CHAPTER 1

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

Beaver Valley Power Station Unit 1 (BVPS-1) and BVPS-2 are located on a site in Shippingport Borough on the Ohio River in Beaver County, Pennsylvania. BVPS-2 uses a pressurized water reactor nuclear steam supply system (NSSS) and turbine generator, both furnished by Westinghouse Electric Corporation. It is similar in design concept to several projects already licensed or under review by the USNRC. The balance of the unit, including the containment structure, is designed and constructed by the Applicants, with the assistance of their agent, Stone & Webster Engineering Corporation.

The NSSS was originally designed for a warranted power output of 2,660 MWt, which was the license application rating, with an equivalent station net electrical output of approximately 836.0 MWe, assuming an atmospheric wet bulb of 44.85°F coincident with a dry bulb temperature of 53.4°F. 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 engineered safety features system design was originally based on 2,780 MWt core power. The reinforced concrete containment structure design was originally based on a core power of 2713 MWt and operation at subatmospheric pressure (approximately 9.5 psia). The NSSS design was based upon an expected ultimate output of approximately 2,774 MWt. This NSSS output resulted from a core power of 2,766 MWt and 8 MWt from the reactor coolant pumps with an equivalent station net electrical output of approximately 870.0 MWe, assuming an atmospheric wet bulb of 44.68°F coincident with a dry bulb temperature of 53.2°F. However, all safety analyses were originally based on power levels that conservatively reflect plant operating conditions.

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 core power level was subsequently licensed to 2,900 MWt in 2006 per Amendment 156. 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). The containment conversion to an atmospheric containment (License Amendment 153) was based on the uprated core power level of 2900 MWt. 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 1,009 MWe.

The remainder of Chapter 1 summarizes the principal design features and safety criteria of BVPS-2. Comparisons with other pressurized water reactor nuclear power stations now proposed or authorized for operation which employ essentially the same technology and basic engineering features are provided. The facilities shared between BVPS-1 and BVPS-2 are also discussed in Chapter 1.

The Final Safety Analysis report was originally prepared in accordance with the guidelines of Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 3, November 1978) as discussed in Section 1.8.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 General

Beaver Valley Power Station - Unit 2 (BVPS-2) incorporates a three-loop, closed-cycle, pressurized water nuclear steam supply system (NSSS), a tandem compound turbine generator, engineered safeguards systems, radioactive waste systems, a fuel handling system, and other facilities and auxiliaries required for a nuclear power plant.

Beaver Valley Power Station - Unit 1 (BVPS-1) facilities shared by BVPS-2 include the following:

1. Intake structure (Section 9.2.1),
2. Alternate intake structure (Section 9.2.1),
3. Control building,
4. Portions of the service building,
5. Portions of the auxiliary building, including the solid waste extension boron recovery system (BRS) (Section 9.3.4.6), gaseous waste system (Section 11.3), liquid waste system (Section 11.2), and primary grade water system (Section 9.2.8),
6. Portions of the turbine building, including the demineralized water makeup system (Section 9.2.3),
7. Ultimate heat sink (Section 9.2.5),
8. Primary grade water storage tanks,
9. Meteorological tower,
10. Interconnecting tunnels,
11. Cooling tower elevated release point,
12. Potable and sanitary water system (Section 9.2.4),
13. Site drainage system,
14. Fire protection system (Section 9.5.1),
15. Portions of the communications systems (Section 9.5.2),
16. Emergency diesel generators during a station blackout event (Section 8.3.1.1.19)

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 site area and adjacent Ohio River provide a minimum exclusion radius of 1,500 feet. The nearest continuously occupied residence is located about 2,300 feet from the reactor. The low population zone area distance is 3.6 miles. The population center distance is 17 miles. The area is primarily industrial with some agricultural activity. A more complete description of the site is presented in Section 2.1.

1.2.3 Structures

The general arrangement of structures for BVPS-2 is as shown on Figure 1.2-1. The major building areas include the containment structure, auxiliary building, fuel and decontamination building, safeguards area, main steam and cable vault area, turbine building, service building, diesel generator building, waste handling building, condensate polishing building, cooling tower, refueling water storage tank (RWST) enclosure, primary demineralized water storage tank enclosure, emergency outfall structure, cooling tower pump house, gaseous waste storage area, and the control building.

Except for the steel framed turbine building and cooling tower pump house, the structures are constructed predominantly of reinforced concrete. The containment is a steel-lined, reinforced concrete cylinder with a hemispherical dome and flat base. The cooling tower is a natural draft hyperbolic type with a reinforced concrete shell.

Structures housing safety-related equipment are the containment structure, safeguards area, main steam and cable vault, auxiliary building, fuel and decontamination building, control building, diesel generator building, service building, RWST enclosure, and primary demineralized water storage tank enclosure.

The BVPS-2 structures are separate from BVPS-1 structures. Passage between BVPS-1 and BVPS-2 is via a personnel bridge between the auxiliary building of BVPS-2 and the coolant recovery area of BVPS-1, and via an underground passageway between the auxiliary building of BVPS-2 and entrance area of BVPS-1.

Further information on the function, design, and layout of the plant structures is presented in Section 3.8.

1.2.4 Nuclear Steam Supply System

The NSSS consists of a pressurized water reactor, reactor coolant system (RCS), and associated auxiliary systems. The RCS is arranged as three reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump (RCP) and a steam generator. An electrically heated pressurizer is connected to the hot leg of one reactor coolant loop.

High pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes, via the coolant loop piping to the steam generators. Here it gives up its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire RCS is composed of leaktight components to ensure that fluids are confined to the system.

The core is of the multi-region type, which may utilize varying enrichments between regions.

In the initial core loading, three fuel enrichments are used. Fuel assemblies with the highest enrichments are placed at the core periphery, or outer region, and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, a portion of the fuel is replaced and new fuel is loaded in accordance with the core reload design.

Rod cluster control assemblies are used for reactor control and consist of clusters of cylindrical absorber rods. The absorber rods move within guide tubes in certain fuel assemblies. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. Downward trip of the rod cluster is by gravity.

The RCPs are Westinghouse Electric Corporation (Westinghouse) vertical, single-stage, centrifugal pumps of the shaft-seal type. The system piping arrangement and the coastdown feature of the RCPs upon loss of electrical power are designed so that adequate coolant flow is maintained to cool the reactor core under all required circumstances considered in the safety analysis.

The steam generators are Westinghouse vertical U-tube units which contain Inconel tubes. Integral moisture separation equipment reduces the moisture content of the effluent steam to one-quarter of one percent, or less. The reactor coolant loop stop and bypass valves are motor-operated gate valves remotely controlled from the main control room.

The reactor coolant piping and all of the pressure-containing and heat transfer surfaces in contact with primary coolant are stainless steel or stainless steel clad except the steam generator tubes and fuel tubes, which are Inconel and Zircaloy, respectively. The steam generator tube sheet is Inconel clad on the primary side, and the steam generator channel head divider plate is Inconel. Reactor core internals, including control rod drive shafts, are primarily stainless steel.

An electrically heated pressurizer, connected to one reactor coolant loop, maintains RCS pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

Auxiliary systems and components are provided to charge makeup water to the RCS, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control, cool system components, remove decay heat when the reactor is shut down, and provide for emergency safety injection.

1.2.5 Instrumentation and Control Systems

The instrumentation and controls for the reactor protection system, engineered safety features actuation system, and other safety-related systems, meet the requirements of IEEE Standard 279-1971, "Criteria for Nuclear Power Plant Protection Systems." In addition, other applicable criteria are met as described in Sections 3.1 and 7.1.2.

The nonsafety-related instrumentation and controls provide reliable control and allow continuous monitoring of the plant status without degradation of safety-related instrumentation. Design details are described in Section 7.7.

The application of instrumentation and controls to individual systems is detailed in the FSAR section describing the system.

The reactor is controlled by a coordinated combination of chemical shim and control rod assemblies, which are required for load-follow transients and for start-up and shutdown. The chemical shim is a soluble neutron absorber, boron, in the form of boric acid. The boric acid is added during cold shutdown, partially removed at start-up, and adjusted in concentration during core lifetime to compensate for such effects as fuel depletion and accumulation of fission products which tend to slow the nuclear chain reaction.

The control system allows the plant to accept step load increases of 10 percent and ramp load increases of 5 percent/minute over a load range of 15 percent to, but not exceeding, 100 percent power under normal operating conditions subject to xenon limitations. Equal step and ramp load reductions are possible, over the range of 100 to 15 percent of full power.

Technical Specifications require an anticipatory reactor trip following turbine trip above approximately 49% of full reactor power. The turbine bypass system's capability to permit a 50 percent external load rejection without turbine or reactor trip is discussed in Section 10.4.4.

Control of the reactor and the turbine generator is accomplished from the main control room, which contains all instrumentation and control equipment required for start-up, operation, and shutdown, including normal and accident conditions. The turbine generator controls are designed for manual operation with the operator selecting the load set point and loading rate. The NSSS can follow the turbine generator from loads of 15 to 100 percent power. If during rapid turbine generator loading (5 percent/minute), the response of the control rods and chemical shim is not adequate to supply the needed reactivity, the reactor coolant temperature will drop, resulting in an increase of reactivity.

1.2.6 Radioactive Waste System

Radioactive wastes are collected, processed, and disposed of in a safe manner complying with appropriate regulations: in particular, the U.S. Nuclear Regulatory Commission Regulations, 10 CFR 20 and 10 CFR 50, Appendix I Annex. The three interrelated radioactive waste treatment systems for radioactive liquid, gaseous, and solid wastes are described in detail in Chapter 11.

The radioactive liquid waste system, in combination with the steam generator blowdown system, collects and purifies radioactive liquid waste generated during operation and refueling for either recycling within the plant or discharge. The process operations available to treat liquid wastes are filtration, evaporation, and demineralization. Connections are provided to process liquid wastes with BVPS-1 facilities, when necessary. The system is described in Section 11.2.

Radioactive gaseous wastes are treated before release to the environment. Degasification and purification of the reactor coolant letdown reduces the in-containment radiation exposures and the in- plant consequences of any reactor coolant leakage. The degasification and purification processes produce gaseous streams, which, together with hydrogenated vents, are passed through charcoal delay beds to provide holdup time for the decay of noble gases and the removal of iodine. Air ejector vents are selectively delayed in separate charcoal beds. Aerated streams, produced by other phases of plant operation, are passed through high efficiency particulate air (HEPA) filters for particulate removal and charcoal adsorbers for iodine removal, as needed. Periodic batch disposal of degasifier effluent gases (primarily hydrogen) are sent to the BVPS-1 gaseous waste disposal system for discharge. Degasifier gases may be recycled back to the volume control tank. Sweep gases are filtered by a HEPA filter and, along with the containment vacuum pump discharge, are sent to BVPS-1 for final release to the environment. The plant is also equipped with seven gaseous waste storage tanks to provide the available storage space for hydrogen gas, prior to batch discharging for one unit going to cold shutdown. This system is described in Section 11.3.

The radioactive solid waste system provides packaging and storage facilities for the eventual shipment offsite and ultimate disposal of solid radioactive waste material. Available process operations include dewatering and pH adjustment of beaded resins, powdered resins, evaporator bottoms, and solidification of the waste with an in-drum cement system. Dry radioactive waste produced in the operation and maintenance of BVPS-2 may be baled or packaged, for shipment to an authorized offsite disposal location. Provisions for shielding during the processing and shipping preparations are included. As a backup, connections are provided to process beaded resin and evaporator bottoms with the BVPS-1 solidification facilities. This system is described in Section 11.4.

1.2.7 Fuel Handling

The reactor is refueled by equipment which handles the spent fuel underwater from the time it leaves the reactor vessel until it is placed in a shipping cask for shipment from the site. Underwater transfer of spent fuel provides an economic and transparent radiation shield, as well as a reliable coolant for removal of decay heat.

The fuel handling system is divided into two areas: the reactor cavity area, which is flooded for refueling; and the fuel pool, which is external to the containment and is always accessible to plant personnel. The two areas are connected by a fuel transfer system which carries the fuel through a containment penetration.

The design of the fuel transfer tube shielding inside the containment utilizes a labyrinth of steel and concrete shields. Outside of the containment, the shielding design utilizes a notched interlocking wall arrangement. Section 12.3.2.5 provides a detailed description of the fuel transfer tube shielding and arrangement.

Spent fuel is removed from the reactor vessel by a refueling machine and placed in the fuel transfer system. In the fuel pool, the fuel is removed from the transfer system and placed into storage racks. After a suitable decay period, and when offsite storage facilities or processing facilities are available, the fuel is removed from storage and loaded underwater into shipping casks for offsite transport. Storage is provided for no more than 1,690 spent fuel assemblies. Spent fuel storage is discussed in Section 9.1.2.

1.2.8 Turbine and Turbine Auxiliaries

1.2.8.1 Turbine Generator

The turbine is an 1,800 rpm, tandem-compound, four-flow, single-stage reheat unit with provision for six stages of feedwater heating. The turbine-generator is provided with an electro-hydraulic control system and with redundant emergency trip systems for turbine overspeed. The output of the turbine-generator is described in Section 10.2.

The generator is a direct driven, three-phase, 60 Hz, 22 kV, 1,800 rpm hydrogen inner-cooled, synchronous generator rated at 1,070 MVA at 0.92 power factor, and 0.61 short-circuit ratio at maximum hydrogen pressure of 75 psig.

1.2.8.2 Main Steam System

The main steam system delivers steam from the steam generators via three main steam lines to a 38 inch header which feeds four 28 inch lines to the turbine generator. The 38 inch header also supplies steam to the moisture separator reheaters, the auxiliary steam system, the gland seal steam system, and the turbine bypass valves. Branches from each of three lines from the steam generators supply steam to the turbine driven auxiliary feed pump.

1.2.8.3 Main Condenser

The main condenser condenses the turbine exhaust steam and maintains the turbine back pressure at 2 in Hg abs when operating at turbine guarantee conditions with approximately 65°F circulating water inlet temperature. The condenser includes provisions for accepting steam bypassed around the turbine-generator. Deaeration of condensate is accomplished in the condenser.

1.2.8.4 Main Condenser Air Removal System

The main condenser air removal system uses steam jet air ejectors for normal operation and vacuum priming ejectors for start-up. The system evacuates noncondensible gases from the main turbine and condenser during plant start-up, and maintains the condenser essentially free of gases during operation. The system handles all inleakage of noncondensible gases through the turbine seals, condensate, feedwater, and steam systems, including the steam generators in the event of primary to secondary leakage.

1.2.8.5 Turbine Gland Sealing System

The turbine gland sealing system provides clean, relatively moisture- free steam to the seals of the turbine throttle valve stem glands and the turbine shaft glands. The sealing steam is normally provided from the 38-inch main steam header. Leakoff from the seals is directed to the gland steam condenser. The generator shaft seals are sealed with lubricating oil to prevent hydrogen leakage.

1.2.8.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided to pass steam directly to the main condenser under the control of a pressure/temperature control regulator. Steam is bypassed to the condenser whenever the turbine trips or during start-up and cooldown. The turbine bypass system is capable of discharging steam flow directly to the main condenser. The capability of the turbine bypass system to mitigate a turbine and reactor trip upon loss of electrical load is discussed in Section 10.4.4.

1.2.8.7 Circulating Water System

The circulating water system provides the condenser with a continuous supply of cooling water. The circulating water system is a pumped, closed-loop system utilizing an air-cooled, natural draft hyperbolic cooling tower as a heat sink. Four one-quarter capacity circulating water pumps are provided to pump cooling water from the discharge of the condenser to the tower. The cooling water is then gravity-fed back to the condenser. Makeup water is provided from the Ohio River by the service water system. Water quality is controlled by blowdown to the Ohio River.

1.2.8.8 Condensate and Feedwater Systems

The condensate and feedwater systems supply condensate from the condenser hotwell to the steam generators. The condensate is normally pumped by two of three 50-percent design capacity condensate pumps through the full flow condensate polishing system to the intercooler and aftercooler of the air ejector, and the gland seal condenser. The condensate then flows through drain coolers and five stages of low pressure heaters. The drain coolers and low pressure heaters are split into two one-half capacity parallel streams. The last low pressure heaters discharge to the suction of two parallel, motor-driven, steam generator feedwater pumps. The discharge of the steam generator feedwater pumps passes through two one-half capacity parallel heaters and into the steam generators. The feedwater flow to each steam generator is controlled by a feedwater flow control valve located downstream of the heaters.

1.2.8.9 Condensate Polishing System

The Condensate Polishing System has been retired in place and is no longer connected to the Condensate System. Cleanup of the condensate is accomplished by the Steam Generator Blowdown System.

1.2.9 Electrical Systems

The main generator is an 1,800 rpm, 22 kV, three-phase, 60 Hz, hydrogen inner-cooled unit rated at 1,070 MVA at 0.92 power factor. One 1020 MVA main step-up transformer is provided to deliver power to the 345 kV switchyard.

The station service system consists of auxiliary transformers, 4,160 V switchgear, 480 V unit substations, 480 V motor control centers, 120 V ac vital and essential buses, and 125 V dc batteries and equipment. The normal source of station service power is obtained from the main generator through the unit station service transformers. The preferred source is available from the 138 kV high voltage switchyard through the system station service transformers. The BVPS-2 one-line electrical diagram is shown on Figure 8.3-1.

Two onsite, emergency diesel generators are provided to supply power in the event of complete loss of normal and preferred alternate station service power. Each emergency diesel generator supplies power to separate and redundant trains of the plant emergency station service system. Each emergency diesel generator has sufficient capacity for operation of all safety-related equipment which must be operated to mitigate the effects of a design basis accident (DBA) or to shut down the unit in a safe manner.

In addition, a third onsite, nonsafety diesel generator is provided to supply power to significant but nonsafety electrical loads whose loss, in the event of a complete loss of normal system ac power, would result in substantial equipment damage.

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 DBA conditions, the systems can maintain the integrity of the containment and keep potential exposures to the public within the radiation dose criteria given in 10 CFR 50.67, as appropriate, even when operating with only partial effectiveness (Chapter 6). The following systems are provided:

1. The steel-lined, reinforced concrete containment structure provides a highly reliable barrier against the escape of radioactivity. The structure and all penetrations, including access openings and ventilation ducts, are of proven design.

2. The emergency core cooling systems cool the core by injecting borated water into the reactor coolant loops from the accumulators, the high- and low-head safety injection pumps, and the recirculation spray pumps (during the recirculation phase following a loss-of-coolant accident (LOCA).

3. The quench spray subsystem of the containment depressurization system provides a spray of borated water to the containment atmosphere. The recirculation spray system provides a spray of borated water buffered by the containment sump pH control system to the containment atmosphere. Following the DBA, the containment pressure is reduced by the containment depressurization system. The sodium tetraborate in the containment sump pH control system combined in the recirculation spray removes iodine from the containment atmosphere. Subsequent long-term cooldown and depressurization is accomplished by the recirculation spray system.

4. Radioactive leakage through containment penetrations to contiguous plant areas is removed by the supplementary leak collection and release system. The effluent is monitored for activity and discharged to the atmosphere at a release point above the containment structure.

5. The post-DBA hydrogen control system has the capability of purging a portion of the containment atmosphere to the atmosphere outside containment following a DBA. The containment spray system provides mixing of the containment atmosphere.

6. The containment isolation system isolates pipe lines which penetrate the containment boundary in accordance with Appendix A of 10 CFR 50, General Design Criteria 55 through 57, so that in the event of a LOCA, radioactivity is not released to the environment.

7. The habitability system for the main control room is provided to ensure that the control room operators are able to remain in the area and operate the nuclear power unit safely under all conditions, including during and following a postulated DBA.

1.2.11 Cooling Water and Other Auxiliary Systems

Safety-related auxiliary systems are as follows:

1. The service water system transfers heat from the primary component cooling water system and other safety-related systems to the ultimate heat sink. This system operates during all normal, upset and faulted conditions.

2. The primary component cooling water system is an intermediate cooling system provided to transfer heat from the reactor auxiliary systems and from systems containing potentially radioactive liquids. The primary component cooling water system provides cooling water for systems and components required for a safe shutdown.

3. The emergency diesel generator fuel oil storage and transfer system is designed to store and supply sufficient fuel oil for seven days of continuous operation of each diesel engine. Independent emergency diesel generator cooling water systems, redundant starting air systems, and lubrication systems, are provided for each diesel.

4. Air conditioning and ventilation systems for safety-related areas control ambient air temperature and provide a suitable environment for personnel and equipment, with features to provide protection against the spread of airborne radioactive contamination. Areas subject to radioactive release have provisions for particulate and gaseous radiation monitoring and filtration of the ventilation exhaust air.

5. The spent fuel pool cooling system removes residual and decay heat from the spent fuel stored in the fuel pool.

6. The functions of the chemical and volume control systems include the control of boron concentrations in the RCS, maintenance of the proper RCS inventory, removal of fission and corrosion products, water chemistry control and continuous supply of filtered water to the reactor coolant pump seals. Portions of the system also provide emergency core cooling following a postulated accident.

7. The auxiliary feedwater system serves as an emergency backup for supplying feedwater to the secondary side of the steam generators upon loss of normal feedwater.

8. The residual heat removal system removes residual and decay heat from the core during reactor cooldown at RCS temperatures of 350°F and below.

Auxiliary systems which are nonsafety-related are as follows:

1. The boron recovery system stores and processes borated radioactive water from the RCS. The system employs degasifiers, evaporators, filters, and demineralizers to produce primary grade water and concentrated boric acid

solution for reuse in plant or disposal. Some of the equipment is shared with BVPS-1.

2. The vent and drain system collects hydrogenated or aerated fluids from various systems and transfers them to either the boron recovery system or the appropriate waste system.

3. The compressed air systems supply service and instrument air required for normal operation (Section 9.3.1).

4. The fire protection system includes a water system shared with BVPS-1, a low pressure CO_2 system, a Halon system, portable fire extinguishers, and a smoke detection system.

5. The fuel pool purification system clarifies and purifies the water in the fuel pool, the refueling cavity, the transfer canal, and the refueling water storage tank.

6. The sampling systems provide the capability to collect representative reactor and steam plant liquid and gaseous samples at the sampling sinks for laboratory analysis.

7. The demineralized water and cask washdown system is used during all modes of operation to supply high quality water to various reactor plant and turbine plant systems for makeup, sample sink flushing, hose stations for decontamination and other miscellaneous services requiring demineralized water. This system is connected to the BVPS-1 demineralized water system which provides all demineralized water for both units.

8. The primary grade water system is a storage and distribution system that supplies reactor plant auxiliary systems exclusively. It also supplies makeup water to the reactor plant. The storage tanks and pumps for this shared system are located in BVPS-1.

9. The chilled water system provides chilled water to the containment atmosphere recirculation coolers and to various building air cooling equipment.

10. The turbine plant component cooling water system is an intermediate cooling system provided to transfer heat from the turbine plant equipment to the service water system.

1.2.12 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.

Removed in Accordance with RIS 2015-17

FIGURE 1.2-1
SITE PLAN
BEAVER VALLEY POWER STATION UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

K:\u2\ufsar_fig.1.2-1.dgn

1.3 COMPARISON TABLES

The comparisons with other stations provided herein reflects the status of Beaver Valley Power Station – Unit 2 at the time of the issuance of the Operating License. This section 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.

1.3.1 Comparison with Similar Facility Designs

Beaver Valley Power Station - Unit 2 (BVPS-2) utilizes proven mature designs. The nuclear steam supply system (NSSS) is of proven design and incorporates systems, equipment, and technology which have been successfully applied in more than 100 units designed by Westinghouse Electric Corporation. The balance of the unit, including the containment structure, was designed and constructed by the Applicant through its agent, Stone & Webster Engineering Corporation (SWEC). The SWEC design incorporated mature design concepts which had been utilized in nine operating nuclear power plants and seven nuclear plants which were in various stages of design, U.S. Nuclear Regulatory Commission (USNRC) review, and construction.

In Table 1.3-1, the general design features of BVPS-2 are compared with those of Beaver Valley Power Station - Unit 1 (BVPS-1), Millstone Unit 3 (Northeast Utility Company), and North Anna Units 1 and 2 (Virginia Electric Power Company). The plant comparison follows the general outline of the Final Safety Analysis Report (FSAR) chapters and is based on a single unit of each design. The USNRC has reviewed these designs extensively; BVPS-2 design incorporates the experience gained in these applications.

1.3.1.1 Comparison of Nuclear Steam Supply Systems

The NSSS for BVPS-2 is similar to that of the other units, except for power level differences. In addition, Millstone Unit 3 has four reactor coolant loops while the other units each have three loops.

1.3.1.2 Comparison of Engineered Safety Features

The engineered safety features (ESF) compared are the emergency core cooling system (ECCS), containment heat removal system, containment combustible gas control system, containment isolation system, control room habitability, and the emergency filtration system. The ESF are the same, except BVPS-2 and Millstone Unit 3 utilize two of the recirculation spray pumps to inject recirculated containment sump water as part of the ECCS. The BVPS-2 recirculation spray pumps also supply the high head safety injection pumps in the ECCS recirculation mode. Beaver Valley Power Station - Unit 1 and North Anna Units 1 and 2 utilize low head safety injection pumps to perform this function.

1.3.1.3 Comparison of Containment Concepts

The containment concept, as shown by the comparison of parameters in Table 1.3-1, is the same as that of the other plants listed.

These plants have already been extensively reviewed and approved by the USNRC for operation.

1.3.1.4 Comparison of Instrumentation Systems

Instrumentation and controls are functionally similar to those at the other plants. The term functionally similar is intended to mean similar in the basic operating and safety functions of the compared systems to which this applies. The specific features of BVPS-2 design are shown in detail and described in applicable sections of the FSAR.

1.3.1.5 Comparison of Electrical Systems

Sections 8.2 and 8.3 of Table 1.3-1 provide a summary comparison of the electrical systems and parameters. While the transmission systems and onsite power systems differ due to utility preference, the emergency power systems, ac vital bus systems, and 125 V dc systems are similar in design.

1.3.1.6 Comparison of Waste Management Systems

Sections 11.2, 11.3, and 11.4 of Table 1.3-1 provides a summary comparison of the waste management systems. The liquid systems are functionally similar for all units compared. The gaseous waste systems are functionally similar for all units compared except that North Anna Units 1 and 2 use recombiners for gaseous waste volume reduction and all other designs utilize the charcoal delay bed concept for radioactive gas management. Beaver Valley Power Station - Unit 2 utilizes a prefilled, cement-in-drum solid waste system, while BVPS-1 uses an in-line, cement, solid waste system. The other plants use an urea-formaldehyde or Dow process binder solidification agent.

1.3.1.7 Comparison of Other Nuclear Plant Systems

The auxiliary systems (fuel pool cooling, component cooling water, service water, and boron recovery systems) are functionally similar for all units compared. Some differences occur due to siting, plant arrangement, and system design; however, the design basis for the auxiliary systems is essentially the same. In addition, North Anna Units 1 and 2 share the same fuel pool cooling and purification system while BVPS-1 and BVPS-2 share tankage subsystems of the boron recovery system.

1.3.1.8 Comparison of Structural Design Characteristics

Sections 2.1, 2.5, 3.3 and 3.8 of Table 1.3-1 compare the BVPS-2 structural design criteria with those of the other plants. Some differences occur due to different site conditions. However, the basic parameters that define structural loadings are essentially the same.

1.3.2 Comparison of Final and Preliminary Information

The significant changes between the preliminary and final design of BVPS-2 are listed in Table 1.3-2. The sections that address these changes are identified by their FSAR and PSAR section numbers. These changes have occurred since the submission of the PSAR and prior to issuance of the operating license. They have been approved and controlled in accordance with administrative procedures and are within the scope of the principal design criteria. New systems and equipment that are post-PSAR are also included in the listing.

1.3.3 References for Section 1.3

U.S. Nuclear Regulatory Commission (USNRC) 1981. Asymmetric Blowdown Loads on PWR Primary Systems; Resolution of Generic Task Action Plan A-2. NUREG-0609.

