output, MWt	2,652	2,441	2,200	2,200
91	Reactor Heat Output, Btu per hr	$9{,}051 \times 10^6$	$8{,}331 \times 10^6$	$2{,}479 \times 10^6$	$7{,}479 \times 10^6$
92	Operating Pressure, psig	2,235	2,235	2,235	2,235
93	Reactor Inlet Temperature, F	543.5	543	546.2	546.2
94	Reactor Outlet Temperature, F	610.9	605.8	602.1	602.1
95	Number of Loops	3	3	3	3
96	Design Pressure, psig	2,485	2,485	2,485	2,485

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM (100 PERCENT POWER)(Cont'd)					
97	Design Temperature, F	650	650	650	650
98	Hydrostatic Test Pressure (Cold, psig)	3,107	3,107	3,107	3,107
99	Coolant Volume, Including Total Pressurizer, ft^3	9,458	9,458	9,088	9,088
100	Total Reactor Flow, gpm	265,500	265,500	268,500	268,500
REACTOR DESIGN PARAMETERS OF THE REACTOR VESSEL					
101	Material	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA-302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel
102	Design Pressure, psig	2,485	2,485	2,485	2,485
103	Design Temperature, F	650	650	650	650
104	Operating Pressure, psig	2,235	2,235	2,235	2,235

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
REACTOR DESIGN PARAMETERS OF THE REACTOR VESSEL (Cont'd)					
105	Inside Diameter of Shell, inches	157	157	155.5	155.5
106	Outside Diameter Across Nozzles, inches	247.7/8	252	236	236
107	Overall Height of Vessel and Enclosure Head, ft-inches	40-5	40-5	41-6	41-6
108	Minimum Clad Thickness, inches	5/32	5/32	5/32	5/32
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS					
109	Number of Units	3	3	3	3
110	Type	Vertical U-tube with integral-moisture separator	Vertical U-tube with integral-moisture separator	Vertical U-tube with integral-moisture separator	Vertical U-tube with integral-moisture separator
111	Tube Material	Inconel	Inconel	Inconel	Inconel
112	Shell Material	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel
113	Tube Side Design Pressure, psig	2,485	2,485	2,485	2,485

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

REACTOR DESIGN PARAMETERS OF THE STEAM GENERATORS (Cont'd)		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
114	Tube Side Design Temperature, F	650	650	650	650
115	Tube Side Design Flow, lb per hr	33.6×10^6	33.57×10^6	33.43×10^6	33.93×10^6
116	Shell Side Design Pressure, psig	1,085	1,085	1,085	1,085
117	Shell Side Design Temperature, F	600	600	556	556
118	Operating Pressure, Tube Side, Nominal, psig	2,235	2,235	2,235	2,235
119	Operating Pressure, Shell Side, Maximum, psig	1,005	1,005	1,005	1,005
120	Maximum Moisture at Outlet at Full Load, percent	0.25	0.25	0.25	0.25
121	Hydrostatic Test Pressure, Tube Side, (Cold), psig	3,107	3,107	3,107	3,110

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS					
122	Number of Units	3	3	3	3
123	Type	Vertical, single stage mixed flow with bottom suction and horizontal discharge	Vertical, single stage mixed flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge
124	Design Pressure, psig	2,485	2,485	2,485	2,485
125	Design Temperature, F	650	650	650	650
126	Operating Pressure, Nominal, psig	2,235	2,235	2,235	2,235
127	Suction Temperature, F	549	543	546.5	546.5
128	Design Capacity, gpm	88,500	88,500	89,500	89,500
129	Design Head, ft	280	280	260	260
130	Hydrostatic Test Pressure (Cold), psig	3,107	3,107	3,107	3,107

TABLE 1.4-1 (CONT'D)

COMPARISON OF DESIGN PARAMETERS*
(Initial Ratings)

		Beaver Valley Final Report	Surry Final Report	Turkey Point No. 3 or No. 4 Final Report	H. B. Robinson Final Report
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS (Cont'd)					
131	Motor Type	A-C Induction single speed	A-C Induction single speed	A-C Induction single speed	A-C Induction single speed
132	Motor Rating	6,000 Hp	6,000 Hp	6,000 Hp	6,000 Hp
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING					
133	Material	Austenitic SS	Austenitic SS	Austenitic SS	Austenitic SS
134	Hot Leg - ID, inches	29	29	29	29
135	Cold Leg - ID, inches	27.5	27.5	27.5	27.5
136	Between Pump and Steam Generator - ID, inches	31	31	31	31
137	Design Pressure, psig	2,485	2,485	2,485	2,485

* The comparison of design parameters provided herein were considered valid at the time the BVPS-1 Operating License was issued.

Table 1.5-1

EFFECT OF ADDING EIGHTH GRID ON 17 X 17 SEVEN GRID DESIGN TESTS

Test	Parameter	Effect *
Fuel Assembly Structural Test	Axial Stiffness	Negligible effect at blowdown impact forces(9)
	Lateral Impact	Additional grid shares impact load(9)
Prototype Assembly Test	Pressure Drop	The margin between seven grid design ΔP and D loop results(10) is adequate to cover the additional ΔP resulting from the additional grid (less than five percent increase in ΔP)
	Lift Force	The margin between seven grid design(10) lift force and D loop results is adequate to cover the additional lift force resulting from the additional grid
	Rod Vibration	Decreased span length results in improved vibration characteristics and reduced rod wear
Departure from Nucleate Boiling	DNB Correlation	Addition of a grid increases mixing which increases DNB margin
Incore Flow Mixing	TDC	TDC increases as grid spacing decreases(4)

* References provided in the "Effect" column are located in the References for Section 1.5, pages 1.5-27 and 1.5-28.

Table 1.5-2

DNB TEST FACILITY COLUMBIA UNIVERSITY HEAT TRANSFER LABORATORY, LOOP CHARACTERISTICS

Flow Rate	400 gpm maximum 40 gpm minimum
Working Pressure	3500 psia maximum
Test Section Inlet Temperature:	650 F maximum
Test Section Outlet Temperature:	700 F maximum
Test Section Heated Length:	16 ft. maximum
Power Input to Test Section:	7.5 MWe maximum

Table 1.5-3

REFLOOD TEST FACILITY

INITIAL CONDITIONS

Flooding rate	0.6 to 6 inch/second, variable
Pressure	20 to 100 psia
Subcooling	5 to 150 F
Peak rod power	0.4 to 1.0 KW/ft
Initial Rod temperature	1100 F to 1700 F

Table 1.5-4

D^2NB PHASE I TEST PARAMETERS

<u>INITIAL STEADY STATE CONDITIONS</u>	
Pressure	2250 psia
Test section mass velocity	2.5×10^6 lb/hr-ft^2
Inlet coolant temperature	560 F
<u>TRANSIENT CONDITIONS</u>	
Simulated breaks	Various break sizes will be simulated to cover range of typical large and small breaks

Table 1.5-5

D^2NB PHASE II TEST PARAMETERS

INITIAL STEADY STATE CONDITIONS	
Pressure	1750 to 1900 psia
Test section mass velocity	2.0 to 3.0 x 10^6 lb/hr-ft^2
Core inlet temperature	530 to 560 F
TRANSIENT RAMP CONDITIONS	
Pressure decrease	0 to 350 psi/sec (subcooled depressurization)
Flow decrease	0 to 100 percent per second
Inlet enthalpy	Constant

Table 1.5-6

SRBT INTERNAL PRESSURE AND HEAT RATES FOR A 17 x 17 FUEL ASSEMBLY

Heat Rate	Internal Pressure
(725 F to 1940 F)	(psi)
5 F/sec	1200, 1800
25 F/sec	1200, 1800
100 F/sec	1200, 1800

Table 1.5-7

CHARACTERISTICS OF A & B LOOPS LOW FLOW/HIGH-PRESSURE

HYDRAULIC FACILITIES

A & B LOOPS DESIGN PARAMETERS

Maximum Flow Rate	150 gpm at 300 ft
Maximum Pump Head	335 ft at 60 gpm
Maximum Allowable Temperature	650 F
Normal Working Pressure	2000 psi
Normal Working Temperature	600 F

Table 1.5-8

CHARACTERISTICS OF D-LOOP MEDIUM FLOW/HIGH-PRESSURE

HYDRAULIC FACILITIES

D-LOOP DESIGN PARAMETERS

Maximum Flow Rate	4400 gpm
Maximum Allowable Pressure	2400 psi
Maximum Allowable Temperature	650 F
Normal Working Pressure	2000 psi
Normal Working Temperature	600 F
Pump Head at 3000 gpm	290 ft
Maximum Pump Head	340 ft (at 1500 gpm)
Main Loop Flow Measurement	10 inch Venturi
Auxiliary Flow Measurement	6 inch Venturis (2 inch Branch Lines)

Table 1.5-9

CHARACTERISTICS OF E-LOOP LOW FLOW/HIGH-PRESSURE HYDRAULIC FACILITY

E-LOOP DESIGN PARAMETERS

Maximum Flow Rate	2000 gpm at 130 ft 1000 gpm at 260 ft
Maximum Working Pressure	Pump Head