USNRC 1981. Standard Review Plan For the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800.

Tables for Section 1.3

(Tables in Section 1.3 are historical)

TABLE 1.3-1

DESIGN COMPARISON (HISTORICAL)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Introduction					
Reactor type	1.1	PWR	Same*	Same*	Same*
Reactor manufacturer	1.1	Westinghouse	Same*	Same*	Same*
Site Characteristics					
Exclusion area boundary (minimum) (ft)	2.1	1,500	2,000	1,795	4,432
Low population zone (mi)	2.1	3.6	3.6	2.4	6
Safe shutdown earthquake (horizontal) (g)	2.5	0.125	0.125	0.17	0.12
Operating basis earthquake (horizontal) (g)	2.5	0.06	0.06	0.09	0.06
Structural Design Category I					
Normal wind (mph)	3.3	80	Same*	115	Same*
Tornado region	3.3	1	1	1	1
Foundation type	3.8	Sand and gravel	Sand and gravel	Bedrock	Rock
Reactor					
Nominal core power (MWt)	4.1	2,689	2,689	3,411	2,775
Fuel	4.2	17 x 17	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Reactivity control	4.2	Reactor control rods and boric acid shim	Same*	Same*	Same*
Nuclear design	4.3	Slightly enriched U0 ceramic pellets in Zircaloy-4 tubing	Same*	Same*	Same*
Reactor Coolant System and Connected Systems and Equipment	5.1				
Reactor vessel	5.3	Cylindrical with welded hemispherical bottom head and removable hemispherical top head	Same*	Same*	Same*
Reactor coolant pumps	5.4.1	3 single-speed centrifugal units driven by air-cooled 3-phase induction motors	Same*	Same* except that it has 4 RCPs due to four loop design	Same*
Steam generators	5.4.2	Vertical U-tube	Same*	Same*, except 4 units	Same*
Residual heat removal system	5.4.7				
Number of pumps		2	Same*	Same*	Same*
Number of heat exchangers		2	Same*	Same*	Same*
Pressurizer	5.4.10	Vertical cylindrical vessel using electric heaters for maintaining system pressure	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Engineered Safety Features	6.2				
Containment	6.2				
Type	6.2	Subatmospheric (9-11 psia)	Same*	Same*	Same*
Design pressure (psig)	6.2	45	Same*	Same*	
Design leak rate (percent per day)	6.2	0.1	Same*	0.9	Same*
Containment heat removal systems	6.2.2	Quench spray system, recirculation spray system	Same*	Same*	Same*
Containment isolation system	6.2.4	Complies with General Design Criteria 54,55,56, and 57	Same*	Same*	Same*
Emergency core cooling system	6.3	Injection of borated water by accumulators, charging/HHSI pumps, and LHSI pumps during injection phase; recirculation of spilled coolant from containment sump by recirculation and charging/HHSI pumps	Same* as BVPS-2, except LHSI pumps recirculate the containment sump water	Same* as BVPS-2, except that the RHR pumps perform the same function as the LHSI pumps	Same as BVPS-2, except LHSI pumps recirculate the containment sump water

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Control room habitability	6.4	Radiation shielding, control room pressurization system, emergency air filtration, air-conditioning and ventilation, portable fire protection, personnel protective equipment and first aid, food and water storage, utility and sanitary facilities. Some components and systems are shared with BVPS-1	Same*	Radiation shielding, control room pressurization system, emergency air filtration, air-conditioning and ventilation, portable fire protection, personnel protective equipment and first aid, food and water storage, utility and sanitary facilities	Radiation shielding, control room pressurization system, emergency air filtration, air-conditioning and ventilation, fire protection, personnel protective equipment and first aid, food and water storage, utility and sanitary facilities, remote air intakes
Emergency filtration systems	6.5	Control room area pressurization filtration system and supplementary leak collection and release system used to mitigate the consequence of an accident	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Instrumentation and Controls					
Reactor trip system	7.2	Process instrumentation and control system, nuclear instrumentation system, solid state logic protection system, reactor trip switchgear, manual actuation circuit	**	**	**
Engineered safety feature systems	7.3	Process instrumentation and control system, solid state logic protection system, engineered safety features test cabinet, automatic transfer from injection phase to recirculation phase	**	**, Same* except manual transfer from injection phase to recirc-ulation phase	**
Systems required for safe shutdown	7.4	Monitoring indicators, controls, pumps, fans, diesel generators, valves, and heaters	**	**, Same* except has the capability for a safety grade cold shutdown from the auxiliary shutdown panel	**
Safety-related display instrumentation	7.5	Feedwater and steam systems parameters, containment pressure, RWST water level, pressurizer water level, containment recir-	**	**	**

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
		culation sump level, nuclear instrumentation, reactor coolant system parameters, reactor control system parameters			
Other safety systems	7.6	Instrumentation and control power supply system, ESF protection channels power supply, RCS loop isolation valve interlocks, residual heat removal isolation, accumulator motor-operated isolation valve, switchover from injection to recirculation, refueling interlocks	**	Instrumentation and control power supply system, ESF protection channels power supply, residual heat removal isolation, accumulator motor-operated isolation valve, switchover from injection to recirculation, refueling interlocks	Instrumentation and control power supply system, ESF protection channels power supply, RCS loop isolation valve interlocks, residual heat removal isolation, accumulator motor-operated isolation valve, switchover from injection to recirculation, refueling interlocks
Control systems not required for safety	7.7	Reactor control system, rod control system, plant control system interlocks, pressurizer pressure control, pressurizer water level control, steam generator water level control, turbine bypass control, incore instrumentation. Designed for 85% loss of external electrical	**	Same*, except 50% load rejection capability without reactor trip	**

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
		load without tripping the reactor and for turbine trip below 70% power without reactor trip			
Electrical Power					
Transmission system to site	8.2.1	3 @ 345 kV lines 3 @ 138 kV lines	Same*	3 345 kV lines	3 500 kV lines
AC power system	8.3.1				
Unit main transformer		1 @ 945 MVA	Same*	2 @ 630 MVA	3 @ 330 MVA
Unit station service transformer		2 @ 32 MVA	Same*	1 @ 40 MVA 1 @ 50 MVA	3 @ 20 MVA
System station service transformers (reserve)		2 @ 32 MVA	Same*	1 @ 45 MVA 1 @ 50 MVA	3 @ 30 MVA
Emergency power system	8.3.1				
Emergency 4.16 kV buses		2 @ 1,200 amp	Same*	2 @ 2,000 amp	2 @ 1,200 amp
Diesel generator sets (2,000 hr rating)		2 @ 4,535 kW	2 @ 2,850 kW	2 @ 5,335 kW	2 @ 3,000 kW
AC vital bus system	8.3.1				
Inverters		4 @ 20 kVA	Same*	4 @ 15 kW	3 @ 15 kVA 1 @ 20 kVA
Dist. cabinets		4	Same*	4	4

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
125 V dc power system	8.3.2				
Unit batteries (125 V)		4 safety-related 2 @ 1,650 AH 2 @ 1,050 AH 2 nonsafety 2 @ 2,400 AH 2 @ 2,400 AH	4 safety-related 2 @ 1,800 AH 1 @ 1,650 AH 1 @ 1,680 AH 1 nonsafety 1 @ 2,400 AH	4 safety-related 2 @ 1,650 AH 2 @ 750 AH 2 nonsafety 2 @ 2,550 AH	4 safety-related 1 @ 1,500 AH 1 @ 800 AH
Battery chargers		5 safety-related 4 @ 100 amp 1 spare @ 100 amp 2 nonsafety 1 @ 200 amp 1 @ 150 amp	4 safety related 4 @ 100 amp 1 nonsafety 1 @ 150 amp	6 safety-related 2 @ 200 amp 2 @ 50 amp 2 spare @ 200 amp 3 nonsafety 2 @ 200 amp 1 spare @ 200 amp	4 safety-related 4 @ 250 amp 2 spares 2 @ 250 amp
Auxiliary Systems					
Fuel storage and handling					
New fuel storage	9.1.1	Dry storage in steel and concrete structure within the fuel building for 1/3 of a core (53 fuel assemblies) plus 17 spare fuel assemblies	Same*	Same*, but will use spent fuel area for storage of new fuel using the dry storage area as an optional backup location	Common area for both units. Storage the same as BVPS-2 for each unit

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Spent fuel storage	9.1.2	High density poison racks with storage for 1,088 spent fuel assemblies in a reinforced stainless steel-lined pool within the fuel building	High density racks with storage for 833 spent fuel assemb-lies in a reinforced stainless steel-lined pool within the fuel building	Storage for 1,835 spent fuel assemblies with room for 1 full core off-load	Common area for both units with storage for 966 spent fuel assemb-lies in a reinforced concrete pool within the fuel building
Spent fuel pool cooling and cleanup	9.1.3	2 pumps and 2 coolers; 2 pumps, 2 filters, and 1 demineralizer for purification	Similar to BVPS-2	Similar to BVPS-2, except 2 additional filters in the purification system	2 pumps and 2 coolers; 3 pumps, 2 filters, and 1 demineralizer for purification
Fuel handling system	9.1.4	System has provisions to prevent fuel handling and cask drop accidents	Similar to BVPS-2	Similar to BVPS-2	Similar to BVPS-2
Water Systems					
Service water system	9.2.1	2 redundant flow paths supplied by three 50% capacity service water pumps supplying cooling water to primary and secondary component cooling systems, control room cooling, charging pump cooling water system, and rod control area air-conditioning systems. For accident conditions, each SWS pump is a 100% capacity pump with all systems isolated except containment recircula-	3 river water pumps supplying cooling water to primary component cooling water system, control room cooling, charging pump cooling water system, and pump and motor bearing cooling on river and raw water pumps. For accident con-ditions the same as BVPS-2. Turbine plant component cooling water heat exchangers are cooled by a	2 redundant flow-paths, each con-taining two 100% capacity service water pumps, sup-plying cooling to component cooling systems, charging pump cooling, con-trol buidling air-con-ditioning, SI pump cooler and rod con-trol area air-condi-tioning on loss of power. For acci-dent conditions, service water sup-plies containment recirculation spray	2 redundant flow-paths supplied by four 50% capacity service water pumps (normal operation of both units) and two 50% capacity auxiliary service water pumps supplying component cooling systems, control room cooling, charging pump coolers, instrument air compressors, and pipe penetra-tion cooling. For accident conditions, all systems isolated

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
		tion spray coolers, charging pump coolers, control room cooling, and emergency diesel generator cooling. A standby service water system, consisting of two 100% capacity pumps, takes suction from an alternate intake structure and discharges to the redundant service water headers, to provide cooling for unit shutdown and cooldown after loss of the seismic Category I intake structure.	separate river water system	cooler, containment recirculation pump, ventilation units, SI and charging pump coolers, diesel generators, control building air-conditioning and post-accident sample cooler	except containment recirculation spray cooling, charging pump coolers, control room cooling, instrument air compressors, and pipe pene-tration cooling
Ultimate heat sink	9.2.5	Ohio River	Same as BVPS-2	Long Island Sound (Atlantic Ocean)	Service water reservoir with Lake Anna backup
Other water systems	9.2.2.1 9.2.2.2 9.2.3 9.2.4 9.2.6	Primary and secondary plant component cooling water systems, chilled water system, demineralized water makeup, potable and sanitary water systems, and condensate storage facilities.	Same* Same*	Same* Same*	Same* Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Process auxiliaries					
Compressed air systems	9.3.1	2 redundant compressors supply air for instruments and service air system. 2 redundant compressors supply containment instrument air system, and 1 compressor supplies condensate polishing air.	Same as BVPS-2*, except there is no condensate polishing air system and the containment instrument air compressors are inside the containment.	2 redundant compressors supply air for instruments. 1 compressor supplies service air and 2 redundant compressors supply containment instrument air. Also, 2 additional compressors supply air required for cold shutdown.	For both units, 2 compressors supply service air and 2 compressors supply instrument air. For Unit 1, 2 com-pressors supply containment instrument air. For Unit 2, 4 com-pressors supply containment instrument air (2 operating, 2 back-up)
Process sampling system	9.3.2	Collects reactor plant and turbine plant gaseous and liquid samples for chemical and radiochemical analysis	Same*	Same*	Same*
Equipment and floor drainage system	9.3.3	Collects and treats potentially radioactive liquid drainage and associated entrained gases	Same*	Same*	Same*
Chemical and volume control system	9.3.4	Letdown and charging system is used for reactivity control, purification of reactor coolant, RCS inventory control, and provides high pressure flow to the ECCS	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Boron recovery system	9.3.4.6	Evaporative concentration of letdown boric acid and production of primary grade water for recycle (shared tankage located on BVPS-1)	Same*	Same*, except for shared portion of BVPS-2	Same*, except for shared portion of BVPS-2
Air-conditioning, heating, cooling, and ventilation systems					
Control room and control building heating, ventilation, and air-conditioning system	9.4.1	Provides heating, ventilation, and air-conditioning to control room, computer room, process instrument room, equipment room, cable spreading room, and the auxiliary building cable tunnel	Same*	Same*, excluding chiller room and cable spreading area	Provides heating, ventilation, and air-conditioning to control room, office and computer room, process instrument room, relay room, and communications room. Control room emergency bottled air supply system with subsequent filtered emergency ventilation system
Containment atmosphere recirculation system	9.4.7	Maintains controlled environment for personnel and equipment during normal operation	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Other heating, cooling, and ventilation systems	9.4.2 to 9.4.6 9.4.8 to 9.4.16	Fuel building, auxiliary building, turbine building, waste handling building, emergency diesel generator building, condensate polishing building	Same*, except for condensate polishing building	Same*	Same*
Other auxiliary systems	9.5				
Fire protection system	9.5.1	Detects, extinguishes, and mitigates effects of fires that may occur	Same*	Same*	Same*
Emergency diesel generator cooling water system	9.5.5	Maintains diesel generator jacket water within specified temperature limits by service water system	Same*	Same*	Maintains diesel engine jacket water within specified temperature limits by self-contained radiator cooling system
Additional auxiliary systems	9.5.2, 9.5.3, 9.5.4, 9.5.6, 9.5.7, and 9.5.8	Communications systems, lighting systems, redundant emergency diesel generator support systems including: 1 nonsafety (black) diesel generator, fuel oil storage and transfer, starting, lubrication, and combustion air intake and exhaust systems.	Same*, except for the nonsafety diesel generator	Same*, except for the nonsafety diesel generator	Same*, except for the nonsafety diesel generator

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Steam and Power Conversion System					
Turbine generator	10.2	Westinghouse tandem-compound, 4-flow, 1,800 rpm steam reheat machine; 1 double-flow HP turbine and 2 double-flow LP turbines	Same*	Same*, except General Electric 6-flow turbine generators	Same*
Main steam supply	10.3	797 psia, 11.61 x 10 lb/hr steam flow, 518°F	Same*	960 psia, 15.05 x 10 lb/hr steam flow, 540°F	850 psia, 12.2 x 10 lb/hr steam flow, 525°F
Other features of steam and power conversion system					
Main condensers	10.4.1	Double shell, single-pass, divided water box	Same*	Same*	Same*
Main condenser evacuation system	10.4.2	Steam jet air ejectors with auxiliary steam priming ejectors for initial evacuation	Same*	Steam jet air ejectors with vacuum pumps for initial evacuation	Same*
Turbine bypass system	10.4.4	Passes up to 90% of full-load steam flow to allow 85% to 100% step load reduction without a reactor trip. (See Section 10.4.4 for details.)	Passes up to 90% of full-load steam flow to allow up to 100% step load reduction without reactor trip	Passes up to 40% of maximum steam flow to allow up to 50% step load reduction without reactor or turbine trip	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Circulating water system	10.4.5	Removes heat from the main condenser by circulating water through 1 natural draft cooling tower	Same*	Removes heat from the main condenser by circulating water from Long Island Sound	Removes heat from the main condenser by circulating water from Lake Anna
Condensate cleanup system	10.4.6	Full flow condensate polishing demineralizers are provided at the discharge of the condensate pumps. Here powdered resin demineralizers are capable of maintaining a steam generator chemistry well below maximum requirements.	None	Same*	Same*
Condensate and feedwater system	10.4.7	Returns condensed steam from condenser, and drains from regenerative feedwater heaters (6-stage heater cycle), to steam generators while maintaining water inventories throughout system	Same*	Same*	Same*
Auxiliary feedwater system	10.4.9	Supplies necessary cooling water to steam generators for decay heat removal, feedwater line malfunction, or main steam line break	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Auxiliary steam and condensate system	10.4.10	Supplies heating throughout the plant to various heating and processing equipment, and recovers the condensed steam from the equipment served. Normal source of auxiliary steam is main steam. Auxiliary boiler used when reactor not at power	Same*	Same*	Same*, except normal source of auxiliary steam is second point extraction.
Radioactive Waste Management					
Liquid waste management system	11.2				
Type of processing					
Evaporation		Yes	Yes	Yes	Yes
Demineralization		Yes	Yes	Yes	Yes
Filtration		Yes	Yes	Yes	Yes
Treatment of radioactive waste					
High activity waste	11.2	All liquid waste can be evaporated or filtered depending on its activity. The distillate from the evaporator can be demineralized and filtered. There is no	Evaporation and/or demineralization/ filtration	Similar to BVPS-1	Similar to BVPS-1

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
		separation of high and low activity streams prior to liquid waste tanks			
Low activity waste	11.2	See "High activity waste"	Filtration (low activity waste can be routed to the high activity waste tanks and evaporated)	Filtration (evaporation and subsequent operations are optional)	Similar to Millstone Unit 3 with the addition of a clarifier before release to the environment
Steam generator blowdown	10.4.8	Blowdown piped to flash tank where steam is piped to 2nd point heater and liquid is piped to main condenser via 4th point heaters for processing through condensate polishing demineralizers, or liquid is piped to demineralizers for processing prior to routing to the condenser.	Blowdown piped to flash tank and steam is then piped to the third point heaters and liquid is piped to demineralizers for processing prior to routing to the condenser	Similar to BVPS-2 except steam piped to 4th point heater	Blowdown piped to flash tank where steam is piped to roof vent and liquid processed by clarification.
Gaseous waste management systems	11.3				
Type of treatment					
Degasification		Yes (occurs in boron recovery system)	Yes (occurs in boron recovery system)	Yes	Yes (called gas strippers in boron recovery system)
Decay of noble gases in high activity gas streams		Yes	Yes	Yes	Yes

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Filtration of low activity gas streams		Yes	Yes	No	Yes
Recombiners		No	No	No	Yes
Treatment of streams					
Continuous degasification of reactor letdown capability		Yes	Yes	Yes	Yes
Degasification of letdown to boron recovery system		Yes	Yes	Yes	Yes
Degasification of reactor plant gaseous drains		Yes	Yes	Yes	Yes
Decay method for gases stripped in degasifier		Adsorption on charcoal for minimun of 30 days and 2 days xenon and krypton decay, respectively, before recycle or release to atmosphere through BVPS-1 process vent on BVPS-1 cooling tower	Same*	Adsorption on charcoal for minimum of 60 days and 4 days xenon and krypton decay, respectively, before recycle or release to the environment	Recombination of hydrogen in the gaseous waste stream to reduce storage requirements in waste gas decay tanks before release to the environment
Low activity air streams (nonventilation streams)		HEPA/charcoal filter assemblies in the process vent on BVPS-1	Same*	Release through Millstone 1 stack	Similar to BVPS-2
Solid waste management	11.4				
Type of treatment					
Solidification		In-drum	In-line	In-container	In-line (1 waste disposal building for both units)

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Solidification agent		Cement	Cement	Dow process binder	Urea-formaldehyde
Inputs (type) treated					
Boron evaporator bottoms		None; processed by BVPS-1	Yes	Yes	Yes
Waste evaporator bottoms		Yes	Yes	Yes	Yes
Spent beaded resins		Yes	Yes	Yes	Yes
Powdered resins		Yes	No	Yes	Yes
Filtered elements		Yes	Yes	Yes	Yes
Miscellaneous waste (contaminated clothing, tools, paper products, etc)		Yes	Yes	Yes	Yes
Radiation Protection					
Radiation Protection Design Features	12.3				
Shielding		Shielding thickness and coverage determined for each area to assure maximum design dose rates are not exceeded and to prevent activation of components within containment	Same*	Same*	Same*

TABLE 1.3-1 (Cont)

Section and Characteristics	Referenced in Section	BVPS-2	BVPS-1	Millstone (Unit 3)	North Anna (Units 1 & 2)
Ventilation		Ventilation, filtration of ventilation streams, and atmospheric release of ventilation streams to maintain comfortable environment and limit airborne radioactivity below concentration limits of 10 CFR 20, Appendix B	Same*	Same*	Same*

NOTES:

*Where the word 'Same' appears, the design feature of the listed unit is the same as that of BVPS-2.
**Instrumentation and Controls are functionally similar. The term functionally similar is intended to mean similar in the basic operating or safety functions of the compared systems to which this applies. The specific features of the BVPS-2 design are described in detail in applicable sections of the UFSAR.

TABLE 1.3-2

COMPARISON OF FINAL AND PRELIMINARY INFORMATION (HISTORICAL)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
2.2.3 /2.1	Addressed monitoring of main control room for accidental release of chlorine in accordance with Regulatory Guide 1.95.
2.5 /2.5	Discussion of subsurface conditions within main plant area revised due to discovery of zone of loose granular soil that required densification.
3.7B.3.1.2 /None	Former selection of highest elevation curves of amplified response spectra (ARS) for piping analysis now envelopes all applicable ARS curves to assure more conservative analysis. Increased ARS peak broadening from ±15% to +25% -20% to be comparable with equipment analysis.
3.9B.3.2.1 /15.6.2.3	Seal leakage tests, previously addressed as being performed at the same pressure used in the hydrostatic tests, has been changed to read leakage tests performed at performance test pressures.
3.9N.4.4 /None	The control rod drive mechanism (CRDM) seismic supports were upgraded because subsection NF requires that the component supports be the same code classification as the equipment they are supporting. Since the CRDM pressure housings are ASME Code Class 1, the supports must also be Class 1.
3.10B /15.5.3.1	Added Standard Review Plan (SRP) Section 3.10 statement dated 24 November 1975 (USNRC NUREG-75/087) in addition to existing IEEE Standard 344-1971 position.
3.10N /None	The seismic and environmental requalification program has been necessitated by a change in licensing codes and regulations from IEEE Standards 323-1471 and 344-1971 to the methodology of IEEE Standard 344-1975 and the environmental test envelopes and analytical techniques associated with IEEE Standard 323-1974.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
Appendix 3A /None	Piping analysis computer program changed from SHOCK 3, PIPESTRESS, STRESS COMBINER, AND NCCODE to current Stone & Webster Engineering Corporation (SWEC) computer program, NUPIPE-SW.
4.3.2.1 /None	Hafnium conversion made to improve control rod design.
5.2.2.5 /None	Addressed mounting, support, and reaction force capability of pressurizer relief valves.
5.2.3.4.6 /None	Analysis of socket welded fittings made to show compliance of the socket weld, of the attached piping, and of the branch nozzle to the requirements of an ASME Section III fatigue evaluation.
5.2.5 /4.2.4 & 9.7	Provisions made for continuous monitoring of leakage from reactor containment pressure boundary into containment sumps with Class 1E narrow range level transmitters and programmable controllers.
5.3 /None	Reactor vessel weld material program provides the information necessary to satisfy USNRC Bulletins 78-12 and 78-12A.
5.3.3.1 /4.4.2	The reactor pressure vessel insulation provides a thermal neutron shielding to reduce activation of the upper vessel, nozzles, and neutron shield tank.
5.4.1 /None	As a result of the change to the surge and protection for the reactor coolant pump (RCP) motors, the RCP turn-to-turn insulation is rated to accept 4,160 V, indicating that surges 2.5 times higher than expected could be accommodated during pre-operational tests without harming the motors.
5.4.3.2 /None	Thermal sleeves in the reactor coolant loop branch nozzles have been deleted to simplify the nozzle design and to show technical improvement.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
5.4.4.1 /None	Modifications to steam generators integral flow restrictor to provide significant design enhancement. Benefits include improved mass and energy calculation, reduced blowdown during a steam break, and reduced steam break size from 4.6 ft^2 to 1.4 ft^2.
5.4.7.2 /4.1.2.1	The change from a one train RHR system to a two train RHR system provides maximum operational flexibility concurrent with maintenance capabilities.
5.4.7.2.1 /4.1.2.1	The RHR interlocks have been revised to reflect an auto closure set point on increasing reactor coolant system (RCS) pressure >750 psig to allow full relief valve potential and preclude premature isolation of the RCS low pressure relief projection.
5.4.7.2.2 /None	The residual heat removal (RHR) miniflow change from automatic to manual provides protection for the RHR pumps from deadheading against high discharge pressure.
5.4.7.2.3 /None	The reactor vessel head vent change is a part of the cold shutdown modification discussed in UFSAR Section/PSAR Reference 5.4.7.2.6, and provides venting capability from the reactor vessel head should this action become necessary.
5.4.7.2.6 /None	The extensive change in the approach to safety grade cold shutdown provides a significant improvement in the ability to bring the plant to a cold shutdown condition in the event of an abnormal occurrence.
5.4.11.2.1 /None	Pressurizer relief tank cooldown and drainage provide to protect against overfilling of tanks and introduction of water into the degassification system.
Appendix 5A /None	Same as for UFSAR Section/PSAR Reference 5.4.7.2.6.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
6.2.1.1.3.1 /6.4	Refueling water storage tank (RWST) capacity increased because of containment depressurization calculation.
6.2.1.1.3.7 /10.3.5	Automatic closure of motor-operated discharge valves in auxiliary feedwater lines to each steam generator replaced with cavitating venturies to limit flow to the steam generator affected by a pipe break.
6.2.1.3 /None	Same as for UFSAR Section/PSAR Reference 7.3.1.1.
6.2.2.2.1 /6.4	Quench spray system (QSS) header diameters and design of nozzles changed to provide more uniform delivery of smaller droplets spray for better thermal effectiveness.
6.2.4 /5.2	Changes to containment isolation system include: 1. Elimination of pressurizer dead weight calibrator. 2. Increased number of penetrations dedicated to containment leakage monitoring from one to two. 3. Added new penetration for chemical addition line to QSS.
6.3.2 /6.3.2.1.2	To provide for reduced plant maintenance/radiation exposure, without compromise in plant safety, the boron injection tank was eliminated.
6.3.2.2 /6.3.2.2.2	The automatic transfer to recirculation design provides for those essential operations which establish a flow path from the containment sump to the RCS to continue reactor core cooling flow without operator action. The PSAR design required two manual operator actions.
6.3.3 /6.3.2.2.1	The coincidence of low pressurizer water level and pressure for the safety injection logic was deleted, and the existing safeguards actuation logic was converted to two-out-of-three low

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	pressurizer pressure signals. This modification meets the signal failure criterion.
6.4.2.3 /9.13.4	Added quick-acting chlorine detectors to outdoor air intakes of main control room envelope to enhance its habitability, as determined from toxic chemical analysis. Added redundant, Category 1 area radiation monitor to isolate the control room area environment to maintain control room habitability during design basis accident.
6.5.2 /6.4	Use of positive displacement pump to add sodium hydroxide solution to quench spray, instead of gravity feed system proposed earlier, to improve control of chemical addition.
6.5.3 /5.3.1	Changed elevated release point for fission product control from top of cooling tower to top of containment.
7.2.1.1.2 /None	Steam line break protection system changed to enhance plant safety and to increase plant availability by preventing spurious safety injection actuation. The analysis for the deletion of reactor trip following a turbine trip below 70 percent power is addressed in Section 15.2.3. The deletion of coincidence of low pressurizer water level and pressure for the safety injection logic is addressed in FSAR Section/PSAR Reference 6.3.3.
7.2.2.2.3 /7.2.2.2.1	The deletion of the RCP breaker position allows the breaker open indication to be eliminated as a source of reactor trip when this indication is from a single relay source, without affecting plant safety. That is, this change can potentially eliminate nuisance trips.
7.3.1.1 /7.3.2.1.1	Implementation of N-1 loop steam line break protection system that actuates safety injection and steam line isolation on receiving two-out-of-three low steam line pressure signals from any one steam line. The protection logic will increase the