Table 1.5-10

CHARACTERISTICS OF G-LOOP EMERGENCY CORE COOLING SYSTEM FACILITY

G-LOOP DESIGN PARAMETERS

Component	Material	Rated Press. (psi)	Rated Temp. (F)	Typical Press. (psi)	Operating Temp. (F)
Test Vessel	Carbon Steel	2000	650	1000	545
Downcomer Side Tank	Carbon Steel	2000	650	1000	545
In-Line Mixer	Carbon Steel	2000	650	1000	545
Mixer Accumulator	Stainless Steel	2500	650	1800	100
Flash Chamber	Carbon Steel	3000	700	2800	660
Separators Nos. 1 & 2	Carbon Steel	2000	650	1000	545
Spray Accumulators Nos. 1 & 2	Carbon Steel	2000	650	1800	150
Spray Accumulator No. 3	Stainless Steel	2500	650	1800	150
Reflood Tank	Stainless Steel	Atmos.	212	Atmos.	150
Primary Piping	Carbon Steel	2000	650	1000	545

Table 1.5-11

CHARACTERISTICS OF H-LOOP HIGH-FLOW HYDRAULIC FACILITY

h-LOOP DESIGN PARAMETERS

Maximum Flow rate	14,000 gpm
Pressure Drop Across Vessel Model	120 psi
Minimum Vessel Outlet Pressure	10 psig
Flow Accuracy	1/2 percent
Water Temperature Range	70 F to 200 F Maximum
Loop-to-Loop Temp. Variation	2 F
Maximum Loop-to-loop Flow Rate Variation	3 percent

Table 1.5-12

CHARACTERISTICS OF J-LOOP DELAYED DEPARTURE FROM NUCLEATE BOILING AND HEAT TRANSFER FACILITY

J-LOOP DESIGN PARAMETERS

Test Fluid	Demineralized Water
Design Pressure	2500 psia
Design Temperature	650 F
Maximum Flow Rate (hot)	450 gpm
Power Input to Test Vessel	3,500,000 watts (maximum)
Primary Test Heat Exchanger Rating	11,400,000 Btu/hr

Table 1.5-13

CHARACTERISTICS OF K-LOOP BORON THERMAL REGENERATION TEST

K-LOOP DESIGN PARAMETERS

Total Tank Capability	30,000 gal
Chiller Capacity	48 ice-tons
Maximum Ion Exchange Resin Test Volume	75 ft^3
Maximum Test Process Rate Capability	10 gpm/ft^2 bed area
Maximum Flow Test Capability	200 gpm
Minimum Boron Storage Mode Fluid Temp.	50 F
Maximum Boron Release Mode Fluid Temp.	160 F

Table 1.5-14

CHARACTERISTICS OF FLETCH-SET EMERGENCY CORE COOLING SYSTEM FACILITY

FLETCH-SET DESIGN PARAMETERS

100 Rod Bundle Maximum Power	1000 kW
Maximum Bundle Flooding Rate	86 gpm Water
Temperature Range	100 F to 200 F
System Pressure	0 to 60 psia

Table 1.5-15

CHARACTERISTICS OF SINGLE ROD TEST LOOP HEATER ROD DEVELOPMENT FACILITY

SINGLE ROD TEST LOOP DESIGN PARAMETERS

Maximum Operating Pressure	2250 psia
Maximum Operating Temperature	650 F
Maximum Flow Rate	10 gpm
System Capacity	5 gal
Maximum Power Available	200 kW
Piping Size	1 inch and 3 inches

Table 1.7-1

NONESSENTIAL COMMON STRUCTURES

Decontamination Building

Meteorological Tower and Instrument House

Solid Waste Disposal Building

Service Building (Partial)

Portions of Auxiliary Building housing - shared systems

Discharge Structure

Security Gatehouse

Warehouse and Offsite Warehouse

Primary Clean Area, Laboratories, Offices, and Personnel Facilities in Service Building

Primary Auxiliary Building (Liquid and Gaseous Waste Processing)

Primary Water Supply Pumphouse

Coolant Recover Area

Emergency Response Facility

Emergency Response Facility Diesel Generator Building

Emergency Response Facility Substation

Waste Handling Building

Old Steam Generator Storage Facility (OSGSF)

Table 1.7-2

NONESSENTIAL COMMON SYSTEMS, SUBSYSTEMS AND COMPONENTS

Water Treatment System

Sewage Conveyance System

Waste Disposal System (Liquid Waste, and Steam Generator Blowdown Processing, Gaseous Waste Decay Tanks, Gaseous Waste Process Vent on Unit 1 Cooling Tower, and Spent Resin Disposal Equipment Only)

Boron Recovery System (evaporators, associated equipment and same tankage only)

Fire Protection System

Decontamination System

Air Conditioning Systems for common structures

Nitrogen System

Table 1.7-3

NONESSENTIAL COMMON ELECTRICAL SYSTEMS

48V Battery and Battery Chargers

PAX Telephone System

Page Party Telephone System

Transmission Lines for Outside Source Station Service

Transformer Power Supply

Switchyard for Generator (separate generator circuit breaker)

Duct Line for Generator 1 and 2 Main Circuit Breaker Control

Intake Structure Motor Control Centers and Unit Substations

Table 1.8-1

REACTOR COOLANT SYSTEM MAJOR COMPONENT/PART MATERIALS

REACTOR COOLANT SYSTEM
MAJOR COMPONENT/PART MATERIALS

Reactor Vessel Components	ASME Code, ANSI or AWS Material(2)
Shell & Bottom Head Plates (other than core region)	SA533 Gr A, B or C, Class 1 or 2 (Vacuum treated)
Shell Plates (Core region)	SA533 Gr A or B, Class 1 (Vacuum treated)
Replacement Closure Head Vessel Flange & Nozzle Forgings	SA508 Class 2 or 3
Nozzle Safe Ends	SA182 Type F304 or F316
CRDM & Appurtenances - Upper Head	SB166 or 167 and SA182 Type F316
Instrumentation Tube Appurtenances - Lower Head	SB166 or 167 and SA182 Type F304, F304L, or F316
Closure Studs	SA540 Class 3 Gr B23 or B24
Closure Nuts	SA540 Class 3 Gr B23 or B24
Closure Washers	SA540 Class 3 Gr B23 or B24
Core Support Pads	SB166 with Carbon less than 0.10%
Monitor Tubes & Vent Pipe	SA312 or 376 Type 304 or 316 or SB167
Vessel Supports & Seal Ledge	SA516 Gr 70 Quenched & Tempered or SA533 Gr A, B or C, Class 1 or 2. (Vessel supports may be of weld metal buildup or equivalent strength)
Cladding	Stainless Steel Weld Metal Analysis A-7 (Reactor Vessel) 308L, 309L (Replacement Closure Head)
Stainless Weld Rod	Type 308, 309 or Type 312
Head Lifting Lugs(1)	SA533, Grade B, Class 2
Welds - Submerged Arc	Mil-E-18193, Type B-4(3)
Welds - Manual Metal Arc	SA316 Type 8018-C-3(3) SFA-5.14 CL. ERNiCrFe-7 (Head Vent & RVLIS tube stub to pipe, CRDM tube to CRDM flange) SFA-5.5 CL. E9018M (Lift Lug to Replacement Closure Head) SFA-5.11 CL. ENiCrFe-7 (J-groove)

Table 1.8-1 (CONT'D)

REACTOR COOLANT SYSTEM
MAJOR COMPONENT/PART MATERIALS

Reactor Vessel Components (Cont'd)	ASME Code, ANSI or AWS Material(2)
O-Ring Head Seals	Inconel 718 (i.e., SB 637 Grade 718)
Insulation	Stainless Steel
Steam Generator Components	
Pressure Forgings	SA508 Class 3
Nozzle Safe Ends	SA-336 CL F316LN
Channel Heads	SA-508 Class 3 Forging
Tubes	SB-163 Alloy 690
Cladding	Alloy 690 & Stainless Steel
Closure Studs	SA-193 GR B7 (Plasma Bond Treated)
Tube Plugs	Alloy 690 (when needed)
Cladding for Heads	Type 309L or 308L
Weld Rod	Type 316L, Type 308L, or Type 309
Pressurizer Components	
Pressure Plates	SA533 Gr A, B or C, Class 1 or 2
Pressure Forgings	SA508 Class 2 or 3
Nozzle Safe Ends	SA182 or 376 Type 316 or 316L and Ni-Cr-Fe Weld Metal F-Number 43

Table 1.8-1 (CONT'D)