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	margins to the trip set points during the start up phase where problems are most likely to occur.
	The modification to the solid state protection system incorporates a manual initiation of steam line isolation at the system level. Amendment 12 of the PSAR, Question 7.5, committed this design modification to the USNRC.
7.4.1.2.1 /None	A process equipment modification was made to nuclear steam supply system instrumentation signals to the emergency shutdown panel.
7.4.1.3 /None	In the event of an exposure fire in the instrumentation and relay room, cable spreading room, west communication room, and the cable tunnel, an alternate shutdown panel is provided in the auxiliary building to provide a means of alternative shutdown capability that bypasses all equipment and electrical cables located in the fire areas.
7.5.3 /None	Bypass inoperable status indication system added to monitor Beaver Valley Power Station - Unit 2 (BVPS-2) safety systems for operability prior to an accident condition.
	Safety parameter display system added to BVPS-2 to monitor specific plant parameters used in the emergency response facilities and nuclear data link.
7.6.5 /7.6.1 & 7.6.5	Transfer from emergency core cooling system injection mode to recirculation mode is automatic instead of manual, and depends on RWST level and a safety injection/CIB signal. Also see change for UFSAR Section/PSAR Reference 6.3.2.2.
7.6.2.1 /7.6.1	The suction valves of the RHR pumps can be powered from either Class 1E bus. This ensures that after an isolation that the valves can be opened for RHR pump operation. Interlocks are provided to prevent paralleling of the two Class 1E power sources.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
7.6.7.2 /4.2.2	Incorporation of RCS cold overpressurization mitigation to provide for potential transients by utilizing the existing power-operated relief valves with modifications to their actuation logic.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
7.7.1.2 /3.3.2.5.3	Deletion of part length control rods made to comply with regulatory restrictions on use of part length rods.
9.1.1 /9.12	Capacity of hoist for handling new fuel assemblies increased from 5 tons to 10 tons.
9.1.2 /9.12	Changes to storage and handling of spent fuel include:
	1. Capacity of spent fuel pool increased from 274 spent fuel assemblies to 1,059 assemblies.
	2. Assurance of an effective multiplication factor, K_{eff} of 0.90, formerly accomplished by center-to-center spacing of spent fuel assemblies, is now attained by use of neutron absorbing material, where K_{eff} will be less than 0.95.
	3. Fuel building hoists changed from two hoists of 2 tons and 5 tons capacities to two hoists, each of 10 tons capacity.
	4. Failed fuel containers are not being provided now; however, provisions have been made for future installation.
9.1.4.2.3.4 /None	Fuel transfer system upgraded to help ensure trouble-free operation during initial refueling and core loading operations. The replacement enhances the reliability and maintainability of the fuel transfer system.
9.1.5 /None	Provided description of new system, overhead heavy load handling system, that is used throughout plant.
9.2.1 /9.9	Station service water system changes include:
	1. Service water system load changes for normal operation include:
	a. Component cooling water heat exchangers increased from one to two,

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	b. Added air-conditioning unit to safeguards area,
	c. Added one set of cooling coils to main steam valve house,
	d. Charging pump lube oil coolers decreased from two to one per pump, and
	e. Added motor control center cooling coils.
	2. Service water system load changes for post-DBA operation are:
	a. Added a set of cooling coils for motor control center,
	b. Added rod control area air-conditioning unit,
	c. Added safeguards area air-conditioning unit, and
	d. Added main steam valve house cooling coils.
	3. The former provisions for chemical addition to recirculation coolers for corrosion control are deleted.
	4. All equipment with safety-related function and not located above probable maximum flood level is now flood-protected.
	5. Primary component cooling water temperature is not controlled by flow of service water through recirculation coolers. Water temperature now a function of service water flow.
	6. Portion of flow from cooling tower makeup diverted to service water discharge for reasons of avoiding sediment buildup in discharge lines.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
9.2.2.1 /9.4	Primary component cooling water system has these criteria and equipment changes: 1. Cooling water pumps are now environmentally qualified for normal, abnormal, and accident conditions operation, as addressed in Section 3.11. 2. Additional equipment that utilizes the cooling water of this system include: primary drains coolers, auxiliary steam degasifier coolers, radiation monitors, overhead gas compressors, and containment instrument air compressors. 3. The containment air recirculation cooling coils have been deleted from Figure 9.4-4.
9.2.4 /9.11	Potable water will originate from wells instead of from Beaver Valley Power Station - Unit 1 (BVPS-1) treated river water supply, with head pressure provided by an elevated storage tank.
9.2.7 /10.3.9	Turbine plant component cooling water system changes are: 1. Control valve added to discharge of each heat exchanger. 2. Makeup water to system is from demineralized water system instead of condensate system. 3. Surge tank capacity increased from 2,900 gallons to 3,173 gallons. 4. Total dynamic head rating of dual volume horizontal pumps changed from 120 feet to 160 feet, with same capacity of 11,000 gpm. 5. Additional equipment served by this system include: tank drain cooler, condensate polishing air compressor, evaporator reboiler drain coolers, and condensate polishing sample sink cooler.
9.2.8 /9.11	Primary grade water storage tank addressed earlier no longer in present design.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
9.3.1 /9.8	Changes to compressed air system include: 1. Containment instrument air compressors, air dryer, and air receiver tanks (including new one of 25 ft^3 capacity) now located in main steam line valve area instead of within containment, although suction is still taken from within containment. 2. Containment instrument air compressors rating changed from 110 psig at 50 scfm discharge to 120 psig at 30 scfm. Station air compressors rating increased from 110 psig at 358 scfm to 120 psig at 728 scfm. 3. Addition of refrigerant-type air dryer with desiccant filter bypass arrangement to containment instrument air system. 4. Addition of desiccant-type filter bypass arrangement to station instrument air dryer. 5. Both station air and containment instrument air compressors now have capability of operating with power from the onsite nonsafety diesel generator. 6. Addition of third system, condensate polishing air system.
9.3.2.1 /9.6	Reactor plant and process sampling system changes include: 1. Addition of conditioning rack for blowdown grab samples. 2. Addition of on-line pH and Na conductivity monitors for each steam generator. 3. Addition of sampling capabilities for: gaseous waste storage tanks, primary drains transfer tanks, primary grade water, gaseous waste surge tank, letdown flow, fuel pool, cesium removal ion exchangers, and RHR system liquids.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	4. Local sampling capabilities provided for RWST, boric acid tank, and QSS chemical addition tank.
	5. Provisions made for exhausting potentially radioactive sample panel gases to supplementary leak collection and release system.
9.3.2.3 /None	Addressed new post-accident sampling system.
9.3.3 /9.7 & 4.2.4	Changes to equipment and floor drainage system are:
	1. Reactor coolant system loop drains formerly directed to both primary drains transfer and pressurizer relief tanks now directed to primary drains transfer tank only.
	2. Valve stem leakoffs now go directly to primary drains transfer tank in containment instead of to pressurizer relief tank.
	3. Addressed containment sump monitoring with Class 1E level transmitters and programmable controller.
9.3.4 /None	The chemical and volume control system continuous degasification allows for admittance of hydrogen whenever a low pressure alarm condition exists.
9.3.4.6 /9.2	Classification of that portion of boron recovery system from degasifiers and downstream piping to gaseous waste system downgraded from Safety Class 3 to non-nuclear safety (NNS) class.
9.4.1 /9.13.4	Same as for UFSAR Section/PSAR Reference 6.4.2.3.
9.4.3 /9.13.2	Changes to auxiliary and radwaste area ventilation system include:
	1. Ambient air temperature lower limit for ventilation equipment area and general areas increased from 60°F to 65°F.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	2. Air temperature limits for shielded cubicles at all levels (except those in radwaste area) changed from between 65°F and 104°F to between 65°F and 120°F. For radwaste area cubicles, the limits are between 65°F and 104°F.
	3. Upper temperature limit during a DBA for component cooling water pump area and charging pump cubicles increased from 104°F to 120°F.
9.4.6 /9.13.7	Design temperature of the emergency diesel generator building ventilation system is 122°F. Secondary supply fans have been added in order to meet environmental qualification requirements.
9.4.9 /None	Upgraded main steam and feedwater valve area ventilation system to safety-related Safety Class 3 to maintain environmental qualification temperatures.
9.4.11 /6.2.4	Area design temperatures for safeguards area ventilation system following a DBA reduced from 140°F for first hour and dropping to 120°F after shutdown of quench spray and low head safety injection pumps, to 120°F maximum at beginning of DBA and below 120°F after 3 hours when aforementioned pumps are shut down.
9.4.13 /9.15	Added filtration and exhaust subsystem for new gaseous waste storage tank and cask washdown areas to decontamination building ventilation system.
9.4.15 /10.3.3.1	Addressed gland seal steam exhaust ventilation system that filters and monitors noncondensable gases before being discharged to environment, previously discharged directly to atmosphere.
9.4.16 /None	Addressed new condensate polishing building ventilation system, a NNS class system.
9.5.4 /9.14	Changes to emergency diesel generator fuel oil storage and transfer system include:
	1. Total capacity of two fuel oil storage tanks increased from 40,000 gallons to

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	58,000 gallons to provide for 7 days full load operation of each diesel generator.
	2. Cross-connection piping between fuel oil storage tanks now eliminated.
	3. Fuel oil transfer pumps, formerly positive displacement type, mounted outside of storage tanks, and with capacity of 10 gpm, are now vertical centrifugal type, mounted inside of tanks, and with capacity of 40 gpm.
	4. Discharge line relief valves replaced with recirculation orifices.
	5. Fuel strainers, formerly located in common fuel lines to diesel generators, now located in discharge of each oil transfer pump.
	6. Engine-mounted fuel oil tank and day tank, each of 550-gallon capacity, replaced with single 1,100-gallon floor-mounted day tank for each diesel generator.
	7. All storage tanks and associated piping are either indoors or encased in concrete, with no components in direct contact with soil.
9.5.6 /8.5.2	Changes to emergency diesel generator starting system include:
	1. Both air compressors for each diesel generator now motor-driven, as opposed to formerly one motor-driven and one diesel/electric-driven.
	2. Two air receivers now provided for each diesel generator instead of former six air bottles.
10.2.1 /10.3.3.2	Turbine-generator exciter maximum output rating at 1,800 rpm increased from 3,300 kWe and 500 V dc to 3,900 kWe and 525 V dc.
10.3.1 /10.3.1	The residual heat release and atmospheric dump piping and equipment have increased in size and capacity, and the Category I boundary extended to

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	accommodate cold shutdown.
10.3.2 /10.3.1.2	Main steam supply system changes include: 1. Replaced nonreturn valves with main steam isolation valves to prevent reverse flow of steam in event of accidental pressure reduction. 2. Main steam line trip valves changed from swing disk-type to hydraulically-operated ball type, held open by solenoid-operated mechanical latch. 3. Parallel trip valve configuration for turbine-driven auxiliary feedwater pump steam supply changed. Presently, three pairs of series solenoid-operated trip valves are provided, one pair in each of the three 3-inch lines between the 32-inch main steam line and the 3-inch turbine drive common header. 4. Main steam line trip valves changed from hydraulically operated ball-type to pneumatically operated wye pattern globe valve.
10.4.4 /10.3.1	Added cooling tower pumps and low-low T_{avg} permissive interlocks to turbine bypass control valves capabilities.
10.4.5 /10.3.4	Circulating water system changes include: 1. Total dynamic head of cooling tower pumps reduced from 96.4 feet to 73 feet through use of natural draft counter-flow cooling tower with its lesser static lift requirements. 2. Addition of flood indicators to: valve pits on suction side of cooling tower pumps, inlet and outlet sides of condenser, and turbine building retention pit, all with flood alarms to main control room.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
10.4.6 /None	Addressed new condensate cleanup system, a NNS class system, consisting of two subsystems, condensate polishing system and powdered resin dewatering system.
10.4.7.2 /10.3.5	Changes to condensate and feedwater systems include:

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	1. Reduction of two half-size motor-driven steam generator feedwater pumps total dynamic head rating from 1,900 feet to 1,694 feet for same 15,200 gpm flow.
	2. Addition of motor-driven start-up feedwater pump to minimize operation of main feedwater pumps at start-up and low loads. Pump can also be paralleled with a main feedwater pump when either one is out of service.
	3. Addition of two separation drain pumps to supplement heater drain pumps during peak plant operation.
	4. Addition of full flow condensate polishing system downstream of condensate pumps.
	5. Feedwater system containment isolation valve changed from a single motor-operated stop check valve outside containment per supply line to a check valve inside containment with an electro hydraulic-operated isolation valve outside containment.
10.4.8 /11.2.5	Steam generator blowdown system changes include:
	1. Blowdown liquids formerly processed by evaporator facilities now are normally processed by the condensate polishing system after they have been routed to the condenser. Processing by evaporator facilities as a backup has been retained.
	2. Former discussion on providing common collection facility for blowdown from both BVPS-1 and BVPS-2 is no longer applicable. Each plant now has an independent processing system for return of blowdown to the condensers.
10.4.9 /10.3.5	Changes to auxiliary feedwater system include:
	1. Former two turbine-driven auxiliary feedwater pumps with motor-drive backup replaced with three pumps, two half-size motor-driven and one full capacity turbine-driven.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	2. Output rating of former two turbine-driven auxiliary feedwater pumps was 700 gpm each. Output ratings of present three pumps are 375 gpm each for two motor-driven ones and 750 gpm for turbine-driven one.
	3. Former continuous recirculation flow to primary plant demineralized water storage tank (PPDWST) now controlled automatically by recirculation control valve on discharge piping of auxiliary feedwater pumps to maintain minimum flow requirements.
	4. Added cross-connection piping between 140,000 gallon PPDWST and 600,000 gallon demineralized water storage tank (DWST) to provide another source of backup water supply.
	5. Hand-wheel type motor-operated control valves in auxiliary feedwater supply lines replaced with hand-pump hydraulic type electro-hydraulic-operated control valves for more positive control in event of loss of power.
	6. Component cooling water to feedwater pump lube oil coolers, formerly in external piping, now in internal lines that are part of pump design.
	7. Added check valves downstream of control valves in auxiliary feedwater lines outside containment to prevent loss of auxiliary feedwater in event of auxiliary feedwater header rupture. Also added check valve in each auxiliary feedwater line inside containment for containment isolation purposes.
	8. Added cavitating venturi flow elements in common auxiliary feedwater supply lines to limit flow in event of main steam or main feedwater line rupture.
	9. Provided redundant safety-related flow transmitters for each auxiliary feedwater line outside containment.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
	10. Added separate chemical feed connection on individual auxiliary feedwater supply lines.
	11. Chemical addition to PPDWST, formerly supplied from chemical feed tank, now provided by injection of chemicals into suction piping.
	12. Each motor-driven feedwater pump delivers water to separate header while turbine-driven pump can be manually aligned for water delivery to either header.
10.4.10 /10.3.2	Additional equipment served by auxiliary steam and condensate system include:
	1. Degasifier steam heaters,
	2. Evaporator reboilers,
	3. Boric acid batch tank,
	4. Carbon dioxide vaporizer, and
	5. Cask washdown area.
11.2 /11.2.4	Liquid waste management systems have these changes:
	1. Added piping connection from evaporator bottoms cooler to liquid waste drain tanks.
	2. Provided additional discharge points for liquid waste tanks that include steam generator blowdown (SGB) hold tanks, to cleanup filter, and then to cooling tower blowdown.
	3. Expanded capability to process liquid waste by use of SGB evaporators and hold tanks as primary means of processing prior to utilizing other facilities of BVPS-1 and BVPS-2.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
11.3 /11.2.3	Gaseous waste management systems changes include: 1. Downgraded gaseous waste disposal (GWD) system from Safety Class 3 to NNS class. 2. Added seven gaseous waste storage tanks.
11.4 /11.2.6	Changes to solid waste management systems include: 1. Formerly, spent resin was flushed to BVPS-1 (without solidification) directly to shipping casks, dewatered, and sealed. Evaporator bottoms were packaged in casks with cement-vermiculite mixture. Both casks were stored at BVPS-1. Now spent resin sludge and evaporator bottoms are normally packaged in 55-gallon drums preloaded with dry cement, with drums stored in condensate polishing building of BVPS-2. Option of disposal via BVPS-1 dewatering facility is still retained, however. 2. Formerly, used cartridge filter elements were mixed with other solid waste in same drum. Now filter elements are packaged by themselves in drums.
11.5 /11.2.7.2	Additional area, effluent, and process monitors added to process and effluent radiological monitoring system to improve system monitoring capabilities.
12.3.2.1 /11.3.2.1	Changes made to configuration design of supplementary neutron streaming shield to facilitate refueling operations.
12.3.2.5 /11.3.2.2	Modified fuel transfer tube shield design to prevent direct line streaming of radiation from spent fuel through seismic gap.

TABLE 1.3-2 (Cont)

FSAR Section /PSAR Reference	Significant Changes Since PSAR
12.3.4 /11.3	Expanded radiation monitoring system with additional area, effluent, and process monitors to improve system monitoring capabilities and to provide post-accident detection.
15.2.3 /None	The analysis for the deletion of reactor trip following a turbine trip below 70 percent power is required in order to demonstrate reactor safety should the fast bus transfer to offsite power fail following the generator motoring delay on turbine trip.
15.6.5 /None	The loss-of-coolant accident analysis of N-1 loop operation is a prerequisite to plant operation with one loop out of service. The analysis must demonstrate acceptable results for accidents initiated from the steady state N-1 loop mode of operation.
	Baffle to barrel region configuration changed from downflow to upflow to reduce baffle plate and baffle bolt loading, and to minimize the potential for excessive baffle joint jetting.

1.4 IDENTIFICATION OF AGENT AND CONTRACTORS

This section identifies the prime agents and contractors for the design, construction, and operation of the Beaver Valley Power Station - Unit 2 (BVPS-2). This section 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.

1.4.1 Applicant

The Central Area Power Coordinating Group (CAPCO), which is comprised of Ohio Edison Company, The Cleveland Electric Illuminating Company, The Toledo Edison Company, and Duquesne Light Company (DLC), was the Applicant for BVPS-2. The Applicant engaged the contractors identified in Sections 1.4.2 through 1.4.5 to perform engineering, procurements, and construction services for the plant. However, DLC retained overall project responsibility which included engineering; construction; testing; and from the time of initial fuel loading, operation and maintenance of BVPS-2. DLC maintained a technically competent and safety-oriented staff with the proper qualifications, training, and licenses.

Most of the senior personnel in the DLC Engineering and Construction Division and Operations Division were active participants in the design, construction, and operation of the Shippingport Atomic Power Station (SAPS). During the BVPS evaluation, engineering design, construction, testing, and operations phases, the same successful managerial techniques that were developed over the years were employed.

In March 1981, DLC established the Nuclear Division to meet the needs of the company's nuclear power goals and objectives. The support functions of the Quality Assurance Department remained with the Nuclear Construction Division.

Duquesne Light Company was engaged principally in the production, purchase, transmission, distribution, and sale of electric energy. Duquesne Light Company served an area of approximately 800 square miles in Allegheny and Beaver Counties. This area, which includes the city of Pittsburgh, is located in southwestern Pennsylvania and has a population of approximately 1,500,000.

A summary of previous experience in the field of power generation shows that DLC was technically qualified to engage in the proposed activities. Duquesne Light Company had been involved in various nuclear projects for over 30 years. They include the following:

1. A study of the feasibility and economics of constructing a nuclear reactor for the use of atomic energy and its possible implementation in the commercial power industry - 1952.

2. Establishment of the Atomic Power Development Department - 1953.

3. An agreement with the U.S. Atomic Energy Commission (USAEC) to operate and maintain SAPS and to conduct an extensive testing program prescribed by the USAEC - 1954.

4. Participation in nuclear research and development programs such as the Westinghouse Electric Corporation Liquid Metal Fast Breeder Reactor project, reactor vessel inspection study, and the plutonium recycle fuel fabrication.

5. The design, construction, and operation of Beaver Valley Power Station - Unit 1 (BVPS-1) - 1976.

1.4.2 Architect Engineer

Stone & Webster Engineering Corporation (SWEC) provided engineering design and construction management services for BVPS-2. The Corporation maintained offices in Boston, Massachusetts, Cherry Hill, New Jersey, Denver, Colorado, New York, New York, and Houston, Texas, with a total manpower resource pool of over 12,000 employees. Approximately 700 engineers, designers, construction specialists, and clerical and administrative personnel were assigned to the BVPS-2 project during its peak level of activity. In addition to its project-dedicated staff, SWEC utilized specialists in various engineering disciplines to ensure that BVPS-2 was designed in accordance with industry codes and standards and met the requirements of the applicable federal, state, and local regulations for commercial nuclear power plants.

SWEC has been associated with the installation of over 77,000 MW of hydroelectric, nuclear, and fossil-fired electric generating facilities for the electric utility industry. The Corporation's experience in the field of nuclear energy dates from 1942 when it participated in initiating the first self-sustaining nuclear chain reaction at the University of Chicago. Since 1954, SWEC has designed and/or constructed the following nuclear power stations which are either presently operating or have operated successfully:

1. Shippingport Atomic Power Plant of Duquesne Light Company and ERDA

2. Army Package Power Reactor (APPR, also known as Al)

3. Yankee Nuclear Power Station of Yankee Atomic Power Company

4. Carolinas-Virginia Tube Reactor of the Carolinas-Virginia Nuclear Power Associates, Inc.

5. Haddam Neck Plant of Connecticut Yankee Atomic Power Company

6. Nine Mile Point Nuclear Station Unit 1 of Niagara Mohawk Power Corporation

7. Maine Yankee Atomic Power Station of Maine Yankee Atomic Power Company

8. Surry Power Station Units 1 and 2 of Virginia Electric and Power Company

9. James A. Fitzpatrick Nuclear Power Plant - Unit 1 of the Power Authority of the State of New York

10. North Anna Power Station - Units 1 and 2 of Virginia Electric and Power Company

11. Beaver Valley Power Station - Unit 1 of Duquesne Light Company

In addition, SWEC had under design and construction at that time the following nuclear power stations:

1. North Anna Power Station - Unit 3 of Virginia Electric and Power Company

2. Millstone Nuclear Power Station - Unit 3 of Northeast Utilities Service Company

3. Shoreham Nuclear Power Station - Unit 1 of Long Island Lighting Company

4. Nine Mile Point - Unit 2 of Niagara Mohawk Power Corporation

5. River Bend Power Station - Units 1 and 2 of Gulf States Utilities.

Also, SWEC provided construction management services for the Demonstration Liquid Metal Fast Breeder Reactor Plant (Clinch River Project) and the Department of Energy's Gas Centrifuge Uranium Enrichment Plant.

1.4.3 Nuclear Steam Supply System Manufacturer

Westinghouse Electric Corporation (Westinghouse) was responsible for supplying the NSSS and fuel for BVPS-2.

Westinghouse has designed, developed, and manufactured nuclear power facilities since the 1950s, beginning with the world's first large central station nuclear power plant (Shippingport), which has produced power since 1957. Completed or contracted commercial nuclear capacity totals were in excess of 97,000 MW. Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Westinghouse manufacturing facilities include the largest commercial nuclear fuel fabrication facility in the world, and the world's most modern heat transfer equipment production facility, as well as other facilities producing nuclear steam supply system (NSSS)

components. Table 1.4-1 lists all Westinghouse pressurized water reactor (PWR) plants to date, including those plants under construction or on order, at the time of BVPS-2 license application.

The U.S. Nuclear Regulatory Commission (USNRC) and the Electric Power Research Institute have contracted with Westinghouse for research into NSSS related activities. Westinghouse experience was also utilized by the USNRC and Metropolitan Edison immediately following the Three Mile Island Unit 2 accident. At the time of license application, the corporation continued to participate with the Westinghouse Owner's Group of utilities in addressing the USNRC'S action plan for corrective actions.

1.4.3.1 Plants in Operation

Westinghouse PWR plants in operation were as follows:

1. Shippingport

   Shippingport was the world's first large central station nuclear power plant. The reactor plant was designed by the Bettis Atomic Power Laboratory, which is operated by Westinghouse under a USNRC contract. Shippingport's PWR has produced power for DLC since December 1957.

2. Yankee-Rowe

   Singled out by the USNRC as a "Nuclear Success Story" Yankee-Rowe went on-line in November 1960. Owned and operated by the Yankee Atomic Electric Company, Yankee-Rowe has progressed from an initial rating of 120 MWe to its present 175 MWe rating. Westinghouse supplied the NSSS and the turbine generator.

3. Trino Vercellese (Enrico Fermi)

   The Trino Vercellese nuclear plant was one of the first Westinghouse designed plants to incorporate chemical shim control of reactivity. Chemical shim has since become a standard feature of Westinghouse PWR control. Trino Vercellese achieved initial criticality in June 1964 and began power operation in October 1964. The plant is rated at 260 MWe.

4. Chooz (Ardennes)

   The Chooz plant is unique in that the Westinghouse PWR and its auxiliaries are housed in man-made caverns. Ardennes, a joint Franco-Belgian undertaking, owned and operated by the Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA), is located in France near the French-Belgian border. Chooz achieved initial criticality in October 1966 and began power operation in 1967.

5. San Onofre Unit 1

San Onofre Unit 1 employs the Westinghouse developed rod cluster control concept which has since become a standard feature on the Westinghouse PWR. Owned by the Southern California Edison Company and the San Diego Gas and Electric Company, the 430 MWe plant is located near San Clemente, California. Westinghouse supplied the NSSS and the turbine generator. Initial criticality was achieved in June 1967, and power operation began in January 1968.

6. Haddam Neck (Connecticut Yankee)

Owned and operated by the Connecticut Yankee Atomic Power Company, this plant went critical in July 1967 and attained full power operation in December 1967. Like San Onofre Unit 1, the plant employs rod cluster control in conjunction with chemical shim control. Westinghouse supplied the NSSS and the turbine generator. The plant has been uprated to 575 MWe.

7. Jose Cabrera - Zorita

The Jose Cabrera Station is located near Zorita, Spain. The 153 MWe plant employs rod cluster control, chemical shim control and a Zircaloy-clad core. Construction began in mid-1965 and power operation began in 1968. Jose Cabrera is owned and operated by the Union Electrica, S. A., a Spanish utility.

8. Beznau Unit 1 and Unit 2

Beznau Unit 1, Switzerland's first commercial nuclear power plant, achieved initial criticality in June 1969 and supplied power to the system in July 1969. The 350 MWe plant was designed and constructed by the Westinghouse-Brown Boveri Consortium for the owner/operator utility, Nordostschweizerische Kraftwerke AG. The plant started producing power less than 4 years after award of the plant contract. Beznau Unit 2 achieved criticality in October 1971 and began commercial operation in early 1972.

9. Robert Emmett Ginna

The Robert Emmett Ginna Plant, owned and operated by Rochester Gas and Electric Corporation, is located in New York on the south shore of Lake Ontario. Westinghouse supplied the 490 MWe plant on a turnkey basis. Construction began in April 1966 with initial criticality being achieved in November 1969 (just 42 months after start of construction). Power was supplied to the system in December 1969.

10. Mihama Unit 1 and Takahama Unit 1

    These plants are owned by the Kansai Electric Power Company, Inc. Mihama Unit 1 is a two-loop, 320 MWe unit and marks the beginning of a line of Westinghouse PWRS supplying the generation needs of the Far East. Westinghouse International Company was the prime contractor for the Mihama project, supplying the NSSS engineering, nuclear fuel, and some major system components. Mihama Unit 1 required only 44 months from the start of site construction to first power production in August 1970. Takahama Unit 1 is a three-loop, 780 MWe unit. Initial criticality was achieved in March 1974.

11. H. B. Robinson Unit 2

    This plant is a three-loop, 707 MWe unit which was built on a turnkey basis for the Carolina Power and Light Company. The plant is located at a site near Hartsville, South Carolina, on a man-made cooling lake. The construction permit was granted in April 1967. The plant achieved criticality in August 1970 and first power to system in October 1970.