REACTOR COOLANT SYSTEM
MAJOR COMPONENT/PART MATERIALS

Pressurizer Components (Cont'd)	ASME Code, ANSI or AWS Material[2]
Cladding	Stainless Steel Weld Metal Analysis A-7
Closure Bolting	SA540
Pressurizer Safety Valve Forgings	SA182 Type F316
Support Skirt	SA516 Grade 70
Instrument Tube Coupling	SA182 F316
Internal Plate	SA240 Type 304
Instrument Tubing	SA213 Type 316
Heater Well Tubing	SA213 Type 316 Seamless
Heater Well Adapter	SA182 F316
Pressurizer Relief Tank[1]	
Shell	ASTM A-285 Grade C
Heads	ASTM A-285 Grade C
Internal Coating	Amercoat 55
Reactor Coolant Stop Valves	
Body and Bonnet	SA351 Gr CF8, CF8A or CF8M
Stem	SA564 Gr 630 Cond 1100°F Heat Treatment
Reactor Coolant Pump	
Pressure Forgings	SA182 Type 304, 316 or 348
Pressure Castings (i.e., impeller, casing)	SA351 Gr CF8, CF8A or CF8M
Tube & Pipe	SA213, SA376 or SA312 - Seamless Type 304 or 316

Table 1.8-1 (CONT'D)

REACTOR COOLANT SYSTEM
MAJOR COMPONENT/PART MATERIALS

Reactor Coolant Pump(Cont'd)	ASME Code, ANSI or AWS Material(2)
Pressure Plates	SA240 Type 304 or 316
Bar Material	SA479 Type 304 or 316
Closure Bolting	SA193 Gr B7 or B8 SA540 Gr B23 or B24, SA453 Gr 660
Flywheel	SA533 Gr B, Class 1
Shaft(1)	ASTM A-182 Grade F347
Reactor Coolant Piping	
Reactor Coolant Pipe	Code Case 1423-1 Gr F302N or 316N, or SA351 Gr CF8A or CF8M Centrifugal Castings
Reactor Coolant Fittings	SA351 Gr CF8A or CF8M
Branch Nozzles	SA182 Gr F304 or 316 or Code Case 1423-1 Gr F304N or 316N
Surge Line & Loop Bypass	SA376 Type 304 and 316 or Code Case 1423-1 Gr F304N or 316N
Auxiliary Piping 1/2" through 12" and Wall Schedules 40S Through 80S (Ahead of second isolation valve)	ANSI B36.19(4)
All Other Auxiliary Piping (Ahead of second isolation valve)	ANSI B36.10(4)
Socket Weld Fittings	ANSI B16.11(4)
Piping Flanges	ANSI B16.5(4)
Auxiliary Piping Valves (Class I)	SA182 Type 304 or 316 or SA351 Gr CF8, CF8A or CF8M
Welding Materials	SFA 5.4 and 5.9 Type 308 or 308L (except Reactor Coolant Piping welds for Steam Generators)
	SFA 5.9 ER 316L (Reactor Coolant Piping welds for Steam Generators)

Table 1.8-1 (CONT'D)

REACTOR COOLANT SYSTEM
MAJOR COMPONENT/PART MATERIALS

Control Rod Drive Mechanism	ASME Code, ANSI or AWS Material(2)
Pressure Housing	SA182, Gr F304-LN; SA-213, GR TP304LN; SA-336, CIF304LN; or SA-479, Type 304LN or SA351 Gr CF8
Bar Material	SA479 Type 304
Welding Materials	SFA 5.4 and 5.9 Type 308 or 308L

NOTES:

1. Not required for emergency core cooling.
2. Materials listed in this table (except for ANSI material where Note 4 applies) may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."
3. No additionally imposed limits on residual elements.
4. These components/materials have not been evaluated to permit the use of equivalent material as defined in Section 1.8.2. A 10CFR50.59 evaluation shall be performed prior to changing these materials in the facility.

Table 1.8-2

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS I AND II AUXILIARY COMPONENTS

Valves	ASME Code Material(1)
Motor and Manual Operated Gate and Check Valves	
Bodies	SA182 Gr F316
Bonnets	SA182 Gr F316
Discs	SA182 Gr F316
Stems	SA564 Type 630 Cond 1100°F Heat Treatment
Air Operated Valves	
Bodies	SA182 Type F316 or SA351 Gr CF8 or CF8M
Bonnets	SA182 Type F316 or SA351 Gr CF8 or CF8M
Discs	SA182 Type F316 or SA564 Gr 630 Cond 1100°F Heat Treatment
Stems	SA182 Type F318 or SA564 Gr 630 Cond 1100°F Heat Treatment
Auxiliary Relief Valves	
Forgings	SA182 Type F316
Disc	SA479 Type 316
Miscellaneous Valves (2 inches and less)	
Bodies	SA479 Type 316 or SA351 Gr CF8
Bonnets	SA479 Type 316 or SA351 Gr CF8
Discs	SA479 Type 316
Stems	SA479 Type 410 or Type 304

Table 1.8-2 (CONT'D)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS I AND II AUXILIARY COMPONENTS

	ASME Materials[1]
Auxiliary Heat Exchangers	
Heads	SA182 Gr F304 or SA240 Type 304 or 316
Flanges	SA182 Gr F304 or F316
Flange Necks	SA182 Gr F304 or SA240 Type 316 or SA312 Type 304 Seamless
Tubes	SA213 TP304
Tube Sheets	SA240 Type 304 or 316 or SA182 Gr F304 or SA515 Gr 70 with Stainless Steel Weld Metal Analysis A-7 Cladding
Shells	SA351 Gr CF8
Pipe	SA312 Type 304 Seamless
Auxiliary Pressure Vessels, Tanks, Filters, etc.	
Shells & Heads	SA240 Type 304 or SA264 Type 304 Clad to SA516 Gr 70 or SA516 Gr 70 with Stainless Steel Weld Metal Analysis A-7 Cladding
Flanges & Nozzles	SA182 Gr F304 and SA105 or SA350 Gr LF2 with Stainless Steel Weld Metal Analysis A-7 Cladding
Piping	SA312 TP304 or TP316 Seamless
Pipe Fittings	SA403 WP304 Seamless
Closure Bolting & Nuts	SA193 Gr B7 or B8 and SA194 Gr 2H

Table 1.8-2 (CONT'D)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS I AND II AUXILIARY COMPONENTS

Auxiliary Pumps	ASME Code Materials[1]
Pump Casings & Heads	SA351 Gr CF8 or CF8M, SA182 Gr F304 or F316
Flanges & Nozzles	SA182 Gr F304 or F316, SA403 Gr WP316L Seamless
Piping	SA312 TP304 or TP316 Seamless
Stuffing or Packing Box Cover	SA351 Gr CF8 or CF8M, SA240 TP304 or TP316
Pipe Fittings	SA403 Gr WP316L Seamless
Closure Bolting & Nuts	SA193 Gr B6, B7 or B8M and SA194 Gr 2H or Gr 8M

NOTE:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 1.8-3

REACTOR VESSEL INTERNALS FOR EMERGENCY CORE COOLING

(ALL MATERIALS[1] ARE PER ASME CODE)

Forgings	SA182 Type F304
Plates	SA240 Type 304
Pipes	SA312 Type 304 Seamless or SA376 Type 304
Tubes	SA213 Type 304
Bars	SA479 Type 304 & 410
Castings	SA351 Gr CF8 or CF8A
Bolting	SA Pending Westinghouse PF Spec. 70041EA
Nuts	SA193 Gr B8
Locking Devices	SA479 Type 304
Weld Buttering	Stainless Steel Weld Metal Analysis A-7

NOTE:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 1.8-4

CLEANLINESS WATER CHEMISTRY REQUIREMENTS

Grade	A	B	C
Fluoride ion, max ppm	0.15	-	-
Chloride ion, max ppm	0.15	1.0	25
Conductivity, max micromhos/cm	2.0	20	40
pH range*	6-8	6-8	6-8
Visual clarity	No turbidity, oil, or sediment		

(-) Indicates not specified

* When water has been subjected to possible CO_2 absorption such as when retained in storage tanks, the pH requirement may be lowered to 5.8 to compensate for $C0_2$ absorption.

Table 1.8-5

ACCEPTABLE METHODS FOR WELDING AUSTENITIC STAINLESS STEEL

Welding Process Method	Energy Input Range, (Kilojoules/in.)
Manual Shielded Tungsten Arc	20 to 50
Manual Shielded Metallic Arc	15 to 120
Semi-Automatic Gas Shielded Metallic Arc	40 to 60
Automatic Gas Shielded Tungsten Arc-Hot Wire	10 to 50
Automatic Submerged Arc	60 to 140
Automatic Electron Beam-Soft Vacuum	10 to 50
Automatic Electroslag (Post Weld Heat Treated)	