12. Point Beach Unit 1 and Unit 2

    The Point Beach Project consists of two 497 MWe units, which were built on a turnkey basis for the Wisconsin Michigan Power Company and the Wisconsin Electric Power Company. The plants are located near Two Creeks, Wisconsin, 90 miles north of Milwaukee on Lake Michigan. This was the first two-unit station to utilize many common facilities and shared auxiliary systems. The construction permit for Point Beach Unit 1 was granted in July 1967 with initial criticality and first power to the system in November 1970. Point Beach Unit 2 went critical in May 1972 and was available for commercial operation in October 1972.

13. Surry Unit 1 and Unit 2

    The Surry Power Station, two three-loop 822 MWe units, is owned by the Virginia Electric and Power Company. The James River Station is about 30 miles from Norfolk, Virginia. First criticality on Surry Unit 1 was achieved in July 1972. Commercial operation began in September 1972. Initial criticality on Surry Unit 2 was achieved in March 1973.

14. Turkey Point Unit 3 and Unit 4

    Florida Power and Light Company is the owner of a four-unit station on Biscayne Bay, Florida. Turkey Point Units 3 and 4 of the station are three-loop, 745 MWe

plants. Commercial status for Turkey Point Unit 3 was achieved in December 1972. Initial criticality for Turkey Point Unit 4 was achieved in June 1973.

15. Indian Point Unit 2 and Unit 3

    Consolidated Edison Company of New York operates three nuclear units located in Buchanan, New York; two (Units 1 and 2) are owned by the Company and one (Unit 3) is owned by the Power Authority of the State of New York. Units 2 and 3 are Westinghouse PWRS rated at 873 MWe and 965 MWe, respectively. Indian Point Unit 2 achieved initial criticality in May 1973 and Indian Point Unit 3 achieved initial criticality in April 1976.

16. Prairie Island Unit 1

    Northern States Power Company, is the owner of these two-loop, 530 MWe units located in Welch, Minnesota. Initial criticality was achieved in December 1973 for Prairie Island Unit 1, and in December 1974 for Prairie Island Unit 2.

17. Zion Unit 1 and Unit 2

    Commonwealth Edison Company is the owner of these two four-loop, 1,050 MWe units. The units are located on Lake Michigan near Zion, Illinois. Initial criticality was achieved in June 1973 for Zion Unit 1 and in December 1973 for Zion Unit 2.

18. Kewaunee

    Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company are the owners of this two-loop, 541 MWe plant located in Kewaunee, Wisconsin. Initial criticality was achieved in March 1974.

19. Ringhals Unit 2

    Statens Vattenfallsverk (SSPB) is the owner of this three-loop, 822 MWe unit located in Sweden. Initial criticality was achieved in June 1974.

20. Donald C. Cook Unit 1 and Unit 2

    Indiana and Michigan Electric Company is the owner of these four-loop 1,090 MWe plants located in Bridgman, Michigan. These plants are the first to use the Westinghouse Ice Condenser Containment design. Initial criticality was achieved in January 1975 for Unit 1 and March 1978 for Unit 2.

21. Trojan

This four-loop, 1,130 MWe plant is jointly owned by Portland General Electric Company, Eugene Water and Electric Board, and Pacific Power and Light Company. In addition to being the first commercial nuclear plant to operate in the Pacific Northwest (located on the Oregon shore of the Columbia River near Rainier, Oregon), Trojan is the first 17 x 17 fuel-rod-per-assembly plant to achieve criticality. Initial criticality was achieved in December 1975.

22. Beaver Valley Unit 1

This three-loop, 852 MWe plant is jointly owned by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, Beaver Valley Unit 1 is located on the Ohio River, 22 miles northwest of Pittsburgh, Pemsylvania. Commercial operation began in early 1976.

23. Salem Unit 1 and Unit 2

Salem Units 1 and 2, owned jointly by the Public Service Electric and Gas Company, Philadelphia Electric Company, Atlantic Electric Company, and Delmarva Power and Light Company, are located on Artificial Island, a man-made peninsula in Salem County, New Jersey. The 1,090 MWe, four-loop plant achieved initial criticality for Unit 1 in late 1976 and Unit 2 achieved criticality in August 1980.

24. North Anna Unit 1 and Unit 2

Virginia Electric and Power Company owns the two approximately 907 MWe (net) plants located 40 miles north of Richmond, Virginia, on Lake Anna. Unit 1 achieved criticality in June 1980.

25. Joseph M. Farley Unit 1 and Unit 2

The two 899 MWe (net) Alabama Power Company units are located at Dothan, Alabama which is approximately 180 miles south-southwest of Atlanta, Georgia. Unit 1 achieved criticality in August 1977 and Unit 2 achieved criticality in February 1981.

26. Sequoyah Unit 1 and Unit 2

The two 1,148 MWe (net) units are located on the Tennessee River near Chattanooga, Tennessee. These units are owned by Tennessee Valley Authority. Sequoyah Unit 1 received a full power license in September 1980.

1.4.3.2 Westinghouse Facilities

Westinghouse, in its effort to plan for the future, developed a broad range of facilities to satisfy the needs of the nuclear industry. The following paragraphs briefly describe these facilities:

1. Columbia Plant, Nuclear Fuel Division

   The Columbia Plant was capable of performing all operations necessary to manufacture finished nuclear fuel assemblies. These operations include conversion of uranium hexafluoride to uranium dioxide powder, fabrication of fuel assembly grids, complete pellet loading, and final fabrication of assemblies. The plant, located at Columbia, South Carolina, began full production in early 1970. The Columbia Plant was the largest commercial nuclear fuel fabrication facility in the world.

2. Tampa Division

   The Tampa Division Plant was the world's most modern heat transfer equipment production facility. The plant has 236,000 square feet of working space with two manufacturing aisles for the production of steam generators and pressurizers. Transportation facilities include four railroad spurs and a complete barge slip and dock facility for water shipment to all parts of the world. The Tampa Division Plant made its first steam generator and pressurizer shipment in September 1969.

3. Pensacola Division

   The Pensacola Division Plant, located on Escambia Bay on the northwest coast of Florida, was a new 140,000 square foot manufacturing plant for producing precision reactor vessel internals. Contributing to the precision manufacturing capability was an environmental control system which minimized year round temperature changes throughout the shop area. Transportation facilities of the plant include a railroad spur for loading and unloading inside the shop, and access to barge loading facilities on Escambia Bay. Pensacola shipped its first package of reactor internals in July 1970.

4. Cheswick Plant, Electro-Mechanical Division

   The Electro-Mechanical Division was established in Cheswick, Pennsylvania in 1953 to manufacture canned motor primary coolant pumps for nuclear reactors. The product line expanded to include shaft seal pumps (reactor coolant pumps), valves from 4 inches to 31 inches, and control rod drive mechanisms, essential components of the Westinghouse PWR. The facility

occupied 250,000 square feet and contained the most modern facilities available for the production and testing of nuclear plant components.

5. Specialty Metals Division

   The Specialty Metals Division located in Blairsville, Pennsylvania, was completed in late 1967. Several essential PWR component processes were accomplished at Blairsville, including the precision manufacture of inconel tubing for steam generators, and the complete processing of Zircaloy seamless tubing for nuclear fuel cladding. At Blairsville, complete quality control facilities were utilized for the evaluation and analysis of all specialty metal products used in Westinghouse nuclear systems.

6. Westinghouse Nuclear Center

   The headquarters of Westinghouse Nuclear Energy Systems was located just east of Pittsburgh in Monroeville, Pennsylvania. Operating primarily as a headquarters and engineering facility, the complex housed many of the divisions which encompassed Westinghouse's nuclear activities associated with the electric utility industry.

7. Zion Nuclear Training Center

   The Westinghouse Electric Corporation and the Commonwealth Edison Company of Chicago built and operated a nuclear training center at Zion, Illinois. The 28,000 square foot training center contained classrooms, a training reactor, training material center, video recording facilities, and multi-plant nuclear power plant simulators. Westinghouse staffed and operated the center, supplied all the equipment required, and was responsible for the development and presentation of all training programs. Commonwealth Edison provided the building, access to the Zion nuclear units for conducting in-plant observation training, and advised and assisted Westinghouse in developing training programs.

1.4.4 Power Conversion System Supplier

Westinghouse was awarded the contracts to design, fabricate, deliver, and erect the turbine generator for BVPS-2. Westinghouse achieved an excellent operating record in the nuclear turbine generator industry by incorporating product improvements based on operating experience as well as knowledge gained from research and development programs and the capabilities of modern manufacturing facilities. This nuclear operating performance record made Westinghouse one of the leaders in total power produced as well as in reliability and availability.

Continuing Westinghouse commitment to this kind of product evaluation produced modern and reliable nuclear turbine-generator units around the world. The Westinghouse nuclear turbine-generator operating record was as follows:

a. The first Westinghouse nuclear turbine generator was placed in commercial operation in 1957.

b. Forty-five Westinghouse nuclear turbine generators were in service, totaling over 36,000 MW.

c. An additional forty-five Westinghouse nuclear turbine generators were on order, in storage, or being erected.

1.4.5 Consultants

Duquesne Light Company directly or through the architect engineer, SWEC, engaged the service of various consultants to perform work relating to the design, construction, operation, and maintenance of BVPS-2. These consultants included, but were not limited to, the following companies:

1. Hansen, Holley, and Biggs

   This firm provided consultation services on structural design and analyses for SWEC as well as others. The firm included Professors R. J. Hanson; M. J. Holley, Jr.; and J. M. Biggs - all of whom were actively associated with the Massachusetts Institute of Technology (MIT). The firm provided services to SWEC in support of efforts for Yankee Atomic, Connecticut Yankee, Malibu, and Shoreham Nuclear Power Plants.

2. NUS Corporation

   NUS Corporation was retained to provide general consulting services. This corporation was a consulting engineering firm headquartered in Rockville, Maryland, that served 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 numbered over eighty utilities among its clients and provided these clients with a broad spectrum of nuclear-related services including reactor safeguards analysis, reactor siting, and reactor design services.

3. Weston Geophysical Research, Inc.

   Weston Geophysical Research, Inc., was retained by the BVPS-2 project to provide consulting services on seismicity. The Reverend Daniel Linehan, Director of Weston Observatory, was a consultant to Weston Geophysical Research, Inc., and participated in these studies. Father Linehan was 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 were pioneers in the development of shallow refraction survey techniques, especially shear wave velocity determinations, which were necessary for establishing dynamic soil moduli for use in analysis of structural response to earthquakes.

4. Whitman and Rand

   This firm provided consulting services in the area of soil dynamics, geology, and hydrology. Doctor R. V. Whitman, who was associated with MIT, was an outstanding authority in the field of soil dynamics and published many papers on the subject. His studies have included significant work on amplification of earthquake motion within the overburden. Mr. J. R. Rand, formerly Chief State Geologist for the State of Maine, assisted in geology and ground-water hydrology studies.

Tables for Section 1.4

(Tables in Section 1.4 are historical)

TABLE 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (HISTORICAL)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWe Net	Number of Loops
Shippingport	Duquesne Light Company; Energy Research & Development Administration (ERDA)	Pennsylvania	1957	90	4
Yankee-Rowe	Yankee Atomic Electric Company	Massachusetts	1961	175	4
Trino Vercellese (Enrico Fermi)	Ente Nazionale per l'Energia Elettrica (ENEL)	Italy	1965	260	4
Chooz Ardennes	Societe D'Energie Nucleaire Franco-Belge des Ardennes (SENA)	France	1967	305	4
San Onofre No. 1	Southern California Edison Co.; San Diego Gas and Electric Co.	California	1968	430	3
Haddam Neck (Connecticut Yankee)	Connecticut Yankee Atomic Power Company	Connecticut	1968	575	4
Jose Cabrera - Zorita	Union Electrica, S. A.	Spain	1969	153	1
Beznau No. 1	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1969	350	2
Robert Emmett Ginna	Rochester Gas and Electric Corporation	New York	1970	490	2
Mihama No. 1	The Kansai Electric Power Company, Inc.	Japan	1970	320	2
Point Beach No. 1	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1970	497	2
H. B. Robinson No. 2	Carolina Power and Light Co.	South Carolina	1971	712	3
Beznau No. 2	Nordostschweizerische Kraftwerke AG (NOK)	Switzerland	1972	350	2

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Point Beach No. 2	Wisconsin Electric Power Co.; Wisconsin Michigan Power Co.	Wisconsin	1972	497	2
Surry No. 1	Virginia Electric and Power Co.	Virginia	1972	822	3
Turkey Point No. 3	Florida Power and Light Co.	Florida	1972	693	3
Indian Point No. 2	Consolidated Edison Company of New York, Inc.	New York	1973	873	4
Prairie Island No. 1	Northern States Power Company	Minnesota	1973	530	2
Turkey Point No. 4	Florida Power and Light Co.	Florida	1973	693	3
Surry No. 2	Virginia Electric and Power Co.	Virginia	1973	822	3
Zion No. 1	Commonwealth Edison Company	Illinois	1973	1040	4
Kewaunee	Wisconsin Public Service Corp.; Wisconsin Power and Light Co.; Madison Gas and Electric Co.	Wisconsin	1974	535	2
Prairie Island No. 2	Northern States Power Company	Minnesota	1974	530	2
Takahama No. 1	The Kansai Electric Power Company, Inc.	Japan	1974	781	3
Zion No. 2	Commonwealth Edison Company	Illinois	1974	1040	4
Doel No. 1	Indivision Doel	Belgium	1975	390	2
Doel No. 2	Indivision Doel	Belgium	1975	390	2
Donald C. Cook No. 1	Indiana and Michigan Electric Company (AEP)	Michigan	1975	1054	4
Ringhals No. 2	Statens Vattenfallsverk (SSPB)	Sweden	1975	822	3

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Indian Point No. 3	Power Authority of the State of New York (PASNY)	New York	1976	873	4
Salem No. 1	Public Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic Electric Co.; Delmarva Power and Light Co.	New Jersey	1976	1090	4
Trogan	Portland General Electric Co.; Eugene Water and Electric Board; Pacific Power and Light Company	Oregon	1976	1130	4
Beaver Valley No. 1	Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Company	Pennsylvania	1977	852	3
Almaraz No. 1	Union Electrica, S.A.; Compania Sevillana de Electricicdad, S. A.; Hidroelectrica Espanola, S. A.	Spain	1978	902	3
Angra dos Reis No. 1	Furnas-Centrais Electricas, S. A.	Brazil	1978	626	2
Diablo Canyon No. 1	Pacific Gas and Electric Co.	California	1978	1084	4
Donald C. Cook No. 2	Indiana and Michigan Electric Company (AEP)	Michigan	1978	1060	4
Joseph M. Farley No. 1	Alabama Power Company	Alabama	1978	829	3
Ko-Ri No. 1	Korea Electric Company	Korea	1978	564	2
North Anna No. 1	Virginia Electric and Power Co.	Virginia	1978	907	3
Ringhals No. 3	Statens Vattenfallsvert (SSPB)	Sweden	1978	912	3
Ohi No. 1	The Kansai Electric Power Co., Inc.	Japan	1978	1122	4

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Ohi No. 2	The Kansai Electric Power Co., Inc.	Japan	1978	1122	4
Sequoyah No. 1	Tennessee Valley Authority	Tennessee	1978	1148	4
Almaraz No. 2	Union Electrica, S.A.; Compania Sevillana de Electricidad, S.A.; Hidroelectrica Espanola, S.A.	Spain	1979	902	3
Asco No. 1	Fuerzas Electricas deCataluna, S.A. (FECSA)	Spain	1979	902	3
Asco No. 2	Fuerzas Electricas deCataluna, S.A. (FECSA); Empresa Nacional Hidroelec-trica del Riborganzana, S.S. (ENHER); Fuerzas Hidroelectricas del Segre, S.A.; Hidroelectroca deCataluna, S.A.	Spain	1979	902	3
Diablo Canyon No. 2	Pacific Gas and Electric Co.	California	1979	1106	4
Lemoniz No. 1	Iberduero, S.A.	Spain	1979	902	3
Sequoyah No. 2	Tennessee Valley Authority	Tennessee	1979	1148	4
Watts Bar No. 1	Tennessee Valley Authority	Tennessee	1979	1177	4
William B. McGuire No. 1	Duke Power Company	North Carolina	1979	1180	4
Joseph M. Farley No. 2	Alabama Power Company	Alabama	1979	829	3
Krsko	Savske Elektrarne, Ljubljana, Slovenia; Elektroprivreda, Zagreb, Croatia	Yugoslavia	1979	615	2
Ringhals No. 4	Statens Vattenfallsvert (SSPB)	Sweden	1979	912	3

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Salem No. 2	Public Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic Electric Co.; Delmarva Power and Light Co.	New Jersey	1979	1115	4
Lemonis No. 2	Iberduero, S.A.	Spain	1980	902	3
North Anna No. 2	Virginia Electric and Power Co.	Virginia	1979	907	3
Virgil C. Summer	South Carolina Electric and Gas Company	South Carolina	1980	900	3
Watts Bar No. 2	Tennessee Valley Authority	Tennessee	1980	1177	4
South Texas Project Unit No. 1	Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas	Texas	1980	1250	4
William B. McGuire No. 2	Duke Power Company	North Carolina	1981	1180	4
Comanche Peak No. 1	Texas Utilities Generating Co.	Texas	1981	1150	4
Byron No. 1	Commonwealth Edison Company	Illinois	1981	1120	4
Vandellos No. 2	Fuerzas Electricas de Cataluna, S.A. (FECSA); Empresa Nacional Hidroelectrica del Ribagorzana, S.A. (ENHER); Fuerzas Hidroelectricas del Segre S.A.; Hidroelectrica de Cataluna, S.A.	Spain	1981	920	3
Seabrook No. 1	Public Service Company of New Hampshire; United Illuminating Company	New Hampshire	1982	1200	4
Braidwood No. 1	Commonwealth Edison Company	Illinois	1982	1120	4

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Catawba No. 1	Duke Power Company	South Carolina	1982	1145	4
Callaway No. 1	SNUPPS - Union Electric Co.	Missouri	1982	1150	4
Ko-Ri No. 2	Korea Electric Company	Korea	1982	605	2
Comanche Peak No. 2	Texas Utilities Generating Co.	Texas	1982	1150	4
Marble Hill No. 1	Public Service Company of Indiana, Inc.,; Wabash Valley Power Association	Indiana	1982	1130	4
Millstone No. 3	Northeast Nuclear Energy Co.	Connecticut	1982	1156	4
South Texas Project Unit 2	Houston Lighting and Power Co.; Central Power and Light Co. City Public Service of San Antonio; City of Austin, Texas	Texas	1982	1250	4
Napot Point No. 1	National Power Corporation	Philippines	1982	620	2
Sayago No. 1	Iberduero, S.A.	Spain	1982	1000	3
Braidwood No. 2	Commonwealth Edison Company	Illinois	1983	1120	4
Byron No. 2	Commonwealth Edison Company	Illinois	1983	1120	4
Catawba No. 2	Duke Power Company	South Carolina	1983	1145	4
Maanshan No. 1	Taiwan Power Company	Taiwan	1983	907	3
Alvin W. Vogtle No. 1	Georgia Power Company; Oglethorpe Electric Membership Corp.; Municipal Electric Authority of Georgia; City of Dalton, Georgia	Georgia	1983	1113	4

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Wolf Creek Unit No. 1	SNUPPS - Kansas Gas and Electric Company; Kansas City Power and Light Company	Kansas	1984	1150	4
Seabrook No. 2	Public Service Company of New Hampshire; United Illuminating Company	New Hampshire	1984	1200	4
Jamesport No. 1	Long Island Lighting Company; New York State Electric and Gas Corp.	New York	1984	1150	4
Maanshan No. 2	Taiwan Power Company	Taiwan	1984	907	3
Alvin W. Vogtle No. 2	Georgia Power Company; Oglethorpe Electric Membership Corp.; Municipal Electric Authority of Georgia; City of Dalton, Georgia	Georgia	1984	1113	4
Marble Hill No. 2	Public Service Company of Indiana, Inc.; Wabash Valley Power Association	Indiana	1984	1130	4
Shearon Harris No. 1	Carolina Power and Light Co.	North Carolina	1984	900	3
Sterling	SNUPPS - Rochester Gas and Electric Corporation; Central Hudson Gas and Electric Corporation; Niagara Mohawk Power Corporation; Orange and Rockland Utilities, Inc.	New York	1984	1150	4
Tyrone No. 1	SNUPPS - Northern States Power Company	Wisconsin	1984	1150	4

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Atlantic No. 1 (O.P.S)	Public Service Electric and Gas Company; Atlantic Electric Co.; Jersey Central Power and Light Company	New Jersey	-	1150	4
Jamesport No. 2	Long Island Lighting Company; New York State Electric and Gas Corp.	New York	1986	1150	4
Shearon Harris No. 2	Carolina Power and Light Co.	North Carolina	1986	900	3
Callaway No. 2	SNUPPS - Union Electric Company	Missouri	1987	1150	4
Haven No. 1	Wisconsin Electric Power Co.; Wisconsin Power and Light Co.; Wisconsin Public Service Corp.	Wisconsin	1987	900	3
Atlantic No. 2 (O.P.S)	Public Service Electric and Gas Company; Atlantic Electric Co.; Jersey Central Power and Light Company	New Jersey	-	1150	4
Shearon Harris No. 4	Carolina Power and Light Co.	North Carolina	1988	900	3
Haven	Wisconsin Electric Power Co.; Wisconsin Power and Light Co.; Wisconsin Public Service Corp.	Wisconsin	1989	900	3
Unit No. 4	Iberduero, S.A.	Spain	1980's	1000	3
Shearon Harris No. 3	Carolina Power and Light Co.	North Carolina	1990	900	3

TABLE 1.4-1 (Cont)

Plant	Owner Utility	Location	Scheduled Commercial Operation	MWE Net	Number of Loops
Unassigned No. 1 (O.P.S)	Public Service Electric and Gas Company; Atlantic Electric Company	New Jersey	1990	1150	4
Unassigned No. 2	Public Service Electric and Gas Company; Atlantic Electric Company	New Jersey	1992	1150	4

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The requirements for further technical information provided herein reflects the status of Beaver Valley Power Station - Unit 2 (BVPS-2) at the time of the issuance of the Operating License. This section 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.

The Westinghouse Electric Corporation (Westinghouse) report WCAP-8768 (Eggleston 1978) presents descriptions of the safety-related research and development programs which are being carried out for, by, or in conjunction with Westinghouse Nuclear Energy Systems and which are applicable to Westinghouse pressurized water reactors (PWRS).

For each program still in progress, the safety-related program is first introduced and is followed where appropriate by background information. A description of the program follows, 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) that have not yet reached a stage at which it is reasonably certain that the results confirm the expectation. The backup position is one that might be used if the results were to be unfavorable; it is not necessarily the only course that might be taken in the ultimate solution of the problem.

The term "research and development" as used in this report is the same as that used by the U.S. Nuclear Regulatory Commission (USNRC) in 10 CFR 50.2, and means:

1. Theoretical analysis, exploration, or experimentation, or

2. The extension of investigative findings and theories of a scientific or technical nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, material, and processes.

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 could lead to design improvements.

Included in the overall research and development effort are the programs described as follows which are applicable to this plant, but are not required for issuance of either a Construction Permit or Operating License.

1.5.1 Blowdown Heat Transfer Testing

1.5.1.1 Introduction

The USNRC acceptance criteria for emergency core cooling systems (ECCSs) for light-water power reactors were issued in Section 50.46 of 10 CFR 50 on December 28, 1973. They define the basis

and conservative assumptions to be used in the evaluation of the performance of ECCSs. Westinghouse believes that some of the conservatism of the criteria is associated with the manner in which transient departure from nucleate boiling (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 indicated that the time to DNB can be delayed under transient conditions. To demonstrate the conservatism of the ECCS evaluation models, Westinghouse has initiated a program to experimentally simulate the blowdown phase of a loss-of-coolant accident (LOCA). This testing is part of the Electric Power Research Institute (EPRI) sponsored Blowdown Heat Transfer Program, which was started in early 1976. Testing was completed in 1979. A DNB correlation developed by Westinghouse from these test results will be used in the ECCS analyses.

1.5.1.2 Objective

The objective of the Blowdown Heat Transfer Test was to determine the time that DNB occurs under LOCA conditions. This information was used to confirm the existing, or develop a new Westinghouse transient DNB correlation. The steady state DNB data obtained from 15 by 15 and 17 by 17 test programs can be used to assure that the geometrical differences between the two fuel arrays can be correctly treated in the transient correlations.

1.5.1.3 Program

The program was divided into two phases. Phase I tests started from steady state conditions, with sufficient power to maintain nucleate boiling throughout the bundle, controlled ramps of decreasing test section pressure, or flow initiated DNB. By applying a series of controlled conditions, investigation of the DNB was studied over a range of qualities, flows, and pressures relevant to a PWR blowdown.

Phase I provided separate-effects data to permit heat transfer correlation development.

Typical parameters used for Phase I testing are presented in Table 1.5-1.

Phase II simulated PWR behavior during a LOCA to permit definition of the time delay associated with onset of DNB. Tests in Phase II covered the large double-ended guillotine cold leg break. All tests in Phase II were started after establishment of typical steady state operating conditions. The fluid transient was then initiated, and the rod power decay was programmed in such a manner as to simulate the actual heat input of fuel rods. The test was terminated when the heater rod temperatures reached a predetermined limit.

Typical parameters used for Phase II testing are shown in Table 1.5-2.

1.5.1.4 Test Description

The experimental program was conducted in the J-Loop at the Westinghouse Forest Hills Facility with a full length 5 by 5 rod bundle simulating a section of a 15 by 15 assembly to determine DNB occurrence under LOCA conditions.

The heater rod bundles used in this program were internally heated rods, capable of a maximum power of 18.8 kW/ft, with a total power of 135 kW (for extended periods) over the 12-foot heated length of the rod. Heat was generated internally by means of a varying cross-sectional resistor which approximates a chopped cosine power distribution. Each rod was adequately instrumented with a total of 12 clad thermocouples.

1.5.1.5 Results

The experiments in the delayed departure from nucleate boiling facility resulted in cladding temperature and inlet fluid properties measured as a function of time throughout the blowdown range from 0 to 20 seconds.

Facility modifications and installation of the initial test bundle were completed. A series of shakedown tests in the J-Loop were performed. These tests provided data for instrumentation calibration and check-out, and provided information regarding facility control and performance. Initial program tests were performed during the first half of 1975. Under the sponsorship of EPRI, testing was reinitiated during 1976 on the same test bundle. The testing was terminated in November 1976 and plans were made for a new test bundle and further testing during 1978-1979. A DNB correlation developed from these results are used in the Westinghouse ECCS analyses for Beaver Valley Power Station - Unit 2.

1.5.2 Reference for Section 1.5

Eggleston, F.T. 1978. Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977, Summer 1978. WCAP-8768, Revision 2.

Tables for Section 1.5

(Tables in Section 1.5 are historical)

TABLE 1.5-1

DELAYED DEPARTURE FROM NUCLEATE BOILING
PHASE I TEST PARAMETERS (HISTORICAL)

Parameters	Nominal Value
Initial Steady State Conditions	
Pressure (psia)	1,250 to 2,250
Test section mass velocity (lb/hr-ft^2)	1.12 to 2.5 x 10^6
Core inlet temperature (°F)	550 to 600
Maximum heat flux (Btu/hr-ft^2)	306,000 to 531,000
Transient Ramp Conditions	
Pressure decrease	0 to 350 psi/sec and subcooled depressurization from 2,250 psia
Flow decrease (%/sec)	0 to 100
Inlet enthalpy	constant

TABLE 1.5-2

DELAYED DEPARTURE FROM NUCLEATE BOILING
PHASE II TEST PARAMETERS (HISTORICAL)

Parameters	Nominal Value
Initial Steady State Conditions	
Pressure (psia)	2,250
Test section mass velocity (lb/hr-ft^2)	2.5 X 10^6
Inlet coolant temperature (°F)	545
Maximum heat flux (Btu/hr-ft^2)	531,000
Transient Conditions	
Simulated break	Double-ended cold leg guillotine breaks

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 lists topical reports which, prior to issuance of the operating license, provided information in addition to that provided in this Final Safety Analysis Report and which were filed separately with the U.S. Nuclear Regulatory Commission (USNRC) in support of this and similar applications. Therefore, this section 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.