Removed in Accordance with RIS 2015-17

FIGURE 1.2-1
SITE PLAN
BEAVER VALLEY POWER STATION UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

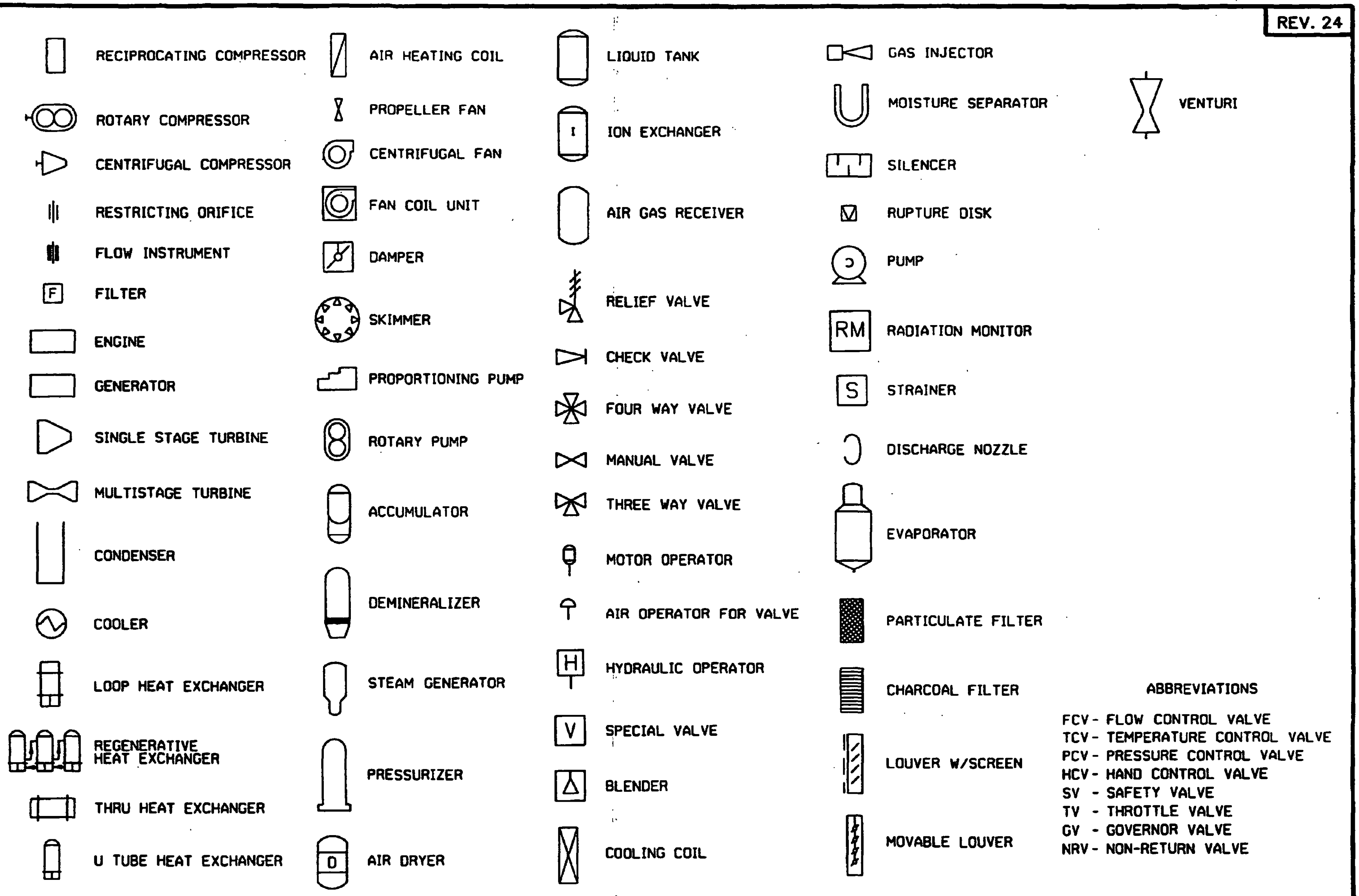

FIGURE 1.2-3
SYMBOLS FOR FLOW DIAGRAMS

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

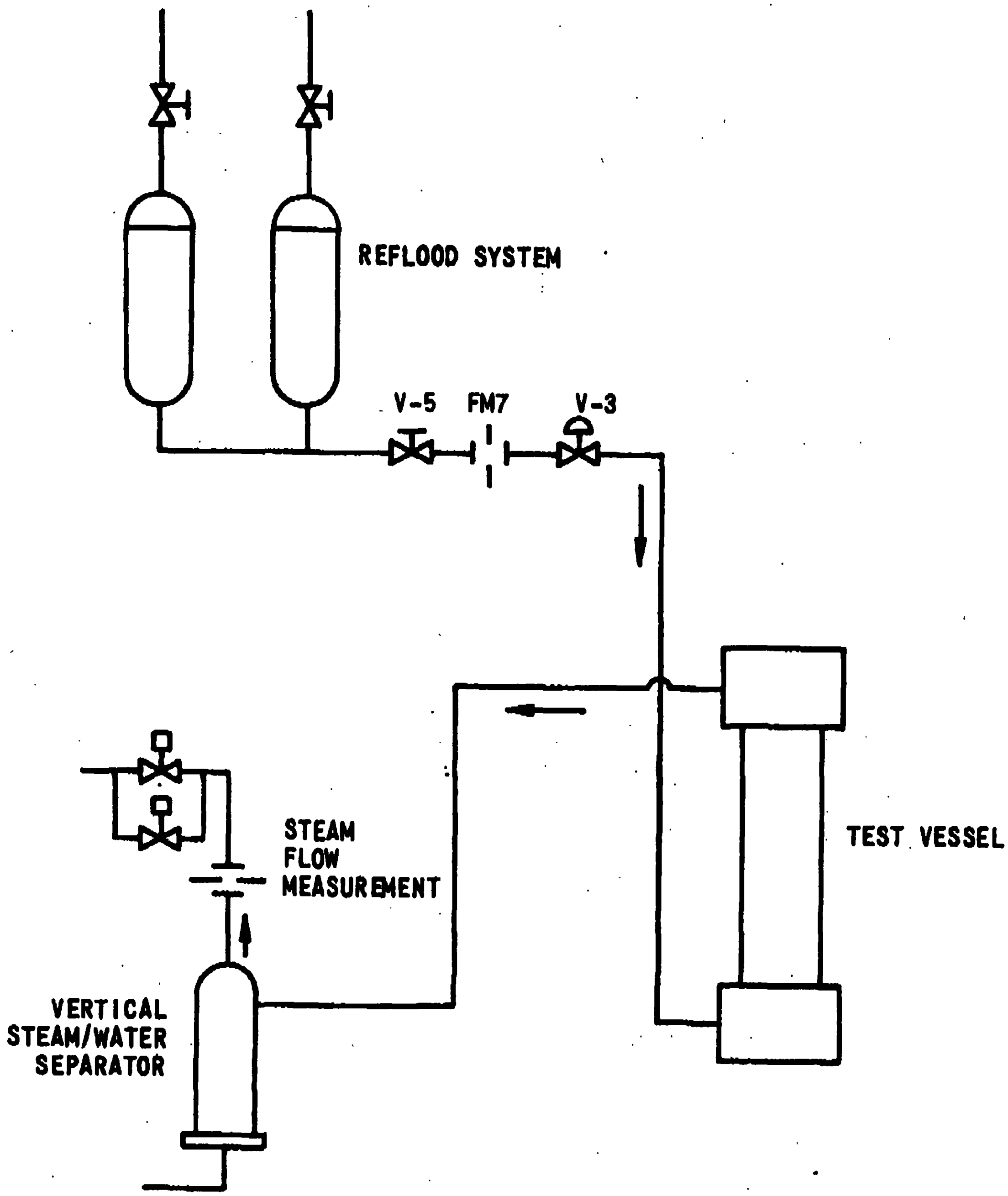

FIGURE 1.5-1
SCHEMATIC OF 17 X 17 REFLOOD TEST FACILITY
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

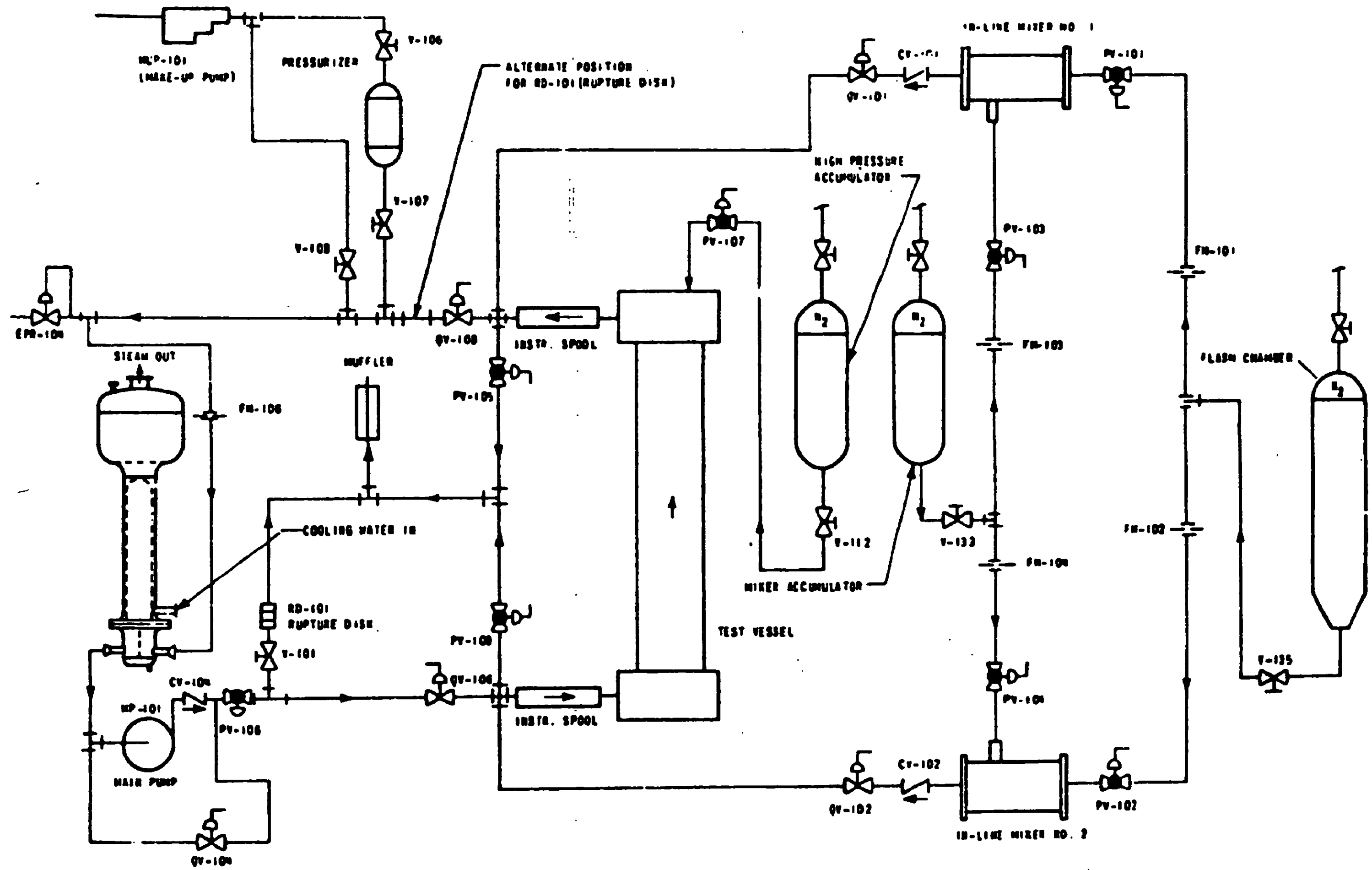

FIGURE 1.5-2
DNB TEST FACILITY SCHEMATIC
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

APPENDIX 1A

1971 AEC GENERAL DESIGN CRITERIA CONFORMANCE

TABLE OF CONTENTS

TABLE OF CONTENTS (CON'D)

* Criteria 1A.6 through 1A.9, 1A.47 through 1A.49, 1A.58, and 1A.59 did not exist in the Original FSAR. To maintain reference consistencies and FSAR text similarities, no change has been made to the Updated FSAR.

APPENDIX 1A

1971 AEC GENERAL DESIGN CRITERIA CONFORMANCE

This appendix provides a discussion of BVPS-I's degree of conformance to the AEC General Design Criteria (GDC) published as Appendix A to 10CFR50 in July, 1971. Section 1.3.2 has previously presented a discussion of Unit I's compliance with the 1967 AEC General Design Criteria. Below is a brief discussion of each applicable 1971 criterion:

1A.1 QUALITY STANDARDS AND RECORDS (CRITERION 1)

Criterion

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Design Conformance

Structures, systems and components important to safety are designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed.

Quality standards related to safety related systems, structures and components are generally contained in codes such as the ASME Boiler and Pressure Vessel Code. The applicability of these codes to structures, systems and components is identified throughout this report. The procedures for identifying and evaluating these codes for applicability and adequacy are given in Appendix A. Where codes are nonexistent or are judged inadequate, the procedures for supplementing, modifying, or establishing new quality standards are also described.

Appendix A describes the Project Quality Assurance Program established to provide adequate assurance that safety related structures, systems and components satisfactorily perform their safety functions. This Appendix describes the procedures for generating and maintaining appropriate design, fabrication, erection and testing records throughout the life of the unit.

1A.2 DESIGN BASIS FOR PROTECTION AGAINST NATURAL PHENOMENA (CRITERION 2)

Criterion

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Design Conformance

The structures, systems and components designated Seismic Category I are designed to withstand, without loss of capability to protect the public, the most severe environmental phenomena ever experienced at the site with appropriate margins included in the design for uncertainties in historical data. The environmental conditions assumed to occur within the containment as a function of time after a postulated LOCA are given in Section 7.1. Potential exterior environmental hazards are discussed and analyzed in Sections 2 and 14 of the FSAR, and the influence of these hazards on various aspects of the BVPS-1 design is discussed in the sections covering the specific systems and components concerned. An outline of the design philosophy for Seismic Category I structures, systems and components is included in Appendix B.

References

1. Section 2, Site
2. Section 3, Reactor
3. Section 4, Reactor Coolant System
4. Section 5, Containment System
5. Section 6, Engineered Safety Features
6. Section 7, Instrumentation and Control
7. Section 8, Electrical Systems
8. Section 9, Auxiliary and Emergency Systems
9. Section 11, Radioactive Wastes and Radiation Protection

1A.3 FIRE PROTECTION (CRITERION 3)

Criterion

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, system, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Design Conformance

The station will be designed on the basis of minimizing the use of combustible materials and of using fire resistant materials to the greatest extent possible. The fire protection system will be designed to furnish the capacity to detect and extinguish any probable fires which might occur at the station.

Reference

Section 9.10, Fire Protection System

1A.4 ENVIRONMENTAL AND MISSILE DESIGN BASES (CRITERION 4)

Criterion

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss- of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events outside the nuclear power unit.

Design Conformance

Interior missiles are adequately treated in the Section 5.2.6.1, exterior missiles are treated in Section 5.2.6.2 and the effects of blowdown jet forces and pipe whip are discussed in Section 5.2.6.3. Structures, systems and components defined as Seismic Category I are listed in Table B.1-1 and discussed in Appendix B.

Engineered safety features actuation system's components are designed and arranged so that radioactive, mechanical and thermal environments accompanying any emergency situation in which the components are required to function do not interfere with that function. The environmental design criteria assumed to occur within the containment as a function of time after a postulated LOCA are given in Section 7.1.

1A.5 SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS (CRITERION 5)

Criterion

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Design Conformance

The intake structure is designed to accommodate necessary water flow from the Ohio River to BVPS-1 and proposed BVPS-2. BVPS-1 is designed for the addition of a second unit without the sharing of any safety system.

1A.10 REACTOR DESIGN (CRITERION 10)

Criterion

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 10. Appropriate fuel margins are included in the plant design.

1A.11 REACTOR INHERENT PROTECTION (CRITERION 11)

Criterion

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 11. A negative reactivity coefficient is a basic feature of the design.

1A.12 SUPPRESSION OF REACTOR POWER OSCILIATIONS (CRITERION 12)

Criterion

The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 12. The design includes provisions to detect and control those power oscillations which might exceed acceptable fuel design limits during operation.

1A.13 INSTRUMENtATION AND CONTROL (CRITERION 13)

Criterion

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 13. Appropriate instrumentation and control systems have been provided to monitor and control pertinent variables and systems over normal and postulated accident conditions.

1A.14 REACTOR COOLANT PRESSURE BOUNDARY (CRITERION 14)

Criterion

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 14. The design, fabrication, erection and testing employed on the reactor coolant pressure boundaries and the extensive quality control measures employed during each of the above phases ensures that these pressure boundaries have extremely low probabilities of abnormal leakage, rapidly propagating failure and gross rupture.

1A.15 REACTOR COOLANT SYSTEM DESIGN (CRITERION 15)

Criterion

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 15. The design of the reactor coolant system and associated pertinent systems includes sufficient margin to ensure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded during normal operation including transients as defined in Section 14.

1A.16 CONTAINMENT DESIGN (CRITERION 16)

Criterion

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Design Conformance

A steel lined reinforced concrete containment structure will enclose the entire reactor coolant system with an essentially leaktight barrier as described in Section 5. Following a DBA, the containment depressurization system, as described in Section 6.4, will reduce the containment pressure, thereby reducing the driving force for the release of radioactivity.

1A.17 ELECTRIC POWER SYSTEMS (CRITERION 17)

Criterion

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Design Conformance

The BVPS-1 electric power system design includes two offsite power systems and two onsite power systems. Each system provides sufficient capability for operating all engineered safety features equipment which must be operated in the event of the postulated accidents.

The BVPS-1 onsite power system, which consists of the a-c onsite power system, 125 v d-c power system, and the 120 v a-c vital bus system has sufficient independence redundancy and testability to perform their safety functions assuming a single failure.

The two physically separated BVPS-1 station offsite power systems are fed by two independent circuits from separate buses in a switchyard common to both.

Both BVPS-1 offsite power systems are designed to be available immediately upon loss of all onsite a-c power sources since an automatic transfer scheme is provided for this purpose at the 4 Kv bus level, as described in Section 8. The electric power system is designed in accordance with the General Design Criterion 17, as discussed in Section 8.

1A.18 INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS (CRITERION 18)

Criterion

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the

systems into operation, including operation of power among the nuclear power unit, the offsite power system, and the onsite power system.

Design Conformance

The availability and proper action of safety related electrical power systems can be tested periodically.