The legend for the review status code letter follows:

- A - U.S. Nuclear Regulatory Commission review complete; USNRC acceptance letter issued.
- AE - U. S. Nuclear Regulatory Commission accepted as part of the Westinghouse emergency core cooling system (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.
- B - Submitted to USNRC as background information; not undergoing formal USNRC review.
- O - On file with USNRC; older generation report with current validity; not actively under formal USNRC review.
- N - Not applicable; that is, open literature, etc.
- U - Actively under formal USNRC review.

Tables for Section 1.6

(Tables in Section 1.6 are historical)

TABLE 1.6-1

MATERIAL INCORPORATED BY REFERENCE (HISTORICAL)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-2048	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements	July 1962	4.3	0
WCAP-2850-L (Proprietary)	Single-Phase Local Boiling & Bulk Boiling Pressure Drop Correlations	April 1966	4.4	0
WCAP-2923	In-Pile Measurement of UOd Thermal Conductivity	1966	4.4	0
WCAP-3269-8	Hydraulic Tests of the San Onofre Reactor Model	June 1964	4.4	0
WCAP-3269-26	LEOPARD: A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094	Sept. 1963	4.3	0
WCAP-3385-56	Saxton Core II Fuel Performance Evaluation, Part II, and Evaluation of Mass Spectrometric and Radiochemical Materials Analyses of Irradiated Saxton Plutonium Fuel.	July 1970	4.3 4.4	0
WCAP-3680-20	Xenon-Induced Spatial Instabilities in Large PWRs	March 1968	4.3	0
WCAP-3680-21	Control Procedures for Xenon-Induced X-Y Instabilities in Large PWRs	Feb. 1969	4.3	0
WCAP-3680-22	Xenon-Induced Spatial Instabilities in Three Dimensions	Sept. 1969	4.3	0
WCAP-3696-8	Pressurized Water Reactor pH - Reactivity Effect Final Report	Oct. 1968	4.3	0
WCAP-3726-1	PUOd-UOd Fueled Critical Experiments	July 1967	4.3	0
WCAP-6065	Melting Point of Irradiated UOd	Feb. 1965	4.2 4.4	0
WCAP-6069	Burnup Physics of Heterogeneous Reactor Lattices	June 1965	4.4	0

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-6073	LASAR: A Depletion Program for Lattice Calculations Based on MUFT and THERMOS	April 1966	4.3	0
WCAP-6086	Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel including Isotopes of Elements Thorium Through Curium	Aug. 1969	4.3	0
WCAP-7015 Rev. 1	Subchannel Thermal Analysis of Rod Bundle Cores	Jan. 1969	4.4	0
WCAP-7048-P-A (Proprietary)	The PANDA Code	Jan. 1975	4.3	A
WCAP-7213-P-A (Proprietary)	The TURTLE 24.0 Diffusion Depletion Code	Feb. 1975	4.3	A
WCAP-7263 (Proprietary)	A Comprehensive Space-Time Dependent Analysis of Loss of Coolant (SATAN IV Digital Code)	Aug. 1971	3.6	0
WCAP-7306	Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors	April 1969	15.4.10	0
WCAP-7308-L (Proprietary)	Evaluation of Nuclear Hot Channel Factor Uncertainties	Dec. 1971	4.3	U
WCAP-7359-L (Proprietary)	Applications of the THINC Program to PWR Design	Aug. 1969	4.4	0
WCAP-7397-L (Proprietary)	Seismic Testing of Electrical and Control Equipment	Jan. 1970	3.10N	0
WCAP-7397-L, Suppl. 1	Seismic Testing of Electrical and Control Equipment	Jan. 1971	3.10N	0
WCAP-7488-L (Proprietary)	Solid State Logic Protection System Description	March 1971	7.1 7.2	A
WCAP-7558	Seismic Vibration Testing With Sine Beats	Oct. 1971	3.10N.5	U
WCAP-7588 Rev. 1-A	An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods	Jan. 1975	15.4	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-7635	MARVEL: A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System	1971	6.2.1.4	O
WCAP-7667-A (Proprietary)	Interchannel Thermal Mixing With Mixing Vane Grids	Jan. 1975	4.4	A
WCAP-7672	Solid State Logic Protection System Description	March 1971	7.1 7.2 7.3	A
WCAP-7695-P-A (Proprietary)	DNB Test Results for New Mixing Vane Grids	Jan. 1975	4.4	A
WCAP-7706	An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients	Feb. 1971	7.1 7.2	U
WCAP-7706-L (Proprietary)	An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients	July 1971	4.6	U
WCAP-7735	Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems	Aug. 1971	5.2	A
WCAP-7750	SATAN IV Digital Code: A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant	Aug. 1971	3.6	O
WCAP-7757-A	The PANDA Code	Jan. 1975	4.3	A
WCAP-7758-A	The TURTLE 24.0 Diffusion Depletion Code	Feb. 1975	4.3	A
WCAP-7765-AR	Westinghouse PWR Internals Vibration Summary Three-Loop Internals Assurance	Nov. 1973	3.9	A
WCAP-7769 Rev. 1	Overpressure Protection for Westinghouse Pressurized Water Reactors	April 1975	15.2	U
WCAP-7775-A	Interchannel Thermal Mixing with Mixing Vane Grids	Jan. 1975	4.4	A
WCAP-7800 Rev. 4-A	Nuclear Fuel Division Quality Assurance Program Plan	March 1975	4.2	A
WCAP-7803	Behavior of Austenitic Stainless Steel in Post-Hypothetical Loss-of-Coolant Environment	Dec. 1971	6.1.1	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-7806	Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods	Dec. 1971	4.3	B
WCAP-7810	Evaluation of Nuclear Hot Channel Uncertainties	Dec. 1971	4.3	U
WCAP-7811	Power Distribution Control of Westinghouse Pressurized Water Reactors	Dec. 1971	4.3	O
WCAP-7817	Seismic Testing of Electrical and Control Equipment	Dec. 1971	3.10N.5	O
WCAP-7817 Suppl. 1	Seismic Testing of Electrical and Control Equipment	Dec. 1971	3.10N.5	O
WCAP-7817 Suppl. 2	Seismic Testing of Electric and Control Equipment (Low Seismic Plants)	Dec. 1971	3.10N.5	U
WCAP-7817 Suppl. 3	Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)	Dec. 1971	3.10N.5	U
WCAP-7817 Suppl. 4	Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)	Nov. 1972	3.10N.5	U
WCAP-7817 Suppl. 5	Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel) (Low Seismic Plants)	March 1974	3.10N.5	U
WCAP-7817 Suppl. 6	Seismic Testing of Electrical and Control Equipment (Type DB Reactor Trip Switchgear)	Aug. 1974	3.10N.5	U
WCAP-7825	Evaluation of Protective Coatings for Use in Reactor Containments	Dec. 1971	6.1.2	A
WCAP-7836	Inlet Orificing of Open PWR Cores	Jan. 1972	4.4	-
WCAP-7838	Applications of the THINC Program to PWR Design	Jan. 1972	4.4	O

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-7907	LOFTRAN: Code Description	June 1972	5.2, 15.0, 15.1, 15.2 15.3, 15.4, 15.5, 15.6	U
WCAP-7908	FACTRAN: A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod	June, 1972	15.0, 15.3 15.4	U
WCAP-7912-A	Power Peaking Factors	Jan. 1975	4.3 4.4	A
WCAP-7912-P-A (Proprietary)	Power Peaking Factors	Jan. 1975	4.3 4.4	A
WCAP-7913	Process Instrumentation for Westinghouse Nuclear Steam Supply Systems (Four Loop Plant Using WCID 7300 Series Process Instrumentations)	Jan. 1973	7.2 7.3	B
WCAP-7916	Single-Phase Local Boiling and Bulk Boiling Pressure Drop Correlations	June 1972	4.4	O
WCAP-7921	Damping Values of Nuclear Power Plant Components	May 1974	3.7N.5	A
WCAP-7924-A	Basis for Heatup and Cooldown Limit Curves	April 1975	5.3	
WCAP-7941-P-A (Proprietary)	Effect of Axial Spacing on Interchannel Thermal Mixing with the R-Mixing Vane Grid	Jan. 1975	4.4	A
WCAP-7956	THINC IV: An Improved Program for Thermal and Hydraulic Analysis of Rod Bundle Cores	Oct. 1973	4.4	A
WCAP-7958-A	DNB Test Results for New Mixing Vane Grids	Jan. 1975	4.4	A
WCAP-7959-A	Effect of Axial Spacing on Interchannel Thermal Mixing with R-Mixing Vane Grid	Jan. 1975	4.4	A
WCAP-7964	Axial Xenon Transients Tests at the Rochester Gas and Electric Reactor	June 1971	4.3	O
WCAP-7979-P-A (Proprietary)	TWINKLE: A Multi-Dimensional Neutron Kinetics Computer Code	Jan. 1975	15.0 15.4	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-7980	WIT-6: Reactor Transient Analysis Computer Program Description	Nov. 1973	15.4	A
WCAP-7988 (Proprietary)	Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB	Oct. 1972	4.4	A
WCAP-8028-A	TWINKLE: A Multi-Dimensional Neutron Kinetics Computer Code	Jan. 1975	15.0 15.4	A
WCAP-8030-A	Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB	Oct. 1972	4.4	A
WCAP-8054 (Proprietary)	Application of the THINC IV Program to PWR Design	Sept. 1973	4.4	O
WCAP-8082 (Proprietary)	Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop	June 1973	6.2	-
WCAP-8099	A Summary Analysis of the April 30 Incident at the San Onofre Nuclear Generating Station Unit 1	April 1973	5.3	B
WCAP-8163	Reactor Coolant Pump Integrity in LOCA	Sept. 1973	5.4	U
WCAP-8170 (Proprietary)	Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)	June 1974	6.2 15.6	AE
WCAP-8171	Calculational Model for Core Reflooding After A Loss-of-Coolant Accident (WREFLOOD Code)	June 1974	6.2 15.6	AE
WCAP-8172	Pipe Breaks for the LOCA Analysis of Westinghouse Primary Coolant Loop	July 1973	6.2	-
WCAP-8174 (Proprietary)	Effect of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles	Aug. 1973	4.4	A
WCAP-8183 Rev. 10	Operational Experience with Westinghouse Cores (up to Dec. 31, 1977)	May 1981	4.2	-
WCAP-8195	Application of the THINC IV Program to PWR Design	Oct. 1973	4.4	O

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8200 Rev. 2 (Proprietary)	WFLASH: A Fortran IV Computer Program for Simulation of Transients in a Multi-Loop PWR	July 1974	15.6	AE
WCAP-8202	Effect of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles	Aug. 1973	4.2	A
WCAP-8218-P-A (Proprietary)	Fuel Densification Experimental Results and Model for Reactor Application	March 1975	4.1 4.2 4.3 4.4	A
WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application	March 1975	4.1 4.2 4.3 4.4	A
WCAP-8236 (Proprietary)	Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident	Dec. 1973	4.2	U
WCAP-8236 Addendum 1 (Proprietary)	Safety Analysis of the 8 Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident	March 1974	3.7N	U
WCAP-8252	Documentation of Selected Westinghouse Structural Analysis Computer Codes	April 1974	3.6	U
WCAP-8252 Rev. 1	Documentation of Selected Westinghouse Structural Analysis Computer Codes	May 1977	3.9	A
WCAP-8253	Source Term Data for Westinghouse Pressurized Water Reactors	July 1975	11.1	B
WCAP-8255	Nuclear Instrumentation System	Jan. 1974	7.2 7.7	B
WCAP-8261 Rev. 1	WFLASH: A Fortran IV Computer Program for Simulation of Transients in a Multi-Loop PWR	July 1974	15.6	AE
WCAP-8264-P-A (Proprietary)	Westinghouse Mass and Energy Release Data for Containment Design	June 1975	6.2	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8278 (Proprietary)	Hydraulic Flow Test of the 17 x 17 Fuel Assembly	Feb. 1974	4.2 4.4	U
WCAP-8279	Hydraulic Flow Test of the 17 x 17 Fuel Assembly	Feb. 1974	4.2 4.4	U
WCAP-8288	Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident	Dec. 1973	4.2 4.4	U
WCAP-8288 Addendum 1	Safety Analysis of the 8 Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident	April 1974	3.7N	U
WCAP-8296-P-A (Proprietary)	Effect of 17 x 17 Fuel Assembly Geometry on DNB	Feb. 1975	4.4	U
WCAP-8297-A	Effect of 17 x 17 Fuel Assembly Geometry on DNB	Feb. 1975	4.4	A
WCAP-8298-P-A (Proprietary)	The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing	Jan. 1975	4.4	A
WCAP-8299-A	The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing	Jan. 1975	4.4	A
WCAP-8301 (Proprietary)	LOCTA IV Program: Loss-of-Coolant Transient Analysis	June 1974	15.6	AE
WCAP-8302 (Proprietary)	SATAN VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant	June 1974	6.2.1.3 15.0 15.6	AE
WCAP-8303-P-A (Proprietary)	Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests	July 1975	3.9N	A
WCAP-8305	LOCTA IV Program: Loss-of-Coolant Transient Analysis	June 1974	15.0 15.6	AE
WCAP-8306	SATAN VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant	June 1974	6.2.1.3 15.0 15.6	AE

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8312-A Rev. 2	Westinghouse Mass and Energy Release Data For Containment Design	Aug. 1975	6.2.1.3 15.6	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8317-A	Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests	July 1975	3.9N	A
WCAP-8324-A	Control of Delta Ferrite in Austentitic Stainless Steel Weldments	June 1974	3.8 5.2 6.1	A
WCAP-8326	COCO: Containment Pressure Analysis Code	June 1974	15.6	AE
WCAP-8327 (Proprietary)	COCO: Containment Pressure Analysis Code	June 1974	15.6	AE
WCAP-8330	Westinghouse Anticipated Transients Without Trip Analysis	Aug. 1974	4.3, 4.6 15.1, 15.2 15.4, 15.8	U
WCAP-8339	Westinghouse ECCS Evaluation Mode-Summary	July 1974	4.3 15.6	AE
WCAP-8340 (Proprietary)	Westinghouse ECCS Plant Sensitivity Studies	July 1974	15.6	AE
WCAP-8341 (Proprietary)	Westinghouse ECCS Evaluation Model Sensitivity Studies	July 1974	15.6	AE
WCAP-8342	Westinghouse ECCS Evaluation Model Sensitivity Studies	July 1974	15.6	AE
WCAP-8356	Westinghouse ECCS Plant Sensitivity Studies	July 1974	15.6	AE
WCAP-8359	Effects of Fuel Densification Power Spikes on Clad Thermal Transients	July 1974	4.3	A
WCAP-8373	Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974	Aug. 1974	3.10N	U
WCAP-8377 (Proprietary)	Revised Clad-Flattening Model	July 1974	4.2	A
WCAP-8381	Revised Clad-Flattening Model	July 1974	4.2	A
WCAP-8385 (Proprietary)	Power Distribution Control and Load-Following Procedures	Sept. 1974	4.3 4.4	U

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8403	Power Distribution Control and Load-Following Procedures	Sept. 1974	4.3 4.4	U
WCAP-8424 Rev. 1	An Evaluation of Loss-of-Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs	June 1975	15.3	U
WCAP-8446 (Proprietary)	17 x 17 Drive Line Component Tests: Phase IE, II, III D Loop Drop and Deflection	Dec. 1974	15.0	A
WCAP-8448	17 x 17 Drive Line Component Tests: Phase IE, II, III D Loop Drop and Deflection	Dec. 1974	15.0	A
WCAP-8454	Analysis of Data from the Zion (Unit 1) THINC Verification Test	May 1976	4.4	A
WCAP-8471 (Proprietary)	Westinghouse ECCS Evaluation Model Supplementary Information	April 1975	15.6	AE
WCAP-8472	Westinghouse ECCS Evaluation Model Supplementary Information	April 1975	15.6	AE
WCAP-8498	Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors	July 1975	4.3	U
WCAP-8516-P (Proprietary)	UHI Plant Internals Vibration Measurement Program and Pre- and Post-Hot Functional Examinations	April 1975	3.9N	A
WCAP-8517	UHI Plant Internals Vibration Measurement Program and Pre- and Post-Hot Functional Examinations	April 1975	3.9N	A
WCAP-8536 (Proprietary)	Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22-Inch Grid Spacing	May 1975	4.4	A
WCAP-8537	Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22-Inch Grid Spacing	May 1975	4.4	A
WCAP-8565-P-A (Proprietary)	Westinghouse ECCS Four Loop Plant (17 x 17) Sensitivity Studies	July 1975	15.6	A
WCAP-8566-A	Westinghouse ECCS Four Loop Plant (17 x 17) Sensitivity Studies	July 1975	15.6	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8567	Improved Thermal Design Procedure	Sept. 1975	4.4	A
WCAP-8575 Suppl. 1 (Proprietary)	Augmented Start-up and Cycle 1 Physics Program	June 1976	4.3	U
WCAP-8576 Suppl. 1	Augmented Start-up and Cycle 1 Physics Program	June 1976	4.3	U
WCAP-8584 (Proprietary)	Failure Modes and Effects Analysis of the Engineering Safeguard Features Actuation System	April 1976	4.6 7.3	U
WCAP-8586-P-A (Proprietary)	Westinghouse ECCS Four Loop Plant (17 x 17) Sensitivity Studies	July 1975	15.6	A
WCAP-8587 Suppl. 1	Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment	Nov. 1978	3.10N 3.11N	
WCAP-8587 Rev. 2	Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment	Feb. 1979	3.10N 3.11N	U
WCAP-8622 (Proprietary)	Westinghouse ECCS Evaluation Model October 1975 Version	Nov. 1975	15.6	A
WCAP-8623	Westinghouse ECCS Evaluation Model October 1975 Version	Nov. 1975	15.6	A
WCAP-8624	General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables	Sept. 1975	3.10N	U
WCAP-8691 Rev. 1 (Proprietary)	Fuel Rod Bowing Evaluation	July 1979	4.2 4.4	U
WCAP-8692 Rev. 1	Fuel Rod Bowing Evaluation	July 1979	4.2 4.4	U
WCAP-8693	Delta Ferrite Production in Austenitic Stainless Steel Weldments	Jan. 1976	5.2	B
WCAP-8708 (Proprietary)	Multiflex A Fortran IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics	Feb. 1976	3.9N	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8709	Multiflex A Fortran IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics	Feb. 1976	3.9N	A
WCAP-8720 (Proprietary)	Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations	Oct. 1976	4.2	A
WCAP-8760	Failure Modes and Effects Analysis of the Engineering Safeguards Features Actuation System	April 1976	4.6 7.3	U
WCAP-8762	New W Correlation WRB-1 For Predicting Critical Heat Flux In Rod Bundles With Mixing Vane Grids	Oct. 1976	4.4	A
WCAP-8766 (Proprietary)	Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant	May 1976	3.9N	A
WCAP-8768 Rev. 1	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1976	June 1977	4.3 5.4.1 5.4.15	B
WCAP-8768 Rev. 2	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977 - Summer 1978	Oct. 1978	1.5 4.2	B
WCAP-8780	Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant	May 1976	3.9N	A
WCAP-8785	Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations	Oct. 1976	4.2	A
WCAP-8822 (Proprietary)	Mass and Energy Releases Following a Steam Line Rupture	1976	6.2.1.4	--
WCAP-8843 (Proprietary)	MARVEL: A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System	Nov. 1977	6.2.1.4	--
WCAP-8844	MARVEL: A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System	Nov. 1977	6.2.1.4	--

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-8859	TRANFLO: Steam Generator Code Description	Sept. 1976	6.2.1.4	--
WCAP-8860	Mass and Energy Releases Following a Steam Line Rupture	Sept. 1976	6.2.1.4	--
WCAP-8872	Design, Inspection, Operation, and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposure As Low As Reasonably Achievable	April 1977	12.3.1.2	--
WCAP-8892-A	7300 Series Process Control System Noise Tests	June 1977	7.1.2	A
WCAP-8963 (Proprietary)	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis	Nov. 1976	4.2	A
WCAP-8964	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis	Aug. 1977	4.2	A
WCAP-8970 (Proprietary)	Westinghouse ECCS Small Break, October 1975 Model	April 1977	15.6	U
WCAP-8971	Westinghouse ECCS Small Break, October 1975 Model	April 1977	15.6	U
WCAP-8976	Failure Modes and Effects Analysis of the Solid State Full Length Rod Control System	Aug. 1977	4.6 7.7	U
WCAP-9004 (Proprietary)	Inlet Orificing of Open PWR Cores	Jan. 1969	4.4	--
WCAP-9168 (Proprietary)	Westinghouse ECCS Evaluation Model, Modified October 1975 Version	Sept. 1977	15.6	U
WCAP-9169	Westinghouse ECCS Evaluation Model, Modified October 1975 Version	Sept. 1977	15.6	U
WCAP-9179 Rev. 1 (Proprietary)	Properties of Fuel and Core Component Materials	July 1978	4.2	U

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-9220 (Proprietary)	Westinghouse ECCS Evaluation Model, February 1978 Version	Feb. 1978	6.2.1.3	A
WCAP-9220-P-A	Westinghouse ECCS Evaluation Model, 1981 Version	Dec. 1981	15.6	U
WCAP-9921	Westinghouse ECCS Evaluation Model, February 1978 Version	Feb. 1978	6.2.1.3	A
WCAP-9221-P-A Rev. 1	Westinghouse ECCS Evaluation Model, 1981 Version	Dec. 1981	15.6	A
WCAP-9224	Properties of Fuel and Core Component Materials	July 1978	4.2	U
WCAP-9227	Reactor Core Response to Excessive Secondary Steam Releases	Jan. 1978	15.1	U
WCAP-9230 (Proprietary)	Report on the Consequences of a Postulated Feedline Rupture	Jan. 1978	15.2	U
WCAP-9231	Report on the Consequences of a Postulated Feedline Rupture	Jan. 1978	15.2	U
WCAP-9279	Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants	Mar. 1978	3.9N	--
WCAP-9283	Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events	Mar. 1978	3.9N	--
WCAP-9292	Dynamic Fracture Toughness of ASME SA 508 Class 2a and ASME SA 535 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metal	March 1978	5.2	--
WCAP-9401-P-A (Proprietary)	Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly	Aug. 1981	4.2.3.4	--
WCAP-9402-A	Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly	Aug. 1981	4.2.3.4	--
WCAP-9485-A (Proprietary)	PALADON: Westinghouse Nodal Computer Code	Dec. 1978	4.3	A

TABLE 1.6-1 (Cont)

REPORT NO.	TITLE	USNRC SUBMITTAL	REFERENCE SECTION(S)	REVIEW STATUS
WCAP-9486-A	PALADON: Westinghouse Nodal Computer Code	Dec. 1978	4.3	A
WCAP-9735 Rev. 1 (Proprietary)	Multiflex 3.0 - A Fortran IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics (III) Advanced Beam Model	1982	3.9N.1.2	--
WCAP-9736	Multiflex 3.0 - A Fortran IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics (III) Advanced Beam Model	1982	3.9N.1.2	--
WCAP-9745	Results of a Westinghouse Review of Environmental Qualification References for WRD Supplied Category II Equipment with Respect to the Staff Positions in NUREG-0588	1980	3.11	--
NS-TMA-2075 (W)	Westinghouse LOCA Mass and Energy Release Model for Containment Design, March 1979 Version	April 1979	6.2.1.3	--
Report (SWEC)	Report on Soil Densification Program, BVPS-2, September 23, 1976	Sept. 1976	3.7B.2.4	--
Report (SWEC)	Control of Heavy Loads at Nuclear Power Plants, Letters 2DLS-13737, March 26, 1982, and 2DLC-4556, April 2, 1982	April 1982	9.1.5	--
RP-8A (SWEC)	Radiation Shielding Design and Analysis Approach for Light Water Reactor Plants	May 1975	12.3	A
SWECO-7703 (SWEC)	Missile Barrier Interaction	Sept. 1977	3.5.3	--
Report (SWEC)	Control Room Habitability Study for BVPS-1 and BVPS-2	Dec. 1981	2.2.3	A

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 Electrical, Instrumentation, and Control Drawings

Table 1.7-1 identified the compiled listings of safety related electrical, instrumentation, and control drawings used on BVPS-2 and is not included in this updated FSAR. Three copies of these drawings were provided in a separate enclosure to the original FSAR submittal.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 listed the piping and instrumentation diagrams (P&IDs) used on BVPS-2 and is not included in this updated FSAR. Two copies of these drawings were provided in a separate enclosure to the original FSAR submittal.

Flow diagrams are included with the systems described throughout the Updated FSAR. Symbols and abbreviations used in these flow diagrams are illustrated in Figure 1.7-1.

1.7.3 Other Detailed Information (Special Reports and Programs)

Table 1.7-3 identifies special evaluations, documentation, and programs which were referenced in the FSAR and were submitted separately from the FSAR.