Testing of automatic operation of the voltage transfer scheme at the 4,160 v level is performed at refueling frequency. Transfer of power from the unit transformers to system transformers is verified operable in both automatic and manual modes, at the frequency required by the Technical Specifications to the BVPS-1 Facility Operating License.

The BVPS-1 batteries which supply control power for operating emergency d-c motor starters and nuclear safety protection systems are kept at a constant voltage and are monitored continuously for voltage variations or undesired ground connections. The BVPS-1 batteries are subjected to periodic inspection.

The loading and automatic starting features of the emergency diesel generators are tested periodically. During these tests the diesel generator is loaded by manually initiating a safety injection signal, causing the required loads to be sequenced onto the emergency bus.

A preventative maintenance program is followed by BVPS-1 to test periodically the insulation values of equipment and circuits.

The inspection and testing of electric power systems is in accordance with General Design Criterion 18 and IEEE Std. 308[1] as more fully discussed in Sections 8.6 and in the Technical Specifications.

1A.19 CONTROL ROOM (CRITERION 19)

Criterion

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Design Conformance

A main control room is a fully shielded room with special ventilation features to meet radiation requirements. It is equipped to operate BVPS-1 safely under normal and accident conditions and to maintain it in a safe condition after a Design Basis Accident.

Redundant equipment, controls and instrumentation mounted on a separate emergency shutdown panel are provided outside the main control room to accomplish a prompt hot shutdown in a safe manner should the main control room become uninhabitable. Also, equipment, controls and instrumentation are located throughout BVPS-1 to provide capability for a subsequent cold shutdown through the use of suitable procedures.

The design of the main control room and the emergency shutdown panel area conforms to the above criteria and is described more fully in Section 7.7.

1A.20 PROTECTION SYSTEM FUNCTIONS (CRITERION 20)

Criterion

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Design Conformance

The BVPS-1 protection system design complies with the intent of criterion 20. The system will automatically initiate the reactivity control systems as described in Section 7. The system will also sense the accident conditions and initiate engineered safety features operation.

1A.21 PROTECTION SYSTEM RELIABILITY AND TESTABILITY (CRITERION 21)

Criterion

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Design Conformance

The BVPS-1 protection system design complies with the intent of criterion 21. The protection system is comprised of redundant independent trains of high functional reliability capable of tolerating a single failure without loss of the protection function, or removal from service of a single component or channel without loss of required redundancy. Independent channel tests can be performed with the reactor at power and the majority of system components can be tested very rapidly by use of built in semi-automatic testers.

1A.22 PROTECTION SYSTEM INDEPENDENCE (CRITERION 22)

Criterion

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Design Conformance

The BVPS-1 protection system design complies with the intent of criterion 22. Independent, redundant and separate subsystems have been provided. Extensive measurement, equipment and location diversity is employed in the design.

1A.23 PROTECTION SYSTEM FAILURE MODES (CRITERION 23)

Criterion

The protection system shall be designed to fall into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Design Conformance

The BVPS-1 protection system design complies with the intent of criterion 23. The system is designed with due consideration of the most probable failure modes of the components under various perturbations of energy sources and environment. Each trip channel is designed to trip on deenergization. Components of the system are qualified by testing for the environments which might result from postulated accident conditions.

1A.24 SEPARATION OF PROTECTION AND CONTROL SYSTEMS (CRITERION 24)

Criterion

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Design Conformance

The BVPS-1 protection and control system design complies with the intent of criterion 24. Failure of or removal from service of any single component or channel of either the protection system or the control system leaves intact a system satisfying the reliability, redundancy and independence requirements of the protection system. The protection system is separate and distinct; control system signals are derived from protection system measurements where applicable. Interconnection is through isolation amplifiers which are classified protection system components. The adequacy of systems isolations has been verified by testing under the conditions of maximum credible faults.

1A.25 PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS (CRITERION 25)

Criterion

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Design Conformance

The BVPS-1 design complies with the intent of criterion 25. The reactivity control systems are such that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

1A.26 REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY (CRITERION 26)

Criterion

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Design Conformances

The BVPS-1 design complies, with possible exception to the preferred rod insertion means, with the intent of Criterion 26. Two independent reactivity control systems of different design principles are provided. One of the systems uses control rods; the other system uses dissolved boron. The boron system is capable of maintaining the reactor core subcritical under cold conditions. The rod control system maintains a programmed average reactor temperature with scheduled and transient load changes; the boron system is capable of controlling the rate of

reactivity change resulting from planned normal power changes including xenon burnout. The control rods are inserted by gravity.

1A.27 COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY (CRITERION 27)

Criterion

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Design Conformance

The BVPS-1 design complies with the intent of criterion 27. Appropriate reactivity margin is available under postulated accident conditions to insure that the capability to cool the core is maintained. The margin includes an allowance for the most reactive rod control cluster being stuck out of the core.

1A.28 REACTIVITY LIMITS (CRITERION 28)

Criterion

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Design Conformance

The BVPS-1 design complies with the intent of criterion 28. The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing both control rods and boron removal are limited to values which prevent rupture of the coolant pressure boundary or disruption of the core or internals to a degree which could impair the effectiveness of ECCS. The appropriate reactivity insertion rate for withdrawal of rods and the dilution of boron in the coolant system are specified in the Technical Specifications.

1A.29 PROTECTION AGAINST ANTICIPATED OPERATION OCCURRENCES (CRITERION 29)

Criterion

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Design Conformance

The BVPS-1 design complies with the intent of criterion 29. The protection and reactivity control systems are designed to ensure an extremely high probability of fulfilling their intended functions. The design principles of diversity and redundancy coupled with a rigorous quality assurance program and analyses support this probability as does operating experience in plants using the same basic design.

1A.30 QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY (CRITERION 30)

Criterion

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 30. The quality levels employed for the reactor coolant pressure boundary are extremely comprehensive. Systems have been included in the plant to detect and to the extent practical, to locate leaks.

1A.31 FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY (CRITERION 31)

Criterion

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagation fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stressed, and (4) size of flaws.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 31. The reactor coolant pressure boundary is designed so that, for all normal operating and postulated accident modes, the boundary behaves in a non-brittle manner and so that the probability of rapidly propagating failure is minimized. Service temperature and pressure, irradiation, cyclic loading, seismic, blowdown and thermal forces from postulated accidents, residual stresses and code allowable material discontinuities have all been considered in the design with appropriate margins for each.

1A.32 INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY (CRITERION 32)

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 32. The reactor coolant pressure boundary will be periodically inspected under the provisions of ASME Code Section XI. A reactor vessel metal surveillance program will be employed in accordance with ASTM E-185-82[(2)] as discussed in detail in Section 4.

1A.33 REACTOR COOLANT MAKEUP (CRITERION 33)

Criterion

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 33. The normal flow path for reactor coolant system charging can be used to ensure appropriate makeup supply for small breaks.

1A.34 RESIDUAL HEAT REMOVAL (CRITERION 34)

Criterion

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 34. The residual heat removal system, consisting of two redundant trains of pumps and heat exchangers, has appropriate heat removal capacity to ensure fuel protection. This system supplements the normal steam and power conversion system which is used for the first stage cooldown. The auxiliary feedwater system complements the steam and power conversion system in the function. The systems together accommodate the single failure criteria.

1A.35 EMERGENCY CORE COOLING (CRITERION 35)

Criterion

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Design Conformance

The BVPS-1 design conforms with the intent of criterion 35. Appropriate core cooling systems have been designed so as to provide for the removal of core thermal loads and for the limiting of metal water reactions to an insignificant level. Suitable redundancy is provided in core cooling systems. The charging/safety injection, accumulator and safety injection systems will accommodate a single active failure and still fulfill their intended safety function.

1A.36 INSPECTION OF EMERGENCY CORE COOLING SYSTEM (CRITERION 36)

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Design Conformance

The BVPS-1 design provides for inspection of the emergency core cooling branch line connections to the reactor coolant system in accordance with the provisions of ASME Boiler and Pressure Vessel Code Section XI. These are the areas of principle stress in the system due to temperature gradients. The remainder of the systems will be verified as to integrity and functioning by means of periodic testing as described in the Technical Specifications. On this basis, the design conforms to the intent of criterion 36.

1A.37 TESTING OF EMERGENCY CORE COOLING SYSTEM (CRITERION 37)

Criterion

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operation sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Design Conformance

The BVPS-1 design conforms to the intent of criterion 37. Periodic tests will demonstrate the integrity, operability and performance of each active component. The system as a whole and the entire operational sequence of actuation, power transfer and cooling water operation will be tested in several phases rather than in one phase during periodic testing.

1A.38 CONTAINMENT HEAT REMOVAL (CRITERION 38)

Criterion

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Design Conformance

Four containment recirculation subsystems, described in Section 6.4.2, each with 50 percent capacity, remove heat from the containment following a DBA. Two electrical buses, each connected to both offsite and onsite power sources, feed the pump motors. Suitable, remote reading, water level indication is provided in the safeguards area for leak detection of safeguards equipment. Containment isolation valves provide containment isolation at the penetrations in accordance with the general design criteria.