Tables for Section 1.7

TABLE 1.7-3

SPECIAL REPORTS AND PROGRAMS

1. Transmitted at time of FSAR Submittal

    a. Failure Modes and Effects Analysis (FMEA)

2. Transmitted after submittal

    a. Fire Protection Evaluation Report (FPER) (Incorporated into the FSAR as Appendix 9.5A in FSAR Amendment 14.)

    b. Beaver Valley Nuclear Power Station Emergency Plan

    c. Beaver Valley Nuclear Power Station Security Plan

    d. Control Room Design Review

    e. Inservice Inspection Program

    f. Equipment Qualification Report

    g. ASME Code Baseline Document (2BVM-179)

    h. Preservice Inspection Program

    I. Technical Specifications

    j. Seismic Design Response Spectra, June 1984

    k. Site Dependent Response Spectra, February 1985

    l. SQRT/PVORT Summary Listing

    m. Test Report on Electrical Separation Verification Test

    n. Detailed Control Room Design Review - Supplemental Report

    o. Fire Protection Safe Shutdown Report

3. Programs reviewed at BVPS-2

    a. Equipment Qualification Documentation (EQD)

    b. Seismic and Dynamic Qualification Program for Safety-Related Equipment

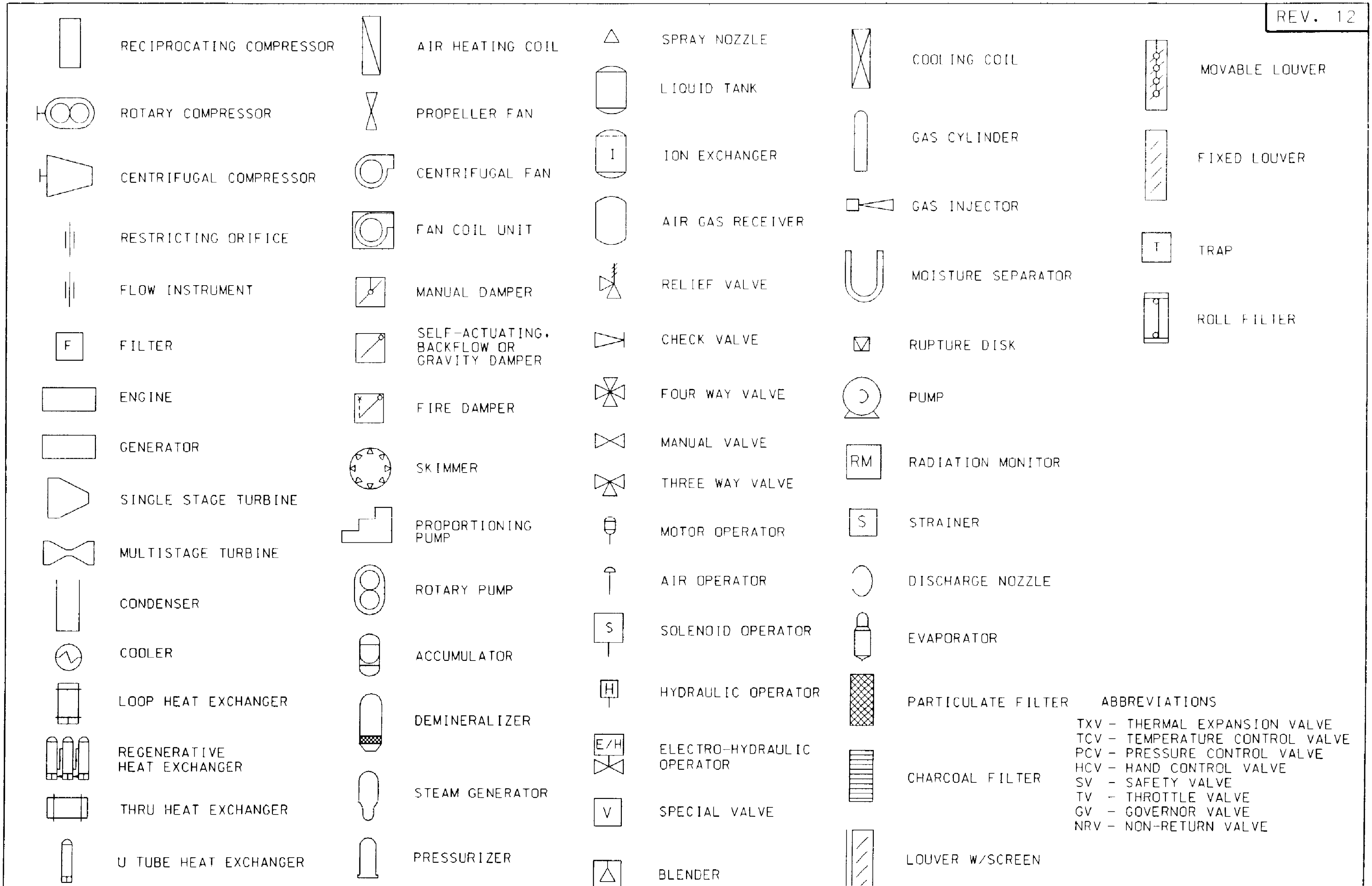

FIGURE 1.7-1
SYMBOLS FOR FIGURES

BEAVER VALLEY POWER STATION UNIT NO. 2
UPDATED FINAL SAFETY ANALYSIS REPORT

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Table 1.8-1 provides an evaluation of the degree of Beaver Valley Power Station-Unit 2 (BVPS) conformance to NRC Division 1 Regulatory Guides. The revisions of the regulatory guides against which BVPS-2 is evaluated are indicated. Any alternatives to the provisions of the regulatory guides are identified and justification is provided where appropriate. FSAR sections applicable to the subject regulatory guide are referenced.

Tables for Section 1.8

TABLE 1.8-1

USNRC REGULATORY GUIDES

RG No. 1.1, Rev. 0
UFSAR Reference Section 6.2.2, 6.3

NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS (NOVEMBER 2, 1970)

Beaver Valley Power Station - Unit 2 meets the intent of Regulatory Guide 1.1 for providing adequate net positive suction head (NPSH) for emergency core cooling and containment heat removal systems pumps with the following alternatives:

> The containment pressure and sump vapor pressure are calculated explicitly on a transient basis and used to calculate the available NPSH for the Recirculation Spray pumps as described in Section 6.2.2.3.2.

RG No. 1.2, Rev. 0
UFSAR Reference Section 5.3

THERMAL SHOCK TO REACTOR PRESSURE VESSELS (NOVEMBER 2, 1970)

The guidance provided by this regulatory guide regarding Section 5.3 thermal shock to the reactor pressure vessel is followed for Beaver Valley Power Station - Unit 2 with the following clarification:

Paragraph C.3

> The vessel design does not preclude the use of an engineering solution to assure adequate recovery of the fracture toughness properties of the vessel material. If additional margin is needed, the reactor vessel can be annealed. This solution was shown to be feasible by EPRI program RP1021-1, "Feasibility and Methodology for Thermal Annealing an Embrittled Reactor Vessel."

RG No. 1.3, Rev. 2

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS (JUNE 1974)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

TABLE 1.8-1 (Cont)

RG No. 1.4, Rev. 2
UFSAR Reference Sections 4.2, 6.2.1, 6.2.4, 6.5.1, 15.1.5, 15.6.3, 15.6.5

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS (JUNE 1974)

The assumptions used for evaluating the potential radiological consequences of a loss-of-coolant accident at Beaver Valley Power Station - Unit 2 are based on Regulatory Guide 1.183. Refer to the position on Regulatory Guide 1.183, later in this table.

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.5, Rev. 0

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (MARCH 10, 1971)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.6, Rev. 0
UFSAR Reference Sections 7.6, 8.3.1.4, 8.3.2, 7.6.1

INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (MARCH 10, 1971)

Methods for establishing the degree of independence between redundant standby (onsite) power sources and between their distribution systems at Beaver Valley Power Station - Unit 2 follow the guidance of this regulatory guide.

RG No. 1.7, Rev. 2
UFSAR Reference Section 6.2.5

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT (NOVEMBER 1978)

The hydrogen control system was originally designed following the guidance of Regulatory Guide 1.7, but with the subsequent 10 CFR 50.44 rule change, the hydrogen recombiners are not required; therefore, the hydrogen recombiners may be maintained, modified or eliminated.

RG No. 1.8
UFSAR Reference Sections 12.5, 13.1, 13.2

PERSONNEL SELECTION AND TRAINING

Application of Regulatory Guide 1.8 to the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

TABLE 1.8-1 (Cont)

RG No. 1.9, Rev. 2
UFSAR Reference Sections 8.3.1.1.15, 14.2.12.54, 14.2.12.55

SELECTION, DESIGN, AND QUALIFICATION OF DIESEL-GENERATOR UNITS USED AS STANDBY (ONSITE) ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS (DECEMBER 1979)

Diesel-generator units used as onsite electric power systems at BVPS-2 have been selected, designed, and qualified following the guidance of this Regulatory Guide with the following alternative and clarifications:

The Class lE diesel generator units were designed and procured following the guidance of Regulatory Guide 1.9, Rev. 0 (March 1971) and IEEE Standard 387-1972. (Revision status at time of design/procurement)

However, the Class lE diesel generators were manufactured to IEEE Standard 387-1977 with the following clarifications:

1. Section 4.1[8]: The BVPS-2 Class lE diesels are qualified utilizing the mild environment concept acknowledged by 10 CFR 50.49(c).

2. Section 4.1[12]: The BVPS-2 Class lE diesels are seismically qualified in accordance with IEEE Standard 344-1971.

3. Section 5.4: The BVPS-2 Class lE diesels are qualified utilizing the mild environment concept acknowledged by 10 CFR 50.49(c).

4. Section 6.3.1: All required tests or analyses have been performed by the manufacturer with the exception of the2-hour short-time/22-hour continuous rating test. This test will be performed at the site in accordance with Section 6.3.1 and the loading sequence specified in Paragraph C.14 of Regulatory Guide 1.9.

RG No. 1.10, Rev. 1
UFSAR Reference Section 3.8.1.6.2

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES (JANUARY 2, 1973)

Regulatory Guide 1.10 was withdrawn (June 1981) and has been superseded by Regulatory Guide 1.136 ("Materials, Construction, and Testing of Concrete Containments (Articles CC-1OOO, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containment)”). However, since a significant portion of Beaver Valley Power Station - Unit 2 design and construction was completed prior to the withdrawal of this regulatory guide, mechanical (Cadweld) splices in reinforcing bars of Category I concrete structures meet the intent of Regulatory Guide 1.10 with the following alternative:

TABLE 1.8-1 (Cont)

Reinforcing bars with a radius of curvature 60 feet-0 inches or greater will be tested at the sampling frequency specified in Paragraphs C.4a and C.4b. Reinforcing bars with a radius of curvature of less than 60 feet-0 inches will be tested using only sister splices with the following frequency for each splicing crew:

1. One sister splice representing the first 10 production splices,
2. Four sister splices representing the next 90 production splices, and
3. One sister splice representing the next and every subsequent 33 production splices.

Testing of the sister splices is in accordance with Paragraphs C.3 and C.5.

The acceptability of this alternative is based on the following:

The second paragraph of Section 3 of Regulatory Guide 1.10 states that production mechanical splice samples for tensile testing should not be used from curved reinforcing sections, and then refers to Paragraph 4b for sampling frequency. Paragraph 4b provides for a combination of production and sister splices, which appears to be an inconsistency.

The regulatory guide assumes that the cadwelders perform splices in the horizontal, vertical, and diagonal directions on the same day. Thus, there would be occasional splices on straight vertical bars which could be alternated with the curve bar splicing to permit the frequency of testing in Paragraph 4b of Regulatory Guide 1.10, which requires both production and sister splices. However, construction is apt to perform splices in one position only for more splices than those requiring another set of tests. The 60-foot radius was established because satisfactory test results have been obtained for splices on bars of this or greater radii.

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. FENOC 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.

TABLE 1.8-1 (Cont)

RG No. 1.11, Rev. 0
UFSAR Reference Section 6.2.4

INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (MARCH 10, 1971)

The guidance of this regulatory guide is followed for instrument lines penetrating primary reactor containment at Beaver Valley Power Station - Unit 2.

RG No. 1.12, Rev. 1
UFSAR Reference Section 3.7B.4.

INSTRUMENTATION FOR EARTHQUAKES (APRIL 1974)

Earthquake instrumentation at Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide with the following exceptions based on industry experience:

1) No triaxial accelerograph unit listed in Section 3.7B.4.2 is located on equipment, piping or supports since experience has shown that data obtained at these locations are obscured by the vibratory motion associated with normal plant operations.

2) There is no immediate indication that the zero period acceleration at the containment mat has been exceeded via a switch. The system will determine within minutes whether actual conditions have exceeded a pre-determined building response spectra and provide indication if the 1/2-SSE has been exceeded.

3) All instruments are oriented to the same azimuths, except one remote accelerograph sensor which is mounted to the containment cranewall. It is oriented such that it will respond to horizontal motion in the radial and tangential directions of the containment.

RG No. 1.13, Rev. 1
UFSAR Reference Sections 9.1.2, 9.1.3, 9.1.4, 9.1.5

SPENT FUEL STORAGE FACILITY DESIGN BASIS (DECEMBER 1975)

The design of spent fuel storage facilities follows the guidance of this regulatory guide with the following alternative:

Local spent fuel pool level alarms are not provided. A fuel pool level alarm is provided in the control room.

The motor-driven platform crane, which is the only crane that is operable over the spent fuel pool, has hoists that are each provided with an adjustable interlock which will stop hoist motion if a preset weight is exceeded. The setpoint is based upon the weight of a single fuel assembly. Accordingly, the interlock must be overridden in order to use the hoist for maintenance operations.

> Administrative procedures require that maintenance operations be carried out along a safe path away from the spent fuel pool. When the motor-driven crane is not in use, administrative procedures require that it be positioned away from the spent fuel pool.

RG No. 1.14, Rev. 1
UFSAR Reference Section 5.4.1.5

REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (AUGUST 1975)

Reactor coolant pump flywheel integrity at Beaver Valley Power Station - Unit 2 is assured by meeting the intent of this regulatory guide as specified in WCAP-8163, September 1973, ("Reactor Coolant Pump Integrity in LOCA") and with the following alternative:

> No post-spin test inspections are performed and pre-spin test inspections are considered adequate because flaw growth attributable to the spin tests (that is, from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than that which nondestructive inspection techniques are capable of detecting.

RG No. 1.15, Rev. 1
UFSAR Reference Section 3.8.1.6.2

TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES (DECEMBER 28, 1972)

Regulatory Guide 1.15 was withdrawn (June 1981) and has been superseded by Regulatory Guide 1.136, Rev. 2, June 1981 (Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, - 2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments")). However, since a significant portion of Beaver Valley Power Station - Unit 2 (BVPS-2) design and construction was completed prior to the withdrawal of this regulatory guide, testing of reinforcing bars for Category I concrete structures at BVPS-2 follows the guidance of this Regulatory Guide 1.15, Rev. 1.

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. FENOC 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.

TABLE 1.8-1 (Cont)

RG No. 1.16, Rev. 4
UFSAR Reference Chapter 16

REPORTING OF OPERATING INFORMATION - APPENDIX A TECHNICAL SPECIFICATIONS (AUGUST 1975)

Reporting of operating information for Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide, except for the Monthly Operating Report, which will follow the guidance provided in NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report" dated May 15, 1997.

RG No. 1.17, Rev. 1
UFSAR Reference Section 13.6

PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE (JUNE 1973)

The protection of Beaver Valley Power Station - Unit 2 against industrial sabotage is provided for by following the guidance of this regulatory guide.

RG No. 1.18, Rev. 1
UFSAR Reference Section 3.8.1

STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS (DECEMBER 28, 1972)

Regulatory Guide 1.18 was withdrawn (June 1981) and has been superseded by Regulatory Guide 1.136, Rev. 2, June 1981 (Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments")). However, since a significant portion of Beaver Valley Power Station - Unit 2 (BVPS-2) design and construction was completed prior to the withdrawal of this regulatory guide, the structural acceptance test for the containment structure at BVPS-2 follows the guidance of this regulatory guide with the following clarifications:

As permitted by Paragraph C.3 of the Regulatory Guide, minor changes will be made in the selection of locations for deflection measurements around the access openings to account for the thickened ring beam.

The containment structure is a nonprototype structure similar in design and construction to those referenced in UFSAR Section 3.8.1.1.1. Therefore, the regulatory positions for a nonprototype structure are followed.

TABLE 1.8-1 (Cont)

RG No. 1.19, Rev. 1
UFSAR Reference Sections 3.8.1.2.3, 3.8.1.7.2

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS (AUGUST 11, 1972)

Regulatory Guide 1.19 was withdrawn (June 1981) and has been superseded by Regulatory Guide 1.136, Rev. 2, June 1981, ["Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments")]. However, since a significant portion of BVPS-2 design and construction was completed prior to the withdrawal of this regulatory guide, nondestructive examination of primary containment liner welds at BVPS-2 meets the intent of Regulatory Guide 1.19 with the following alternatives:

Paragraph C.1.d

Where leak chase system channels are installed over liner seam welds, the nondestructive examination of these seam welds is to be done by either 1) use of a vacuum box test before application of the channel, or 2) application of soapsuds to the liner seam while the channel is air-pressurized to the calculated peak containment pressure. Necessary repairs are to be made up to completion of a successful leaktight test.

After completion of this test, a test of the channel-to-liner welds is to be performed by either 1) evacuating the air from the channel, and then pressurizing with Freon 22 to calculated peak containment pressure and examining these welds with a Halogen detector, or 2) pressurizing the test channels to containment peak pressure with air; and, if any indicated loss of channel test pressure occurs, within 2 hours as evidenced by the test gage, the channel-to-liner welds should be soap bubble tested. Necessary repairs are to be made up to completion of a successful leaktight test. The Freon is to be purged after the test.

Paragraphs C.2.a, C.7.a, C.7.b, and C.7.d

Nondestructive test methods and acceptance standards are in accordance with ASME Section III, Subsection NE-5300, which corresponds to the referenced Subsection NE-5120.

RG No. 1.20, Rev. 2
UFSAR Reference Section 3.9.2

COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING (MAY 1976)

BVPS-2 follows the guidance of Regulatory Guide 1.20 for a comprehensive vibration assessment program which verifies the structural integrity of the reactor internals for flow-induced vibrations prior to commercial operation.

TABLE 1.8-1 (Cont)

Briefly, this comprehensive program considers both the prototype reactor and subsequent reactors as follows:

> For each prototype reactor internals design, a program of vibration analysis, measurement, and inspection has been developed and reviewed by the USNRC. This is documented in WCAP-7879.

The reactor internals similar to the prototype design are subjected during hot functional testing to the same system flow conditions imposed on the prototype design applicable, and for the same duration. Pre- and post-test inspections will be conducted to assure that the internals are well-behaved and that no excessive motion or wear are experienced. Refer to UFSAR Section 3.9N.2.4 for details on the testing and inspections.

<u>RG No. 1.21, Rev. 1</u>
UFSAR Reference Section 11.5.1

<u>MEASURING, EVALUATING, AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (JUNE 1974)</u>

The monitoring, evaluating and reporting of radioactivity in solid wastes and releases of radioactive materials in liquid and gaseous effluents at BVPS-2 will meet the intent of this regulatory guide with the following alternative:

> For the turbine building drains, grab sample systems are provided for each of the three transfer pump systems. Normally, grab samples will be taken on a weekly basis. However, in the event of a high activity alarm from the main steam discharge monitors or the steam generator blowdown sample monitor, grab samples from the turbine building drains will be taken on a hourly basis. If any of these samples exceeds a predetermined activity level, the isolation valve for that transfer system will be closed and the flow will be diverted to the liquid waste system.

<u>RG No. 1.22, Rev. 0</u>
UFSAR Reference Sections 7.1.2, 7.2, 7.3.2.2, 7.1.2.4

<u>PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS (FEBRUARY 17, 1972)</u>

Periodic testing of protection system actuation functions for Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.23, Rev. 0
UFSAR Reference Section 2.3.3

ONSITE METEOROLOGICAL PROGRAMS (FEBRUARY 17, 1972)

Onsite meteorological programs for Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide.

RG No. 1.24, Rev. 0
UFSAR Reference Section 15.7.1.3

ASSUMPTIONS USED FOR EVALUATING POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE (MARCH 23, 1972)

Beaver Valley Power Station - Unit 2 evaluation of the potential radiological consequences of a pressurized water reactor radioactive gas storage tank rupture meets the intent of this regulatory guide. The following alternatives were considered prudent:

Paragraph C.1.a

In recognition of specific plant equipment arrangements for gaseous waste handling, the system component producing the worst environmental impact was identified and additional conservatism was appropriately applied.

Paragraph C.2

Atmospheric diffusion (X/Q) values were calculated using the latest approved techniques which are provided in Regulatory Guide 1.145.

BVPS takes some alternatives to Section C.3 dose calculation methodology.

RG No. 1.25, Rev. 0

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 (MARCH 23, 1972)

The assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility at Beaver Valley Power Station - Unit 2 are based on Regulatory Guide 1.183. Refer to the position on Regulatory Guide 1.183, later in this table.

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

TABLE 1.8-1 (Cont)

RG NO. 1.26, Rev. 3
UFSAR Reference Section 3.2.2

QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS (FEBRUARY 1976)

Quality group classifications and standards for water-, steam-, and radioactive-waste-containing components of Beaver Valley Power Station - Unit 2 meet the intent of Regulatory Guide 1.26 with the following alternatives:

1. The safety class terminology of ANSI N18.2-1973 and ANSI N18.2a-1975 is used instead of the quality group terminology. Thus, the terms Safety Class 1, Safety Class 2, Safety Class 3, and Non-nuclear Safety (NNS) Class are used instead of Quality Groups A, B, C, and D, respectively, and are consistent with present nuclear industry practice.

2. Paragraph NB-7153 of the ASME Section III Code requires that there be no valves between a code safety valve and its relief point unless special interlocks prevent shutoff without other protection capacity. Therefore, as an alternative to Paragraphs C.1.e and C.2.c, a single safety valve designed, manufactured, and tested in accordance with ASME III Division 1 is considered acceptable as the boundary between the reactor coolant pressure boundary and a lower safety class or NNS class line.

3. Portions of the emergency diesel generator cooling water system, considered by the vendor to be parts of the engine (as distinguished from auxiliary support systems), were built to the manufacturer’s standards rather than ASME III. These are identified in Table 3.2-1 and Section 9.5.5. The components used are of high quality, proven by experience, and were designed, fabricated, erected, and tested under the vendor's Quality Assurance Program which meets the requirements of 10CFR5O, Appendix B. Similar equipment has been accepted by the NRC for other nuclear power plant applications.

4. Regarding Regulatory Positions C.1 and C.2, all instrument tubing, classified as Safety Class 2 or 3, are designed to ASME Section III rules and installed in accordance with the BVPS-2 quality assurance program for safety-related equipment.

TABLE 1.8-1 (Cont)

RG NO. 1.27, Rev. 2
UFSAR Reference Sections 2.4.11.6, 9.2.5

ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (JANUARY 1976)

The ultimate heat sink for Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide.

RG No. 1.28, Rev. 2
UFSAR Reference Sections 17.1.2, 17.2

QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (FEBRUARY 1979)

This regulatory guide does not apply to the Beaver Valley Power Station - Unit 2 (BVPS-2) Quality Assurance Program since it is applicable to construction permit applicants docketed after October 1979. BVPS-2, docketed October 20, 1972, meets the requirements of Appendix B to 10 CFR 50 with the BVPS-2 Design and Construction Quality Assurance Program submitted and approved through Appendix A of the BVPS-2 PSAR. Regulatory Guide 1.28 does not apply to the BVPS-2 quality assurance program for plant operations since it is applicable only to the plant design and construction phase.

RG No. 1.29, Rev. 3
UFSAR Reference Section 3.2.1

SEISMIC DESIGN CLASSIFICATION (SEPTEMBER 1978)

The seismic design classification of structures, systems, and components at Beaver Valley Power Station Unit 2 follows the guidance of Regulatory Guide 1.29 with the following clarification:

> Components within the NSSS vendor scope of supply which are placed in Safety Class 3 per ANSI N18.2, Paragraphs 2.2.3 (1), (3), or (4) may be classified Seismic Category II if failure during or following an ANS Condition II event would result in consequences no more severe than allowed for an ANS Condition III event.
>
> For the Balance of Plant, each component which is required to mitigate the consequences of an accident, as defined in ANSI N18.2, shall be classified Seismic Category I. In addition, all components classified as Safety Class 1, 2, or 3 shall be designated Seismic Category I. Seismic Category I components, structures, and systems shall be designed to remain functional in the event of the safe shutdown earthquake (SSE). All Seismic Category I components are designed and constructed to Quality Assurance (QA) Category I requirements.

TABLE 1.8-1 (Cont)

Portions of structures, systems, or components whose continued function after an SSE are not required, but whose failure could reduce the functioning of other safety-related structures, systems, or components shall be designated Seismic Category II. These structures, systems, or components shall either be seismically designed, located to preclude interactions, further restrained, structurally upgraded, or proven incapable of affecting safety.

Seismic Category I design requirements shall extend to the first seismic restraint beyond the seismic boundary and shall include the interface portion of the boundary itself (that is, for piping systems, the isolation valve at a boundary between Seismic Category I and nonseismic portions shall be designated Seismic Category I. The piping up to and including the first seismic restraint beyond the valve shall be designed to Seismic Category I requirements but, since the piping is not required to be ASME III Class 1, 2, or 3, shall not be designated Seismic Category I). By this means, the Seismic Category I boundary is defined with respect to safety-related function, and the interfacing portions meet the seismic design requirements in order to ensure the integrity of the boundary.

Structures, systems, and components designed in accordance with Paragraphs C.2 and C.3 of the regulatory guide are designated Seismic Category II and are constructed to QA Category II or III requirements.

RG No. 1.30, Rev. 0
UFSAR Reference Sections 17.1, 17.2

QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRIC EQUIPMENT (AUGUST 11, 1972)

The guidance provided by this regulatory guide for quality assurance requirements for the installation, inspection, and testing of instrumentation and electric equipment was followed during the construction phase of Beaver Valley Power Station Unit 2. Application of Regulatory Guide 1.30 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.31, Rev. 3
UFSAR Reference Sections 4.5.2.4, 5.2.3.4.6, 5.3.1.4, 6.1.1.1, 10.3.6.2

CONTROL OF FERRITE CONTENT IN STAINLESS STEEL WELD METAL (APRIL 1978)

For nuclear steam supply system fabrication, Beaver Valley Power Station - Unit 2 (BVPS-2) meets the intent of Regulatory Guide 1.31 for control of ferrite content in stainless steel weld metal by following acceptable alternative criteria. Westinghouse submitted a delta ferrite verification program for austenitic stainless steel weldments in WCAP-8324-A, June 1974, which the staff subsequently approved as a valid approach in a letter on December 30, 1974.

For balance of plant fabrication, BVPS-2 meets the intent of Regulatory Guide 1.31 by following the guidance of either Safety Guide 31, Regulatory Guide 1.31, Revision 1, Materials Engineering Branch Technical Position MTEB 5-1, or Regulatory Guide 1.31, Revision 3.

Following the guidance of any of the preceding document revisions was based primarily on the revision in effect on the date of the last specification revision wherein the regulatory guide was invoked. Since each revision of the regulatory guide is less restrictive than the foregoing, following the guidance of any of the revisions is considered acceptable.

RG No. 1.32, Rev. 2
UFSAR Reference Sections 7.5, 8.1.5, 8.2, 8.3.1, 8.3.2, 7.5.2.3.1.3

CRITERIA FOR SAFETY-RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS (FEBRUARY 1977)

The design of the safety-related electric power systems for Beaver Valley Power Station - Unit 2 (BVPS-2) follows IEEE Standard 308-1974, and the guidance of Regulatory Guide 1.32, with the following clarifications:

> Two immediate access offsite power circuits are provided. Each circuit is designed to be immediately available following a loss of onsite alternating current power supplies so that sufficient power capacity remains for an orderly shutdown and to supply all train related engineered safety feature loads:
>
> Each battery charger that supplies Class IE 125 V dc systems is designed with full capacity and capability to supply the largest combined demands of the various steady state loads while simultaneously providing sufficient power for adequate charging capacity to restore the battery from the design minimum charged state to the charged state irrespective of the BVPS-2 status during which these demands occur.

For test methods, procedures, and intervals for all Class IE battery performance discharge and service tests, refer to the position on Regulatory Guide 1.129.

RG No. 1.33
UFSAR Reference Sections 13.4, 13.5, 17.2

QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Application of Regulatory Guide 1.33 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.34, Rev. 0
UFSAR Reference Section 5.2.3

CONTROL OF ELECTROSLAG WELD PROPERTIES (DECEMBER 28, 1972)

The guidance provided by this regulatory guide regarding control of electroslag weld properties was followed for fabrication of applicable components for Beaver Valley Power Station - Unit 2.

RG NO. 1.35, Rev. 2

INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (JANUARY 1976)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.36, Rev. 0
UFSAR Reference Sections 5.2.3, 6.1.1

NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (FEBRUARY 23, 1973)

Nonmetallic thermal insulation for austenitic stainless steel used at Beaver Valley Power Station - Unit 2 meets the intent of this regulatory guide. As an alternative to controlled packaging and shipping described in Paragraph C.1, receipt inspection and tests are required by specification. This testing and inspection consists of visual inspection for physical or water damage to all cartons. Damaged cartons are segregated. Potentially contaminated insulation is not accepted, unless randomly selected samples from each carton are shown to be acceptable after being resubjected to the production test outlined in this regulatory guide.

TABLE 1.8-1 (Cont)

RG NO. 1.37, Rev. 0
UFSAR Reference Sections 6.1.1.1, 17.2, 5.2.3.4.1

QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS (MARCH 16, 1973)

During the construction phase, quality assurance requirements for cleaning of fluid systems and associated components at Beaver Valley Power Station - Unit 2 met the intent of this regulatory guide with the following alternatives:

Paragraph C.3

The water quality for final flushes of fluid systems and associated components is at least equivalent to the quality of the operating system water, except for the oxygen content.

Dissolved oxygen content of water cannot be maintained at reactor quality during flushing of open systems.

The maximum particle size criteria for class B cleanness is consistent with Section 3.1.2 except in the Recirculation Spray System (RSS).

Particles of a maximum size of 1/8 inch in any dimension are allowed in the RSS. This particle size limit was chosen since it is smaller than the openings in the recirculation spray system nozzles and the smallest coolant flow channel in the reactor core. Therefore, this exception will have no effect on the recirculation spray systems ability to perform its intended safety function.

Paragraph C.4

Expendable materials, that is, inks and related products, temperature indicating sticks, tapes, gummed labels, wrapping materials (other than polyethylene), water soluble dam materials, lubricants, nondestructive testing penetrant materials and couplants which contact stainless steel or nickel alloy surfaces are in accordance with the following:

1. They do not contain the following as basic and essential chemical constituents: lead, zinc, copper, mercury, cadmium and other low melting point metals, their alloys and/or compounds.

2. Prescribed maximum levels of water leachable chlorides, total halogens, and sulfur and its compounds are imposed on expendable products.

TABLE 1.8-1 (Cont)

Contamination levels in expendable products are based upon safe practices and industrial availability. Contaminant levels are controlled such that subsequent removal by standard cleaning methods will result in the achievement of final acceptable levels which are not detrimental to the materials.

Application of Regulatory Guide 1.37 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.38
UFSAR Reference Section 17.2

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

Application of Regulatory Guide 1.38 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.39, UFSAR Reference Sections 12.5.3, 17.2

HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

Application of Regulatory Guide 1.39 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.40, Rev. 0
UFSAR Reference Sections 3.11, 8.3

QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS (MARCH 16, 1973)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2 since there are no continuous-duty Class IE motors installed inside the containment.

RG No. 1.41, Rev. 0
UFSAR Reference Sections 8.1, 8.3, 14.2.12.54

PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEMS TO VERIFY PROPER LOAD GROUP ASSIGNMENTS (MARCH 16, 1973)

Onsite electric power systems at Beaver Valley Power Station - Unit 2 designed in accordance with Regulatory Guides 1.6 and 1.32 will be tested in accordance with the intent of Regulatory Guide 1.41 during preoperational testing. Following major modifications or repairs appropriate testing will be performed to demonstrate operability and functional capability as required.

TABLE 1.8-1 (Cont)

RG No. 1.42, Rev. 1

INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHT-WATER COOLED NUCLEAR POWER REACTORS (MARCH 1976)

This regulatory guide was withdrawn March 1976.

RG No. 1.43, Rev. 0
UFSAR Reference Section 5.3.1.4, 5.2.3.3.2

CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (MAY 1973)

For balance-of-plant components, Regulatory Guide 1.43 is not applicable, since stainless steel weld cladding of low-alloy steel is not used in fabrication of such components.

For NSSS components, Beaver Valley Power Station - Unit 2 follows the guidance of Regulatory Guide 1.43 with the exception of Paragaph C.1.a. For Paragraph C.1.a, Westinghouse practices are considered to achieve the same purpose as Regulatory Guide 1.43 by requiring qualification of any high heat input process used on ASME SA-508 Class 2 material by following the recommendations of Paragraph C.2 of the regulatory guide.

RG No. 1.44, Rev. 0
UFSAR Reference Sections 4.5.1, 4.5.2.4, 5.2.3.4, 5.3.1.4, 6.1.1.1, 10.3.6.2

CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (MAY 1973)

Control of the use of stainless steels subject to sensitization at BVPS-2 meets the intent of Regulatory Guide 1.44 with the following clarifications and alternatives:

> The following applies to NSSS fabrication with respect to Regulatory Position C.3:
>
> > The Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as all plates, sheets, bars, pipe, and tubes, as well as forgings, fittings, and other shaped products which do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenched. Stainless steel cast metal and weld deposits (including weld deposited safe ends), which contain a minimum of 5 percent ferrite, are not considered to be susceptible to sensitization, and therefore, are not corrosion tested. When testing is required, the tests are performed in accordance with ASTM A262-70,

TABLE 1.8-1 (Cont)

Practice A or E, as amended by Westinghouse Process Specification 84201MW. This process specification supplements the A262 specification since the latter does not define specimen removal location and does not adequately define bend testing criteria for thick and complex stainless steel raw material. The Westinghouse specification requires that 1) specimens be removed from the same location from which mechanical test specimens are removed, and 2) the bend test diameter must be 4X material thickness instead of 1X (Paragraph 36.1, ASTM A262-70). This second modification is based on the fact that almost all stainless steel materials procured by Westinghouse are eventually welded, and the 4X thickness bend test diameter is required for weldments.

The following applies to balance of plant fabrication with respect to Regulatory Position C.6:

1. The ASTM A708-74 standard is used to perform the intergranular corrosion testing for field fabrication and erection of ASME III piping. The radius of the bend specimen is as specified in ASME IX with the weld metal-base metal interface located at the centerline of the bend.

2. Shop fabrication of ASME III piping and field fabrication of ASME III tanks require control of heat input during welding so as to avoid severe sensitization of the weld zone. The heat input during shop fabrication of ASME III piping is based on procedure qualifications in accordance with Regulatory Guide 1.31, Rev. 1, and, during field fabrication of ASME III tanks, is 50,000 (maximum) joules per inch. In addition, the maximum interpass temperature is limited to 350°F. While no testing for intergranular corrosion during weld procedure qualification is required, the above controls ensure that base material will not be severely sensitized during welding.

3. Fabrication of ASME III components other than those identified in Items 1 and 2 above is performed in fabrication shops and requires a maximum interpass temperature of 350°F. While no testing for intergranular corrosion during weld procedure qualification is required, this specific control reduces the possibility of a severely sensitized heat-affected zone during welding. In addition, the need for the shop fabricator to provide unsensitized components is specifically identified in all procurement specifications by requiring supplied material to be capable of meeting ASTM A262, Practice A or E. Shop practice generally recognizes the need to limit heat input during welding through good fit-up, adequate welder accessibility, proper positioning,

TABLE 1.8-1 (Cont)

and close supervision. Finally, most of the pressure retaining components of the reactor coolant pressure boundary piping system are castings and, because of their delta ferrite content, are highly resistant to stress corrosion cracking.

The following applies to the operational phase of BVPS-2.

Stainless wrought products of the AISI Type 304 and 316 will be obtained in the solution heat treatment condition with a sufficiently rapid cooling through the 800°F to 1500°F range by water quench. Materials not meeting this condition (oil or air quench) will be tested for sensitization in accordance with ASTM A262-70, Practice A or E, or with ASTM A708, and bent for examination in accordance with weld test bending requirements of ASME Section IX.

Qualification of AISI Type 304 and 316 welding procedures will be in accordance with the requirements of ASME Section IX except that the maximum amperage (heat input) specified in weld technique sheets shall not exceed that qualified by the controlling Weld Procedure Qualification Report. Materials used for qualification will be commercially available AISI Type 304 and 316. ASTM A708 shall be used as the corrosion test on bend specimens to demonstrate that the welding procedure has not caused the base metal heat affected zone to become excessively sensitized. Corrosion test bend specimens shall consist of two side bends or one face bend and one root bend as required by ASME Section IX taken from the qualification test plate thickness. Corrosion test bend specimens shall be bent over the appropriate radius specified in ASME IX for bend specimens with the weld metal-base metal interface located at the centerline of the bend. The specimens shall be evaluated as stated in ASTM A708 and shall indicate the absence of intergranular attack. Corrosion tests shall be incorporated as part of the weld procedure qualification test record.

RG No. 1.45, Rev. 0
UFSAR Reference Section 5.2.5

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS (MAY 1973)

The leakage detection systems provided to detect leakage from the reactor coolant pressure boundary (RCPB) meet the intent of Regulatory Guide 1.45 with the following clarification of Paragraph C.5 and alternative to Paragraph C.6:

TABLE 1.8-1 (Cont)

Paragraph C.5

The sensitivity and response time of each leakage detection system employed to collect unidentified leakage are as shown in the following table:

SYSTEM	SENSITIVITY AND RESPONSE TIME
Sump levels	1 gpm in less than 1 hour
Air cooler condensate Drain flow rate	1 gpm in less than 1 hour
Air cooler outlet Water temperature increase	5 gpm in 1 hour
Containment atmosphere humidity, temperature, and pressure	5 gpm in 1 hour
Containment atmosphere radioactivity monitor	1 gpm in less than 1 hour assuming primary coolant radioactivity concentrations stated in the environmental report. For other conditions the equilibrium activity of the reactor coolant must be sufficiently high and the equilibrium activity of the containment atmosphere must be below a level that would mask the change in activity corresponding to this leak rate.

Paragraph C.6

BVPS-2 does not fully follow the guidance of Paragraph C.6 in that the containment sump level detection systems which meet the detection sensitivity requirements of normal plant operation (that is, 1 gpm in 1 hour) are not designed to meet seismic requirements. However, a seismically qualified, less sensitive, additional sump level instrumentation system, powered from Class IE electrical buses, is provided with remote displays in the control room to monitor for excessive leakage following seismic events. This instrumentation is qualified for seismic loads up to safe shutdown earthquake loads.

The SWEC interpretation of the RCPB leakage detection system requirements and Paragraph C.5 of Regulatory Guide 1.45 agrees with the USNRC staff interpretation given in Section 5.2.5 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Rev. 2, dated September 1975, and the corresponding USNRC Standard Review Plan, NUREG-75/087, dated November 1975. In addition, the USNRC staff has agreed to this position by accepting the leakage detection systems that have been described in safety analysis reports for SWEC plants (SWESSAR P-1, Section 5.2.7).

TABLE 1.8-1 (Cont)

RG No. 1.46, Rev. 0
UFSAR Reference Section 3.6

PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (MAY 1973)

Beaver Valley Power Station - Unit 2 meets the intent of Regulatory Guide 1.46 for the protection against pipe whip inside containment with the following clarifications and alternatives:

Paragraph C.1.b

The piping breaks are postulated to occur at any intermediate locations where the maximum stress exceeds 3.0 S as calculated by equation (10) and by either equation (12) or equation (13) in Paragraph NB-3653 of the ASME Code Section III.

Paragraph C.2.b

The piping breaks are postulated to occur at any intermediate locations where either the circumferential or the longitudinal stresses derived on an elastically calculated basis under the loadings associated with specified seismic events and operational plant conditions exceeds 0.8 (1.8 S_h + S_A) as calculated with equations (9) and (10) in Paragraph NC-3652 of the ASME Code, Section III (Section 3.6B.2).

Paragraph C.3.a

Longitudinal breaks are not postulated at terminal points nor at locations where the criterion for a minimum number of break locations must be satisfied.

Paragraph C.3.b

Circumferential breaks are not postulated where detailed stress analysis shows that circumferential stress is at least 1.5 times that in the axial direction. Longitudinal breaks are not postulated where axial stress is at least 1.5 times the circumferential stress.

RG No. 1.47, Rev. 0
UFSAR Reference Section 7.1.2

BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS (MAY 1973)

The bypass and inoperable status indication system for Beaver Valley Power Station - Unit 2 follows the guidance provided by this regulatory guide.

Automatic bypass indication is provided in the control room for each safety-related system train and is at the system level.

TABLE 1.8-1 (Cont)

RG No. 1.48, Rev. 0
UFSAR Reference Sections 3.9N.1, 3.9N.3, 3.9B.3

DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (MAY 1973)

The criteria and procedures used in the design of Beaver Valley Power Station - Unit 2 concerning design limits and loading combinations for Seismic Category 1 fluid system components, satisfy the requirements of General Design Criterion 2 and thereby meet the intent of Regulatory Guide 1.48 in an alternate acceptable manner.

To ensure the structural integrity of fluid systems components, the limits given in Section 3.9N.1 will be used in the design and analysis of ASME Code Class 1 components within the Westinghouse scope of supply. For ASME Code Class 2 and 3 components within the Westinghouse scope of supply, the limits given in Section 3.9N.3 are used.

Criteria outlined in Section 3.9B.3 are applicable to balance of plant ASME Code Class 1, 2, and 3 components and are consistent with the applicable provisions of Section III of the ASME Boiler and Pressure Vessel Code. Supplemental criteria relating to assurance of component operability are drawn from developing industry code activities, including the efforts of the ANSI N45 Committees.

RG No. 1.49, Rev. 1
UFSAR Reference Section 1.1

POWER LEVELS OF NUCLEAR POWER PLANTS (DECEMBER 1973)

Beaver Valley Power Station - Unit 2 design and analysis involving power level follows the guidance of this regulatory guide.

RG No. 1.50, Rev. 0
UFSAR Reference Section 5.2.3.3, 5.3.1.4

CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW-ALLOY STEEL (MAY 1973)

Beaver Valley Power Station - Unit 2 (BVPS-2) meets the intent of Regulatory Guide 1.50 for the control of preheat temperature for welding of low-alloy steel with the following clarifications and alternatives:

TABLE 1.8-1 (Cont)

Westinghouse considers that Regulatory Guide 1.50 applies only to ASME Code, Section III, Class 1 components. The Westinghouse practice for Class 1 components follows welding procedures which meet the criteria in Section IX of the ASME Code and WCAP-8577, February 1976, "The Application of Preheat Temperatures after Welding Pressure Vessel Steels," in lieu of Paragraphs C.1.b and C.2 of Regulatory Guide 1.50. Westinghouse experience has shown high quality and integrity for welds using these procedures which have been found acceptable by the USNRC.

For the remainder of BVPS-2 welds, in cases when it is impractical to maintain the preheat temperature until a post- weld heat treatment has been performed (Paragraph C.2), a minimum temperature of 300°F is maintained for two hours per inch of thickness. This practice of maintaining the completed weld at an elevated temperature for the prescribed period of time allows hydrogen to effuse from the weld zone, reducing the tendency to form cracks in the weldment.

RG No. 1.51, Rev. 0

INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS (MAY 1973)

This regulatory guide was withdrawn July 1975.

RG No. 1.52, Rev. 2
UFSAR Reference Section 6.5.1.2

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)

The design, testing, and maintenance criteria for the post-accident engineered safety feature (ESF) atmosphere cleanup system at BVPS-2 meets the intent of this regulatory guide with the following alternatives:

Paragraph C.2.b

Tornado protection is not considered necessary for the ESF atmosphere cleanup systems because of the low probability of a joint occurrence of a tornado and those DBAs that would require operation of the ESF atmospheric cleanup systems (References: SRP 6.5.1 and Regulatory Guide 1.117).

Paragraph C.2.f

Filter component layouts consist of a maximum of four high efficiency particulate air (HEPA) filters high and five HEPA filters wide.

TABLE 1.8-1 (Cont)

Paragraph C.2.g

Pressure drop is indicated at each airstream HEPA filter and annunciated in the main control room.

Paragraph C.2.h

Annunciator functions are incorporated in the overall system design. Annunciators are not safety-related; therefore, they are not designed in accordance with IEEE Standard 279-1971.

Paragraph C.2.j

The trains of the ESF atmosphere cleanup systems are designed to be removed as intact units or as a minimum number of segmented sections. Individual filter components will be removed prior to cutting the housing into segmented sections.

Paragraph C.2.l

1. The ductwork leak tests are performed in accordance with ANSI N510-1975 with the alternative that ASME performance test code 19.5-1971 will be used in lieu of paragraph 6.3.1. The equipment and equipment arrangement based on the above ASME performance test code will provide test results equivalent to results obtained by equipment specified in paragraph 6.3.1 of ANSI N510-1975.

2. The air leakage rate for ductwork will be established based on the air cleaning effectiveness provisions defined in Paragraph 4.12.1 of ANSI N509-1976. Because the ductwork carrying contaminated air upstream of the filters is under negative pressure and there is no leakage from the ductwork to surrounding space, and because any air leak from the ductwork under positive pressure is on the downstream side of the filters (filtered air), the leakage for ductwork upstream of the filters and downstream of the fans is limited to 1 percent of rated flow at internal design pressure.

3. For ductwork between the outlet of filters and the inlet of fans, the leakage is limited to 0.5 percent of rated flow at internal design pressure.

4. The duct leakage for the control room emergency ventilation system is limited to 5 percent of rated flow. This rate of leakage will not adversely affect the system capability to pressurize the control room envelope.

   All components of the system carrying unfiltered outside air are under negative pressure, precluding any leakage of unfiltered air to the control room envelope.

TABLE 1.8-1 (Cont)

5. The filter housing leak tests will be performed in accordance with paragraph 4.12 of ANSI N509-1976 with the alternative that the housing maximum allowable leakage will be as specified in Table 4-3 of ANSI N509-1980. The leakage test procedure will be developed based on Section 6 of ANSI N510-1980.

Paragraph C.3.a

Demisters meet Underwriters Laboratories (UL) Class 2 requirements.

Paragraph C.3.e

Filter and adsorber mounting frames are constructed and designed in accordance with the recommendations of Section 4.3 of ORNL-NSIC-65, except for the frame tolerance guidelines in Table 4.2. The tolerances for HEPA filter and adsorber mountings are sufficient to satisfy the bank leak test criteria of paragraphs C.5.c and C.5.d of this Regulatory Guide.

Paragraph C.3.f

Filter component layouts consist of a maximum of four HEPA filters high and five HEPA filters wide.

Paragraph C.3.g

1. Stainless steel materials for filter housings are procured to ASTM material specification A167 Type 304 in addition to those listed in paragraph 4.3 of NSIC-65/ERDA 76-21.

2. Welding is performed in accordance with AWS D1.1 or ASME IX; therefore, the workmanship samples recommended in paragraph 7.3 of ANSI N509-1976 are not used to demonstrate welders' qualifications to perform production work.

Paragraph C.3.l

The qualification of unused impregnated carbon will be in accordance with Table 5-1 of ANSI N509-1980.

Paragraph C.3.g

1. The type and application of protective coatings on internal surfaces is controlled in accordance with the designer's specifications, which specify high quality materials and application methods in accordance with the coating manufacturer's instructions. These practices are used in lieu of the recommendations in paragraphs 5.7.1 and 5.6.4 of ANSI N509-1976.

TABLE 1.8-1 (Cont)

2. An alternative is taken to paragraph 5.7.2 of ANSI N509-1976. Copies of fan ratings or test reports are not provided. However, certified fan performance curves are furnished.

3. The balancing techniques defined in paragraph 5.7.3 of ANSI N509-1976 will be used except that normal industrial practice will be used to determine the maximum permissible vibration velocity criteria.

4. The fan drawings follow the recommendations of paragraph 5.7.4 of ANSI N509-1976 with the alternative that all information about lubricants and lubrication is contained in operation and instruction manuals.

5. Where AMCA certified ratings are submitted, documentation developed in conjunction with the certification of fans is not furnished in accordance with paragraph 5.7.5 of ANSI N509-1976.

<u>Paragraph C.3.n</u>

1. Welding procedures, welders, and welding operators are qualified in accordance with designer's welding specifications. These specifications are in general conformance with AWS D1.1 and ASME Section IX, which are recommended in paragraph 7.3 of ANSI N509-1976. Production weld visual acceptance criteria, which are based on AWS D1.1, are used in lieu of workmanship samples recommended in ANSI N509-1976, paragraph 7.3.

2. Materials for ductwork are procured to ASTM material specifications, such as A276 Type 304, A500 Gr B, A575 Gr N1020, and A576 Gr 1020, in addition to those listed in paragraph 5.10.6 of ANSI N509-1976.

3. An alternative is taken to paragraph 5.10.3.5 of ANSI N509-1976. While ductwork, as a structure, has a resonant frequency above 25 Hz, this may not be true for the unsupported plate or sheet sections. ANSI N509-1980, which has been issued since the issuance of this regulatory guide, has deleted this provision. Tympanic vibration modes of the duct are not considered in the design because the loads will be small and minimum thickness of duct material is 20 gauge. This is more conservative than SMACNA provisions.

TABLE 1.8-1 (Cont)

Paragraph C.3.p

The following alternatives are taken to paragraph 5.9 of ANSI N509-1976:

1. Isolation dampers used in the supplementary leak collection and release system are neither designed nor constructed to the recommendations of ANSI B31.1. The system is designed to ensure that the leakage through the dampers is from the noncontaminated to the contaminated portion of the system and the flow is exhausted through the filters before being released to the atmosphere. Therefore, the uncontrolled release of radioactivity is precluded and the intent of Section 5.9 of ANSI N509-1976 is satisfied.

2. Butterfly valves used for control room isolation are in accordance with the ASME III code.

3. One Class B damper of each size will be tested for leakage rate instead of testing every damper.

4. Welding is controlled in accordance with the designers' specifications using visual acceptance criteria, which are based on AWS D1.1 in lieu of the standards recommended in paragraphs 5.9 and 7.3 of ANSI N509-1976.

5. The minimum diameter of damper shafts that are 24 inches and under in length shall be 1/2 inch. The minimum diameter of damper shafts that are greater than 24 inches in length through 48 inches in length shall be 3/4 inch.

Paragraph C.4.a

Alternatives are taken to Section 2.3.8 of ERDA 76-21:

1. The control room filtration assembly, due to its size, does not require entry to the housing interior.

   The SLCRS filtration assembly is of gasketless type, therefore handling of the charcoal adsorber trays inside the housing is eliminated. During replacement of HEPA filters, voice communication is sufficient. The filter housing is provided with large doors allowing visual and voice communication between personnel inside and outside of the housing. Therefore, no special communication system need be provided.

2. Decontamination, clothing change, and shower facilities for personnel involved in the filter service are incorporated into the overall plant radiation protection program.

TABLE 1.8-1 (Cont)

3. Due to the large number of duct inlets (registers), it is not feasible to provide prefilters at each register. It should be noted that areas exhausted by filtration systems are not expected to contain significant amounts of dust. Inspection doors are provided at all dampers, reheat coils, and other similar items requiring inspection or maintenance. These doors can serve the functions specified in Section 2.3.8 of ERDA.

Paragraph C.4.d

The ESF atmosphere cleanup systems are operated a minimum of 15 minutes per month.

Paragraph C.5.a

A visual inspection of the ESF atmosphere cleanup system and all associated components is not planned to be made before each in-place airflow distribution test, DOP test, or activated carbon adsorber section leak test, but will be performed after initial installation and on an as-needed basis in accordance with the provisions of Section 5 of ANSI N510-1975.

Paragraph C.5.b

The acceptance testing airflow capacity and distribution test procedure will be developed based on Section 8 of ANSI N510-1980 except for the following alternatives:

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

2. Airflow measurements for the airflow capacity test will be performed in accordance with AABC National Standards for Total System Balance, Fourth Edition, 1982, instead of Section 9 of ACGIH, Industrial Ventilation, as recommended in Paragraph 8.3.1.3 and 8.3.1.4 of ANSI N510-1980. The above alternative will provide consistency with the airflow measurement method for balancing of the plant ventilation systems.

3. The airflow test specified in Paragraph 8.3.1.6 will be performed with the filter bank at 100 percent of design dirty filter pressure drop. The system and equipment instrumentation and surveillance preclude inadvertent operation of the filter banks with the pressure or flow outside of the allowable limits.

4. The test specified in Paragraph 8.3.1.7 of ANSI N510-1980 cannot be performed because the system cannot operate at the flow associated with the pressure drop below clean component pressure drop. The pressure drop of about 50% of the design dirty filters is about equal to the pressure drop of clean components.

TABLE 1.8-1 (Cont)

5. Airflow distribution through prefilters and moisture separator banks is not specified. Therefore, the provisions of Paragraph 8.3.2.3 do not apply.

The surveillance test airflow measurements will be taken following maintenance when required by Technical Specifications using installed instrumentation, when available, or the pitot tube velocity traverse method in accordance with an approved surveillance test procedure.

Paragraph C.5.c

The in-place DOP test of HEPA filters will be performed in accordance with Section 10 of ANSI N510-1980 with the air-aerosol mixing uniformity test in accordance with Section 9 of the above code.

Paragraph C.5.d

The in-place test of the carbon adsorber will be performed in accordance with Section 12 of ANSI N510-1980.

Paragraph C.6.b

Laboratory testing frequency for the activated carbon will meet the recommended minimum test frequency indicated in Table 1 of ANSI N510-1980.

The carbon samples not obtained from test canisters will be obtained with slotted-tube sampler in accordance with ANSI N509-1980.

Table 2

Laboratory tests for methyl iodide penetration for the representative sample will allow a maximum penetration of 3 percent as stipulated in Table 5-1 of ANSI N509-1980.

RG No. 1.53, Rev. 0
UFSAR Reference Section 3.1.1, 7.1.2.6, 15.0.8

APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (JUNE 1973)

Beaver Valley Power Station - Unit 2 meets the intent of Regulatory Guide 1.53 for application of the single failure criterion to protection systems with the following alternatives:

1. As stated in Paragraph C.1, departure from certain provisions may occur.

2. For certain Westinghouse supplied equipment, a fault tree analysis, rather than a failure mode and effects analysis, has been used.

TABLE 1.8-1 (Cont)

3. Final actuation devices are generally capable of periodic in-service testing. However, some devices by necessity can only be tested partially with the unit on line and completely with the unit off line.

4. With regard to Paragraph C.3, single switches supplying signals to redundant channels will have a separation of at least 6 inches or a suitable barrier will be supplied.

5. Compliance with single failure criteria will be verified based on a collective analysis of both the protective system and the final actuation devices or actuators.

RG No. 1.54, Rev. 0
UFSAR Reference Sections 6.1.2, 17.1

QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS (JUNE 1973)

Quality assurance requirements for protective coatings at BVPS-2 meet the intent of this regulatory guide with the following clarification and alternatives:

For the balance-of-plant, ANSI N101.4-1972 requirements for documentation are applied as follows to equipment located in the containment.

For large surface area components, the documents are submitted by the vendors as required by ANSI N101.4-1972. These components include such items as the polar crane, structural steel, concrete, ductwork, uninsulated pipe, neutron shield tank, exteriors of uninsulated tanks and vessels, major equipment supports, and the containment liner.

For manufactured equipment such as pumps, motors, pipe hangers, and supports, the documentation required by ANSI N101.4-1972 is maintained in the seller's files for the complete duration of the contract warranty-guarantee period. A certificate of compliance signed by responsible management personnel is furnished by the seller.

For balance of plant, in lieu of the inspection defined in Section 6.2.4 of ANSI 101.4-1972, inspection is performed in accordance with ANSI N5.12-1974, Section 10, "Inspection for Shop and Field Work."

For nuclear steam supply equipment located in the containment, the following acceptable alternate method is employed.

TABLE 1.8-1 (Cont)

For large surface area components, Westinghouse specifies stringent requirements through the use of a painting specification which includes the use of specific coating systems qualified to ANSI N101.2 and certifications of compliance from the vendors. The vendor's implementation of the specification requirements is monitored during the quality assurance surveillance activities. These components include the reactor coolant system supports, reactor coolant pumps, accumulator tanks, and the manipulator crane.

For intermediate surface area components, Westinghouse employs another specification which also includes the use of coating systems which are qualified to ANSI N101.2. The vendor's compliance with the requirements is also checked during quality assurance surveillance activities. These components include the seismic platform and tie rods, reactor internals lifting rig, head lifting rig, and electrical cabinets.

For both the nuclear steam supply system and the balance-of-plant, Regulatory Guide 1.54 guidelines are not invoked for items such as valve bodies, handwheels, certain electrical cabinets and control panels, loudspeakers, emergency light cases, and small instruments. The total surface area of these items is very small in comparison with the total surface area for which the guidelines are imposed.

The guidelines of this regulatory guide are not applied to routine touch-up work.

No special QA requirements are imposed for the painting of surfaces that will be insulated.

In general, stainless steel and corrosion-resistant alloys are not painted.

<u>RG No. 1.55, Rev. 0</u>
UFSAR Reference Section 3.8.1.6.1

<u>CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (JUNE 1973)</u>

Regulatory Guide 1.55 was, withdrawn (June 1981) and has been superseded by Regulatory Guide 1.136, Rev. 2, June 1981, (Materials, Construction and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the “Code for Concrete Reactor Vessels and Containments”)). However, since a significant portion of Beaver Valley Power Station - Unit 2 (BVPS-2) design and construction was completed prior to the withdrawal of this regulatory guide, concrete placement in Category I structures at BVPS-2 meets or exceeds the intent of Regulatory Guide 1.55 with the following alternatives:

TABLE 1.8-1 (Cont)

1. Shop detail drawings for the reactor containment mat, shell, dome, and internals are checked by the designer. All other reinforcing shop details are checked by engineers at the job site.