1A.39 INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM (CRITERION 39)

Criterion

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the pumps, heat exchangers, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Design Conformance

The containment heat removal system design permits appropriate periodic inspection of its components as described in Section 6.4.

1A.40 TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM (CRITERION 40)

Criterion

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Design Conformance

The containment heat removal system design permits periodic pressure and functional testing as described in Section 6.4.

1A.41 CONTAINMENT ATMOSPHERE CLEANUP (CRITERION 41)

Criterion

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration the hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operating (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Design Conformance

Systems are provided to control fission products and ensure adequate mixing of hydrogen released by DBA (Sections 6.4 and 6.5). These systems are sufficiently redundant to meet the single failure criterion and are operable/functional with either onsite or offsite power sources.

1A.42 INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS (CRITERION 42)

Criterion

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Design Conformance

The containment depressurization system is designed to permit appropriate periodic inspection of the important components, as described in Section 6.4.

1A.43 TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS (CRITERION 43)

Criterion

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Design Conformance

The containment depressurization system is designed to permit periodic pressure and functional testing of their components, as described in Sections 6.4.

1A.44 COOLING WATER (CRITERION 44)

Criterion

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink, shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Design Conformance

The river water system and the primary component cooling water system provide heat removal from various structures, systems and components.

The primary component cooling water system transfers heat from heat exchangers containing reactor coolant or other radioactive liquids and from nonradioactive heat exchangers to the river water system. Three pumps and three heat exchangers for the system are located in the auxiliary building. Any one pump and heat exchanger is capable of performing the required safety function under any operating conditions.

The primary component cooling water system supplies cooling water to the following safety related items: (1) the residual heat removal heat exchangers and (2) fuel pool heat exchangers. The piping supplying these items, including the primary component cooling water pumps and heat exchangers, is Seismic Category I design.

The river water system transfers heat from the primary component cooling water system and other unit systems to the circulating water system which, in turn, transfers heat to the cooling tower. Redundancy is provided throughout the river water system in those portions serving safety-related items.

The intake structure is subdivided so that each of the three 100 percent capacity (with respect to those safety functions) river water pumps is separated and missile protected in Seismic Category I cubicles. The intake for each of these pumps is independent with separate screen and suction arrangement. Two 24 inch physically separated and missile protected river water headers from the intake structures provide full redundancy. Each header is underground until it reaches the auxiliary building basement.

In the event of a design basis accident (DBA), river water is automatically diverted from the 24 inch primary component cooling water headers to the two 24 inch headers which supply the recirculation spray heat exchangers. River water is available at all times to the control room air conditioners, charging pump lube oil coolers and emergency diesel generators.

The recirculation spray heat exchangers and pumps serve to reduce the containment pressure in the event of a DBA. Normal valving of the recirculation spray heat exchangers during unit operation is to permit fully automatic operation of both 100 percent portions of the recirculation spray system during a DBA.

Two independent diesel-generators provide emergency onsite power in the event of loss of normal power. Each generator supplies one emergency electrical bus which, in turn, supplies power to those components essential to the safety related operations of the cooling systems.

The cooling system safety function is assured with only onsite emergency power available.

1A.45 INSPECTION OF COOLING WATER SYSTEM (CRITERION 45)

Criterion

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the system.

Design Conformance

Both the river water system and the component cooling system are designed to permit appropriate periodic inspection to ensure the integrity of the components and the systems as a whole.

References

1. Section 9.4, Component Cooling System
2. Section 9.9, River Water System
3. Section 10.3.9, Turbine Plant Cooling Water

1A.46 TESTING OF COOLING WATER SYSTEM (CRITERION 46)

Criterion

The Cooling Water System shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Design Conformance

Many components of the river water system and the component cooling system are regularly in service during normal operation and, therefore, provide assurance of the availability and performance of the equipment and system.

The design of the river water components and system allows, to the extent practicable, the periodic testing of the operability of the system as required for operation in the loss-of-coolant accident and/or loss-of-unit power.

References

1. Section 9.8, Primary Component Cooling System
2. Section 9.9, River Water System
3. Section 10.3.9, Turbine Plant Cooling Water

1A.50 CONTAINMENT DESIGN BASIS (CRITERION 50)

Criterion

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Design Conformance

The containment structure is designed to leak at a rate which is less than 0.1 percent of the containment volume per day under post DBA conditions. The containment is designed to withstand loads above those that are conservatively calculated to result from a DBA (Sections 14.2.5.1 and 14.3.4) by a margin as discussed below. The containment has a design margin of about two percent over the maximum calculated peak pressure. The two percent is based on very conservative limiting assumptions.

1A.51 FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY (CRITERION 51)

Criterion

The reactor containment boundary shall be designed with sufficient margin to ensure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Design Conformance

Ferritic materials for the reactor containment boundary are specified so that the nil ductility transition (NDT) temperature of the steel is at least 60°F below the lowest of the minimum operating, maintenance, containment building testing or postulated accident temperatures. An applicable technical reference for this subject is provided in Reference 3. Figure 23 of Reference 3 shows the Fracture Analysis Diagram (FAD), which plots stress (as percent of yield strength) vs. the temperature in excess of the NDT temperature. The liner is designed so that no stress exceeds the crack arrest temperature (CAT) curve, as shown in the FAD. This is a very conservative approach, which ensures that flaws of any size will not be propagated to a rapid (i.e., brittle) fracture.

Uncertainties of determining the NDT temperature are minimized by using the Drop Weight Test DWT, per ASTM E-239[4] (previously ASTM E-208) for material five-eighths inch or thicker. The DWT is widely recognized to determine the true material NDT temperature. Plates thinner than five-eighths inch are impact tested by full or subsized Charpy V or by Drop Weight Tear Test methods. Also, the weld procedure qualification will demonstrate that the NDT temperature of the weld metal and heat affected zones follow the same criteria as for the base metal.

1A.52 CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING (CRITERION 52)

Criterion

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Design Conformance

The containment structure and related equipment which will be subjected to the containment test conditions, as described in Section 5.6, will be designed so that the periodic integrated leakage rate testing can be conducted at calculated peak containment pressure as per Appendix J to 10CFR50, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

1A.53 PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION (CRITERION 53)

Criterion

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Design Conformance

The design of the reactor containment provides for access to all important areas for periodic inspection. The design includes the placement of leak test channels over most liner seam welds, which are inaccessible after construction, and penetration-liner welds. These channels are not considered as safety related, however, they may be pressurized to containment design pressure to permit inspecting and testing the leaktightness of the covered areas. The operation of the containment provides for a continuous surveillance program of the leaktightness of the containment. The leakage monitoring system is described in Section 5.4.2.

1A.54 PIPING SYSTEMS PENETRATING CONTAINMENT (CRITERION 54)

Criterion

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Design Conformance

Piping systems penetrating the reactor containment do so in accordance with the design bases set forth for the containment isolation system (Section 5.3.2). This ensures redundancy, reliability and performance capabilities reflecting the importance to safety of isolating the piping systems. Special test connections are provided where required to ensure the capability to determine if individual isolation valve leakage is within acceptable limits. The containment leakage monitoring system (Section 5.4.2.2) also provides the capability of detecting unacceptable containment leakage.

1A.55 REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT (CRITERION 55)

Criterion

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provides for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provided greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Design Conformance

The containment isolation arrangements for all lines that are part of the reactor coolant pressure boundary, and that penetrate primary reactor containment, conform with the design bases listed in Section 5.3.1. These design bases include requiring that one of the arrangements called for in subparagraphs (1) through (4) of criterion 55 be utilized, unless acceptable on some other defined basis (as described in Section 5.3.3). The design bases also include the requirements that isolation valves outside the containment shall be located as close to containment as practical, automatic isolation valves shall take the position providing greater safety upon loss of actuating power, piping associated with the containment isolation system shall be designed, fabricated and tested in accordance with the requirements of piping Class I (QI) or Class II (Q2), as applicable, (Section 6.2-2), and simple check valves are not acceptable as outside containment isolation valves. The specific General Design Criteria met by piping systems penetrating containment are listed on Table 5.3-1. Where "locked closed" valves are called for by the General Design Criteria, administratively controlled, normally closed, manually operated valves are provided.

1A.56 PRIMARY CONTAINMENT ISOLATION (CRITERION 56)

Criterion

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Design Conformance

The containment isolation arrangements for all lines that connect directly to the containment atmosphere and penetrate primary reactor containment conform with the design bases listed in Section 5.3.1. These design bases requiring that one of the arrangements called for in subparagraphs (1) through (4) of this criterion be utilized, unless acceptable on some other defined basis (as described in Section 5.3.3). The design bases also include requirements that isolation valves outside the containment shall be located as close to containment as practical; that automatic isolation valves shall take the position providing greater safety upon loss of actuating power, piping associated with the containment isolation system be designed, fabricated and tested in accordance with the requirements of piping Class I (QI) or Class II (Q2), as applicable, (Section 6.2-2) and that simple check valves are not acceptable as outside containment isolation valves. The specific General Design Criteria met by piping systems penetrating containment are listed on Table 5.3-1. Where "locked closed" valves are called for by the General Design Criteria administratively controlled, normally closed, manually operated valves are provided.

1A.57 CLOSED SYSTEMS ISOLATION VALVES (CRITERION 57)

Criterion

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Design Conformance

The containment isolation arrangements for all lines that penetrate reactor containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere conform with the design bases listed in Section 5.3.1. These design bases include requiring that the outside isolation valve for such closed systems be either automatic, or normally closed, administratively controlled and manually operated. For normally open lines capable of remote manual operation, the valve shall be located as close to the containment as practical and that simple check valves may not be used as the automatic isolation valve.

1A.60 CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT (CRITERION 60)

Criterion

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Design Conformance

In all cases, the design for radioactivity control is justified (1) on the basis of 10 CFR 20 and 10 CFR 50 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and (2) on the basis of 10 CFR 100 or 10 CFR 50.67, as applicable, dosage level guidelines for potential accidents of exceedingly low probability of occurrence.

Control of waste gas effluents is accomplished by charcoal delay beds and holdup of waste gases in decay tanks until the activity of tank contents and existing environmental conditions permit discharges within 10 CFR 20 and 10 CFR 50 requirements. In addition, waste gas effluents are monitored prior to discharge for radioactivity and rate of flow. An accidental burst of the gas surge tank does not result in an activity release greater than 10 CFR 100 limits, based on one percent failed fuel.

Control of liquid waste effluents is maintained by batch processing of all station radioactive liquids, sampling before discharge, controlling the rate of release, and by preventing inadvertent tank discharge. Liquid effluents are monitored for radioactivity and rate of flow. Liquid waste disposal system tankage and evaporator capacity is sufficient to handle any expected transient in the development of liquid waste volume.

Station solid wastes are prepared batchwise for offsite disposal by approved contractors. Solid wastes are prepared for shipment by placement in shielded and reinforced containers which meet Federal Regulation requirements.

References

1. Section 5, Containment System
2. Section 9, Auxiliary and Emergency Systems
3. Section 11, Radioactive Wastes and Radiation Protection
4. Section 14, Safety Analysis

1A.61 FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL (CRITERION 61)

Criterion

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Design Conformance

Safety related components of the radioactive waste and fuel storage systems are designed to allow periodic inspection and testing. Process radiation monitors and flow measuring equipment are provided for surveillance or various station waste process streams. The waste disposal and radiation monitoring systems are designed to satisfy the General Design Criteria given in Section 1.3. In addition, these systems are designed to protect the health of station operating personnel and to limit discharge of radioactive materials from the station so as not to exceed the limits of 10CFR20. The waste disposal systems are discussed in detail in Section 11.

The spent fuel storage pool is designed to meet the requirements of 10 CFR 20 in providing radiation shielding for operating personnel during fuel transfer and during storage of spent fuel. Work areas adjacent to the spent fuel pool transfer canal wall are shielded for personnel access during actual fuel transfers. The spent fuel pool is permanently flooded to provide a minimum of 100 inches of water above a fuel assembly being transferred within the spent fuel pool. Water height above stored fuel assemblies is a minimum of 23 ft. The sides of the spent fuel pool, three of which also form part of the fuel building exterior walls, are 6 ft thick concrete to ensure a dose rate of no more than 2.5 mrem per hour outside the building. Fuel handling shielding is discussed fully in Section 11.3.

The refueling cavity above the reactor vessel is flooded to provide a temporary water shield above the components being withdrawn from the reactor vessel. This height ensures a minimum of 100 inches of water above a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate is less than 50 mrem per hour at the water surface.

The spent fuel handling system is designed to preclude gross mechanical failures which could lead to significant radioactivity releases. Floor and trench drain systems provide backup by collecting leakage which might occur. Design of the fuel storage pool ensures that there is no significant loss of fuel storage coolant under accident conditions. Decay heat from spent fuel is dissipated in the water of the storage pool and subsequently removed by a cooling system. If an accident were to damage the fuel pool cooling system, the heat of the spent fuel would have to be removed by evaporation of the spent fuel water. Make-up water can be supplied from the engine driven fire pump. Redundancy of the fuel pool cooling system components is provided to ensure reliability in maintaining the storage pool water cleanliness, level and heat removal ability. The fuel pool cooling and purification systems are described in detail in Section 9.5.

Radioactive gases, which may leak from spent fuel or radioactive waste disposal tanks in the decontamination and auxiliary building, are collected by the fuel building system. All discharges from these systems are monitored. All monitoring systems are discussed in Section 11.3.

Periodic surveys by Radiation Control personnel using portable radiation detectors ensure that radiation levels outside the shield walls meet design specifications. Section 11.3 describes shielding requirements.

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. For a description of the worst possible accident hypothesized refer to Section 14.

1A.62 PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING (CRITERION 62)

Criterion

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Design Conformance

The spent fuel storage racks are divided into two physical regions, Region 1 and Region 2. A third administratively controlled region, Region 3, is part of Region 2. Each region is defined by fuel enrichment vs. burnup limitations. The racks, free standing on the floor of the spent fuel pool, are sized to hold 1622 spent fuel assemblies (additionally there are 2 failed fuel assembly canisters). The spent fuel assemblies are placed in vertical cells within the rack, continuously grouped in parallel in both directions. Cell pitch is approximately 10.8" for Region 1 and approximately 9" for Region 2. In addition, Boral panels are installed in the walls of the individual cells to maintain subcriticality. The racks are so arranged that the spacing between fuel elements cannot be less than that prescribed. Borated water (approximately 2000 ppm) is used in the spent fuel pool. Even if unborated water were introduced, the spacing and Boral maintain subcriticality with $K_{eff} \leq 0.95$ for stored fuel.

The new fuel assemblies are stored dry in a steel and concrete structure within the fuel building. The assemblies are stored vertically in racks in parallel rows, having a fuel assembly center-to-center distance of about 21 inches. There is storage space for one-third (53 assemblies) of a core plus 17 spare assembly spaces. The steel rack construction prevents possible criticality by requiring that the spacing between fuel elements will not be less than that prescribed. In the event of accidental flooding of the fresh fuel racks, the center to center spacing of the fuel assemblies results in $K_{eff} \leq 0.95$ under full water density conditions and $K_{eff} \leq 0.98$ under low water density (optimum moderation) and aqueous foam conditions. Criticality prevention is discussed in detail in Section 9.12 and Section 3.3.2.7.

During handling, as a result of the hypothetical worst case accident the safeguards are designed such that the consequences of this accident meet 10 CFR 50.67 guidelines. For a complete description of this worst case accident, refer to Section 14.2.

1A.63 MONITORING FUEL AND WASTE STORAGE (CRITERION 63)

Criterion

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Design Conformance

Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. The fuel pool water temperature is continuously monitored. The temperature is displayed in the main control room where an audible alarm will sound should the water temperature increase above a preset level. The radiation level above the fuel pool is continuously monitored by a radiation detector mounted on the bridge of the fuel handling crane. A dose rate in excess of a preset level initiates an audible and visible alarm locally and in the main control room. Continuous surveillance of radiation levels in the waste storage and handling areas is maintained by two ventilation duct-mounted radiation detectors. Radiation levels in excess of preset levels initiate audible and visible alarms locally and in the control room.

For a more detailed description of the above radiation monitoring systems, refer to Sections 9.12 and 11.3.

In the event of a high temperature or high radiation alarm, administrative procedures provide for the protection of personnel and for the initiation of maximum fuel pool cooling and/or purification flow. Radiological control procedures, including appropriate radiation control surveys, are initiated as necessary to decontaminate affected areas. Refer to Sections 14.2 and 11.3 for a more detailed description of emergency procedures.

1A.64 MONITORING RADIOACTIVE RELEASES (CRITERION 64)

Criterion

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accident conditions.

Design Conformance

The containment atmosphere is monitored during normal unit operations and accident conditions, using the containment air particulate and gas monitors, which are located in the auxiliary building. In the event of accident conditions, samples of the containment atmosphere are obtained via a bypass sample line arrangement to provide data on existing airborne radioactive concentrations within the containment. The safe guards areas will be monitored by the ventilation vent sample air particulate and gas monitors. Radioactivity levels contained in the normal facility radioactive effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the unit radiation monitoring systems and by the environmental radiological safety program for this facility as described in Section 11.3.

References to Appendix 1A

1. "IEEE Criteria for Class IE Electric Systems for Nuclear Power and Generation Stations," IEEE Std. 308, The Electrical and Electronic Engineers, Inc.

2. "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM E-185-82, The American Society for Testing Materials.

3. W. S. Pellini and F. J. Loss, "Integration of Metallurgical and Fracture Mechanics Concepts of Transistion Temperature Factors Relating to Fracture-Safe Design for Structural Steel," Naval Research Laboratory Report 6900 (April 1969).

4. "ASTM Test for Locating the Thinnest Spot in Zinc (Galvanized) Coating on Iron or Steel Articles by the Preece Test (Copper Sulfate Dip)," ASTM E-239, The American Society for Testing Materials.