2. Constituents and proportions for design mixes to be used for mass concrete are selected to minimize the effects of shrinkage and heat of hydration. The slump used for mass concrete is 3 inches, the slump used in normal concrete is 4 inches, and a slump of 5 inches is allowed in congested areas of heavily reinforced structures and electrical duct lines to permit placing concrete.

3. Curing and protection of freshly deposited concrete conforms to ACI-301, Chapter 12, except that curing compounds are not used on surfaces to which additional concrete is to be bonded, and where wood and/or metal forms are used and remain in place for curing, the forms are kept wet as required to prevent their opening at the joints and drying out of the concrete.

4. The ACI and ASTM specifications are supplemented as necessary with mandatory requirements relating to types and strengths of concrete, minimum concrete densities, proportioning of ingredients, reinforcing steel requirements, joint treatments, and testing agency requirements.

RG No. 1.56, Rev. 1

MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (JULY 1978)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.57, Rev. 0
UFSAR Reference Section 3.8.2

DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (JUNE 1973)

The design limits and loading combinations for the Beaver Valley Power Station - Unit 2 metal primary reactor containment system components meet the intent of Regulatory Guide 1.57 with the following alternatives which apply only to those portions not backed by concrete:

1. The applicable edition of the ASME Boiler and Pressure Vessel Code for affected ASME III components is identified in the ASME Code Baseline Document.

TABLE 1.8-1 (Cont)

2. The primary stresses, based on elastic analysis, meet the following limits in lieu of the limits specified in paragraph C.1.b(2):

   General membrane...1.5 S_m

   Local membrane...the greater of 1.8S_m or 1.5S_m

   Bending plus local membrane...the greater of 1.8S_m or 1.5S_m.

3. The bellows assemblies were designed, fabricated, tested, and installed to the following criteria in lieu of the requirements of paragraphs C.2.a and b. The fuel transfer tube expansion bellows shall be in accordance with Code Case 1330-3 (special ruling) Special Equipment Requirements Section III. An N-2 form in accordance with Code Case 1177 shall be furnished by the bellows vendor. A duplicate bellows shall be required for pressure and fatigue testing by the Contractor in accordance with ASME III, Winter 1974 Addendum, paragraph NE-3365.2(e)(2). The 15-percent maximum convolution pitch in accordance with paragraph NE-3365.2(c) for unreinforced bellows may be exceeded provided the bellows remain within the elastic range.

RG No. 1.58

UFSAR Reference Sections 17.1, 17.2

QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

Application of Regulatory Guide 1.58 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.59, Rev. 2
UFSAR Reference Section 2.4.2.2

DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (AUGUST 1977)

The determination of the design basis flood and the design of Beaver Valley Power Station - Unit 2 follow the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.60, Rev. 1
UFSAR Reference Section 3.7B.1.1

DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (DECEMBER 1973)

The design response spectra for the seismic design of Beaver Valley Power Station - Unit 2 meets the intent of Regulatory Guide 1.60 with the following alternatives:

The horizontal design response spectra used for seismic analyses are shown on Figures 3.7B-1 and 3.7B-2. The spectra shown for the safe shutdown earthquake (SSE) correspond to a maximum ground acceleration of 0.125g and the spectra shown for the operating basis earthquake (OBE) correspond to a maximum ground acceleration of 0.06g. The vertical design response spectra are taken to be two-thirds of the horizontal design response spectra.

These response spectra are based upon the response spectra used for Beaver Valley Power Station - Unit 1 (Figures 2.5-1 and 2.5-2, BVPS-1 UFSAR, October 27, 1972 Docket 5O-33A) and as revised in response to USAEC Regulatory Position 3 of May 25, 1973 (Question 3.15, BVPS-2 PSAR, Amendment 7, July 9, 1973).

RG No. 1.61, Rev. 0
UFSAR Reference Section 3.7.1

DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (OCTOBER 1973)

The determination of the damping values used for the design of Seismic Category I structures, systems, and components at Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide with the following clarifications:

A value of 4 percent of critical is employed for the primary coolant loop system components for the safe shutdown earthquake (SSE). For discussion and justification of this value, refer to WCAP-7921-AR and WCAP-8288.

Values of 1/2 percent and 1 percent of critical are used for the design basis of piping and piping components for the l/2-SSE and SSE, respectively, except as indicated below. In addition; a value of 2 percent of critical is used for reinforced concrete structures for the l/2-SSE. These values are less than the values given in Table 1 of the regulatory guide and are therefore more conservative.

The higher, frequency dependent damping values provided in ASME III Code Case N-411 may be used for stress reconciliation of piping systems and for support optimization.

TABLE 1.8-1 (Cont)

Values of 4 percent and 8 percent of critical are used for cable tray and conduit systems for 1/2 SSE and SSE, respectively. Existing analysis and testing of strut-supported cable tray and conduit systems constructed similar to the support configurations used at BVPS-2 demonstrate that damping values increase with increasing system response. Levels of damping have been measured in excess of 20 percent of critical. Use of 4 percent and 8 percent of critical for cable tray and conduit systems at BVPS-2 is therefore conservative. For discussion of the analysis and testing, refer to:

Elsabee, F., Anagnostis, S., and Djordjevic, W., 1983. Seismic Evaluation of Electrical Raceway Systems. American Society of Mechanical Engineers, Paper 83-PVP-18.

Koss, P., 1979. Seismic Testing of Electrical Cable Support Systems. 48th Annual Convention of Structural Engineers Association of California, Paper.

RG No. 1.62, Rev. 0
UFSAR Reference Section 7.2.2.2, 7.3.2.2.7, 7.2.2.2.3

MANUAL INITIATION OF PROTECTIVE ACTIONS (OCTOBER 1973)

The design of manual initiation of protective actions at Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide with the following clarification:

Manual initiation of the semi-automatic switchover to recirculation following a LOCA is in general compliance with IEEE 279-1971. However, once safety injection is initiated manually, all automatic functions follow except for the opening of the containment sump isolation valves. These valves remain closed until receipt of a low level signal from the refueling water storage tank level instrumentation.

RG No. 1.63, Rev. 2
UFSAR Reference Section 8.3

ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR LIGHT-WATER COOLED NUCLEAR POWER PLANTS (JULY 1978)

Since Beaver Valley Power Station - Unit 2 (BVPS-2) was docketed before August 31, 1978, the methods described in Regulatory Guide 1.63 were not required to be used in the evaluation of the BVPS-2 Construction Permit application. However, the design and construction of the electric penetration assemblies on BVPS-2 follow the guidance of this regulatory guide, except that a ASME Code data report, (ANI) third party inspection and ASME Code stamping of the penetrations are not required as the penetrations are an extension of the containment liner boundary which is not code stamped as discussed in Section 3.8.1.2.1.2.

TABLE 1.8-1 (Cont)

RG No. 1.64, Rev. 2
UFSAR Reference Section 17.2

QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS (JUNE 1976)

Quality Assurance Programs in effect at the time of systems design were followed in the design of BVPS-2 and meet the intent of Regulatory Guide 1.64 with the following clarifications and alternative:

Duquesne Light Company has developed a Quality Assurance Program which conforms to 10 CFR 50, Appendix B. The DLC Quality Assurance Department verifies, through the audit process, that Stone & Webster Engineering Corporation (SWEC) and Westinghouse Electric Corporation are regularly reviewing the status and adequacy of their own QA programs.

The original BVPS-2 Quality Assurance Program was described in Appendix A of the PSAR. This QA Program was initially implemented in accordance with Regulatory Guide 1.64, Rev. 0, dated October 1973. Subsequent Revisions 1 and 2 of the guide required upgrading of the design process, with the most significant changes being independent design verification by individuals not having immediate supervisory responsibility for the individual performing the design. Accordingly, SWEC Engineering Assurance Division issued a change to Engineering Assurance Procedure (EAP) 3.1, "Verification of Nuclear Plant Designs," which requires the following:

All initial issues of key design documents issued after February 8, 1977, shall be subject to independent objective review.

Subsequent revisions to all key design documents, other than calculations, which contain a change in design concept shall be subject to independent objective review. This review shall be limited to that portion of design being changed. Revisions that do not involve a change in design concept shall be reviewed, approved, and issued in accordance with applicable EAPs.

For calculations, the applicable portions of this EAP and independent objective review requirements contained in EAP 5.3 shall be applied to initial issues and all subsequent revisions.

Westinghouse has also updated their Quality Assurance Program to reflect changes in the regulatory process and, in particular, Regulatory Guide 1.64. Changes are described in Westinghouse Topical Reports WCAP-8370, "Westinghouse Quality Assurance Program," and WCAP-7800,

TABLE 1.8-1 (Cont)

"Nuclear Fuel Division Quality Assurance Program." WCAP-8370 revisions applicable to activities for specific time periods are Rev. 7A (June 1, 1975 - Sept. 30, 1977), Rev. 8A (Oct. 1, 1977 - Oct. 31, 1979), and Rev. 9A (Nov. 1, 1979 - present). WCAP-7800, Rev. 5, is applicable to the activities for the entire time period.

Westinghouse has followed the alternative to Regulatory Guide 1.64 that the designer's immediate supervisor may perform design verification in exceptional cases when the supervisor is the only qualified engineer available. For such a case, justification is documented and approved in advance by the supervisor's management.

Application of Regulatory Guide 1.64 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.65, Rev. 0
UFSAR Reference Section 5.3.1.7

MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (OCTOBER 1973)

Materials and inspection for reactor vessel closure studs meet the intent of Regulatory Guide 1.65 with the following alternatives:

1. The use of modified SA-540, Grade B24, as specified in the ASME Code (Code Case 1605) is permitted by Westinghouse but is not listed in this regulatory guide.

   This alternative is based on ASME Code Case 1605 which has been found acceptable to the NRC for application in the construction of components for water-cooled nuclear power plants within the limitation discussed in Regulatory Guide 1.85, which are followed by the Westinghouse practice and one use of Code Case 1605 for reactor vessel closure stud materials is not precluded by this regulatory guide.

2. A maximum ultimate tensile strength of 170,000 psi is not specified by Westinghouse, as recommended by this regulatory guide.

   The ASME Code requirement for toughness for reactor vessel bolting has precluded the regulatory guide's additional recommendation for tensile strength limitation, since to obtain the required toughness levels, the tensile levels are reduced.

TABLE 1.8-1 (Cont)

Westinghouse has specified both 45 ft-lb and 25 mils lateral expansion for control of fracture toughness determined by Charpy-V testing, required by the ASME Code, Section III, and 10 CFR 50, Appendix G, "Fracture Toughness Requirements," (Paragraph IV.A.4). These toughness requirements ensure optimization of the stud bolt material tempering operation with the accompanying reduction of the tensile strength level when compared with previous ASME Code requirements.

The specification of both impact and maximum tensile strength, as stated in the regulatory guide, results in unnecessary hardship in procurement of material without any additional improvement in quality.

The closure stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with the fracture toughness requirement of 10 CFR 50, Appendix G (Paragraph I.C) although higher strength level bolting materials are permitted by the ASME Code.

The primary concern of paragraph C.1.b(l) concerning a maximum tensile strength is to minimize the susceptibility of the bolting material to stress corrosion cracking. Stress corrosion has not been observed in reactor vessel closure stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environment stressed to 75 percent of the yield strength (given in Reference 2 of the regulatory guide). These data are not considered applicable to Westinghouse reactor vessel closure stud bolting because of the specified yield strength differences and a less severe environment; this has been demonstrated by years of satisfactory service experience.

Additional protection against the possibility of incurring corrosion effects is ensured by:

a. Decrease in level of tensile strength compatible with the requirement of fracture toughness as described previously.

b. Design of the reactor vessel studs, nuts, and washers, allowing them to be completely removed during each refueling permitting visual and/or nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the inservice inspection described in Section 5.2.4.

TABLE 1.8-1 (Cont)

c. Design of the reactor vessel studs, nuts, and washers, providing protection against corrosion by allowing them to be completely removed during each refueling and placed in storage racks on the containment operating deck, as required by Westinghouse refueling procedures. The stud holes in the reactor vessel flange are sealed with special plugs before removing the reactor closure. Thus, the bolting materials and stud holes are never exposed to the borated refueling cavity water.

3. Some studs cannot be removed from the vessel for surface examination per paragraph C.4.a. For these studs, it has been determined that a volumetric examination (refer to Section 5.3.3.7) is sufficient.

4. The supplemental surface examination specified in paragraph C.4.a is not required. Improved volumetric (UT) techniques (per Appendix VIII of ASME XI) have rendered surface examinations on the studs unnecessary.

<u>RG No. 1.66, Rev. 0</u>

<u>NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (OCTOBER 1973)</u>

This regulatory guide was withdrawn October 1977.

<u>RG No. 1.67, Rev. 0</u>

<u>INSTALLATION OF OVERPRESSURE PROTECTION DEVICES (OCTOBER 1973)</u>

The design of piping for safety valve and relief valve stations, which have open discharge systems with limited discharge pipes and which have inlet piping that neither contains a water seal nor is subject to slug flow of water upon discharge of the valves, follows the guidance of this regulatory guide.

<u>RG No. 1.68, Rev. 2</u>
UFSAR Reference Section 14.2

<u>INITIAL TEST PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS (AUGUST 1978)</u>

Initial test program testing for BVPS-2 will be conducted in accordance with the intent of Regulatory Guide 1.68 for those structures, systems, and components designated by the General Design Criteria as discussed in UFSAR Section 3.1 and consistent with the extent of compliance noted in UFSAR Section 14.2.12. Compliance with the various referenced regulatory guides is discussed in UFSAR Section 1.8.

TABLE 1.8-1 (Cont)

Initial test program testing for structures, systems, and components that are unrelated to functions designated in the General Design Criteria will be accomplished consistent with their importance to plant reliability and/or safety. For example, the condenser hotwell level control system at BVPS-2 serves no safety-related function, is not power dependent, and is clearly not governed by the regulatory bases cited in Regulatory Guide 1.68 (10CFR50 Appendices A and B). Testing performed on this system will be reasonable and prudent as determined by DLC, but is not appropriate for discussion or commitment in the UFSAR. Other examples of tests which fall into this category include the Normal AC Power Distribution System Test, Containment Instrument Air System Design Verification Test, Computer Input and Printout Data Test, Seismic Instrumentation Test, Computer Operability Test, Extraction Steam System Test, and Rod Cluster Control Assembly Bank and Boron Worth Measurement Test. Also, the plant performance following MSIV closure at power test will now be performed at 30 percent power rather than at 100 percent power. Performance of this test at 100 percent power is unacceptable because of the potential for damage to the plant with an MSIV closure at this power level.

A graded approach to testing will be implemented to provide adequate assurance, considering the importance to safety of the item, that the item will perform satisfactorily while, at the same time, assuring the testing is accomplished in a cost effective manner.

RG No. 1.68.1, Rev. 1

PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS (JANUARY 1977)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.68.2, Rev. 1
UFSAR Reference Section 14.2

INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS (JULY 1978)

The initial start-up test program to demonstrate remote shutdown capability for BVPS-2 follows the guidance of Regulatory Guide 1.68.2 with the following clarification:

Paragraph C.3

The test does not require the turbine generator to be in operation since it would produce an unnecessary perturbation on the electrical distribution system. Steam can be bypassed to the main condenser with the reactor at a moderate power level.

TABLE 1.8-1 (Cont)

During the test, the only action necessary with regard to the turbine is the reactor trip which is initiated at the switchgear and results in a turbine trip signal. Under actual conditions, the trip would occur prior to evacuating the control room and no further actions regarding the turbine are necessary. Therefore, the purposes specified in the Regulatory Guide (demonstration of design, procedures, procedural familiarity and sufficient numbers of personnel) are in no way dependent on actual operation of the turbine.

RG No. 1.68.3, Rev. 0
UFSAR Reference Section 14.2

PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS (APRIL 1982)

Preoperational testing of instrument and control air systems for BVPS-2 will meet the intent of Regulatory Guide 1.68.3 with the following alternatives and clarifications:

Paragraph C.2

As an alternative to verifying the system relief/safety valve settings, credit will be taken for the purchase of specified code valves with documented set-points. Test results were provided with the valves, as specified in the purchase documents.

Paragraph C.3

A full regeneration cycle will be tested only on the station instrument air dryer since it is the only regenerative/desiccant-type dryer installed at BVPS-2. The refrigerant-type dryer installed in containment will be tested in accordance with the intended design attributes. The BVPS-2 position on testing the relief/safety valves is as stated above for Paragraph C.2.

Paragraph C.6

Both station and containment air compressors are oil-free by design. The station air compressors are the dry screw-type, and the containment air compressors are the water ring-type, Therefore, oil content in the product air would be negligible.

Regarding particulate matter in the product air, credit will be taken for vendor documented qualification of filter efficiency.

Paragraph C.8

As an alternative to testing the backup air supplies (accumulators), credit will be taken for the fail-as-is and fail-safe design that precludes the necessity of an excess stored air supply to effect safe shutdown.

TABLE 1.8-1 (Cont)

Paragraph C.9

Normal valve lineups are a prerequisite to the activation of any system or equipment or the performance of tests. The valve lineup completion/signoff sheets are verified by on-shift supervision, and system interface valves are administratively controlled. This precludes the possibility of plant equipment designated by design to be supplied by the instrument and control air system being supplied by another compressed air system having less restrictive air quality requirements.

Paragraph C.10

BVPS-2 has no installed equipment being supplied by the instrument and control air system and having large air requirements. The largest air requirements is less than 1 percent of the capacity of a single compressor.

RG No. 1.69, Rev. 0
UFSAR Reference Section 12.3.2

CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS (DECEMBER 1973)

Regulatory Guide 1.69 invokes the requirements and recommended practices contained in ANSI N101.6-1972, "Concrete Radiation Shields." The design and construction procedures for Beaver Valley Power Station - Unit 2 (BVPS-2) will meet or exceed the guidance of Regulatory Guide 1.69, with the following alternatives:

1. ANSI N101.6-1972 requires that shop drawings be prepared showing details and dimensions of formwork, and then approved by the responsible engineer before fabrication of the formwork may begin. On BVPS-2, it is the responsibility of field personnel to visually check all formwork. Detail drawings are made only for special applications.

2. Finishing and patching of concrete surfaces after removal of forms will conform to Chapter 9 of ACI-301 rather than Section 8.7.5 of ANSI N101.6. It is not necessary to complete this work within 96 hours after the placing of concrete.

3. Section 8.2.4 of ANSI N101.6 lists the maximum vertical drop of concrete as 5 feet. The maximum vertical drop of concrete during placement operations is 6 feet. Experience has indicated that suitable equipment and provisions are given to prevent segregation of the concrete.

TABLE 1.8-1 (Cont)

RG No. 1.70, Rev. 3
UFSAR Reference Section 1.1

STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS (NOVEMBER 1978)

During construction of BVPS-2, Section 50.34 of 10 CFR Part 50 required that each application for a license to operate a nuclear reactor facility include a final safety analysis report (FSAR). Section 50.34 specified in general terms the information to be supplied in these safety analysis reports. The guidance of this regulatory guide was followed in identifying information to be included in the FSAR, on which the NRC could base its findings requisite to the issuance of the license.

RG No. 1.71, Rev. 0
UFSAR Reference Sections 5.2.3, 5.3.1.4, 10.3.6.2

WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (DECEMBER 1973)

Beaver Valley Power Station - Unit 2 meets the intent of Regulatory Guide 1.71 with the following clarifications and alternatives:

Neither the nuclear steam supply system nor the balance of plant supplier requires qualification of welders for areas of limited accessibility for shop fabrication. Current shop practices have produced high quality welds. In addition, the performance of required nondestructive evaluations provides further assurance of acceptable weld quality.

Limited accessibility qualification or requalification, which is in excess of ASME Code, Section III and IX requirements, is believed to be an unduly restrictive requirement for component fabrication, where the welder's physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised.

For field fabrication and erection of ASME III piping, applicable welds are volumetrically inspected to the requirements and standards of ASME III, Class 1. Volumetric nondestructive examination of the production welds made in areas of limited accessibility assures their acceptable quality.

TABLE 1.8-1 (Cont)

RG No. 1.72, Rev. 2

SPRAY POND PIPING MADE FROM FIBERGLASS-REINFORCED THERMOSETTING RESIN (NOVEMBER 1978)

This guide is not applicable to Beaver Valley Power Station - Unit 2 because fiberglass pipe is not used for Quality Assurance Category I applications.

RG No. 1.73, Rev. 0
UFSAR Reference Section 3.11.2

QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (JANUARY 1974)

Qualification tests of electric valve operators installed inside the containment at Beaver Valley Power Station - Unit 2 follow the guidance of this regulatory guide.

RG No. 1.74
UFSAR Reference Section 17.2

QUALITY ASSURANCE TERMS AND DEFINITIONS

Application of Regulatory Guide 1.74 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.75, Rev. 2
UFSAR Reference Sections 7.1.2, 7.5.2, 8.3.1.4, 8.3.2.2

PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS (SEPTEMBER 1978)

Beaver Valley Power Station - Unit 2 (BVPS-2) follows the guidance of Regulatory Guide 1.75 for physical independence of electrical systems with the following clarifications:

1. General

   For the purposes of electrical separation, equivalent protection is provided through enclosure by rigid aluminum conduit, rigid steel conduit, electro-metallic tubing (EMT), flexible aluminum conduit, and flexible steel conduit. Enclosures provided to meet the requirements of BTP CMEB 9.5-1 are considered equivalent to enclosures provided for electrical separation and will have 1 hour or longer fire rating.

TABLE 1.8-1 (Cont)

Metal clad cable, type MC, utilized in low energy, 120 V ac and 125 V dc nominal circuits and in low density applications is considered adequately protected. As such, the minimum separation between these cables and other cables, or raceway (where required) is 1 in. These cables are further described as follows:

a. Type MC cable is a factory assembly of conductors, each individually insulated, enclosed in a metallic sheath of interlocking type, or a smooth or corrugated tube.

b. Largest conductor size is No. 10 AWG.

c. No more than six conductors.

d. No more than three No. 10 AWG conductors with remaining conductors of smaller size.

e. Aluminum sheath cable (a Type MC cable in which the aluminum is continuously welded) and/or interlocked armor cable may have an overall jacket of neoprene or hypalon.

Type SO or SJO cords for lighting drops to fixtures are size 12 AWG or smaller and supply low energy, 120 V ac or 125 V dc, in low density applications. Adequate protection is provided by 1 inch or greater distance to Class lE raceways.

A raised floor panel can be used as a barrier. Panels are 1 in. thick particle board with 22 gauge steel top and bottom sheets, and are fire rated Class A. These panels are considered a barrier when used in a configuration as shown in IEEE Standard 384-1974, Figures 2, 3, or 4.

The Cable Spreading Areas (CSA - Main Control Room, Cable Spreading room, and Computer Room) are protected areas and are not exposed to potential hazards such as high pressure piping, missiles, flammable material, flooding, or wiring that is not flame retardant. They do not contain high energy equipment such as switchgear, transformers, rotating equipment, or potential sources of missiles or pipe whip and are not used for storing flammable materials.

The General Plant Areas (GPA) have been analyzed for potential hazards and as such are categorized as areas where the damage potential is limited to failures of faults internal to the electrical equipment or circuits.

2. Paragraph C.6

Analyses of potential hazards in Section 5.1.1.1.1 of IEEE Standard 384-1974 are accomplished as follows:

a. The high pressure piping and missile analyses are described in Sections 3.6 and 3.5, respectively.

TABLE 1.8-1 (Cont)

b. The fire protection analyses are outlined in Section 9.5.1 and the Fire Protection Evaluation Report (see Table 1.7-3).

c. Flame retardant characteristics of cable systems are described in Section 8.3.3.

d. The building design for external and internal flooding is described in Section 3.4 and 3.8, respectively. The environmental effects on safety-related components due to internal flooding are described in Sections 3.6B, 3.4, and 3.11.

An extensive test program has been conducted at Wyle Labs in Huntsville, Alabama in accordance with Section 5.1.1.2 of IEEE Standard 384-1974, "IEEE Trial-Use Criteria for Separation of Class lE Equipment and Circuits", to establish minimum separation distances for BVPS-2. A test report, "Test Report on Electrical Separation Verification Testing for Duquesne Light Company's Beaver Valley Power Station - Unit 2", including the Wyle test report has been submitted under separate submittal. (See Table 1.7-3). The conclusions of this report are as follows:

a. In the General Plant Areas, the minimum horizontal spatial separation is reduced from 3 feet to 1 foot.

b. Ventilated tray covers and cable bus enclosures are equivalent to solid tray covers.

c. Protective Wraps.

i) Lengths of cable enclosed in a protective wrap of woven silicon dioxide with a minimum of one inch free space protects adjacent cables from electrically induced problems in the cables within the protective wrap.

ii) Length of cable enclosed in a protective wrap of woven silicon dioxide with a minimum of one inch free air space are protected from electrically induced problems in adjacent cables.

iii) Lengths of cable enclosed in a protective wrap of woven silicon dioxide are protected from electrically induced problems in adjacent cables when the adjacent cables are also enclosed in a protective wrap of woven silicon dioxide.

TABLE 1.8-1 (Cont)

iv) The protective wrap of woven silicon dioxide (Trade Name: SIL-TEMP) is normally 54 mils thick and is wrapped longitudinally around cable(s) with a 100 percent overlap (i.e. two thicknesses). The protective wrap of woven silicon dioxide may also be a tape, nominally 125 mils thick, applied helically with a 50 percent lap (half-lapped). In either case an overall 50 percent lap (half-lapped) of 3M No. 69 glass tape is required.

d. In plant areas (both GPA and CSA) where plant arrangement precludes minimum spatial separation between redundant Class 1E circuits or between Class lE and non-Class lE circuits separation is achieved as follows:

i) Tray to Tray (Section 8.3.1.4.2 Items a.1 through a.20)

Vertical - One inch minimum free air space and a single tray cover

Horizontal - One inch minimum free air space and a tray cover top and bottom on one tray.

ii) Tray to Conduit (Section 8.3.1.4.2 Items b.1 through b.18)

Vertical - One inch minimum free air space

Horizontal - One inch minimum free air space

iii) Cable in Air to Cable in Air (Section 8.3.1.4.2 Items d.1 through d.12)

Vertical - One inch minimum free air space and enclosure of one circuit (group) in conduit or a protective wrap.

or

Zero inch minimum free air space and enclosure of both circuits (groups) in protective wraps.

TABLE 1.8-1 (Cont)

	Horizontal -	One inch minimum free air space and enclosure of one circuit (group) in conduit or a protective wrap.
		or
		Zero inch minimum free air space and enclosure of both circuits (groups) in protective wraps.
iv)	Cable in Air to Tray (Section 8.3.1.4.2 Items e.1 through e.12)	
	Vertical -	One inch minimum free air space and a tray cover
		or
		Zero inch minimum free air space with both a tray cover and enclosure of the cable in a protective wrap.
	Horizontal -	One inch minimum free air space and enclosure of one circuit (group) in conduit or a protective wrap.
		or
		Zero inch minimum free air space and enclosure of both circuits (groups) in protective wraps.
v)	Cable in Air to Conduit (Section 8.3.1.4.2 Items f.1 through f.6)	
	Vertical -	One inch minimum free air space
	Horizontal -	One inch minimum free air space

TABLE 1.8-1 (Cont)

e. Inside Control Switchboards and Instrument Cabinets separation between redundant Class lE or Class lE and non-Class lE wire (bundles) is provided by one of the following:

i) A barrier.

ii) A minimum of one inch free air space.

iii) Enclosing each wire (bundles) in a protective wrap of woven silicon dioxide (no overall glass type).

For justification refer to the test report.

3. <u>Paragraph C.7 (Section 4.6 of IEEE Standard 384-1974</u>

Minimum separation between Class lE and non-Class lE circuits is as specified in Sections 5.1.3, 5.1.4, or 5.6.2 of IEEE Standard 384-1974, except as discussed in the position under Paragraph C.6.

4. <u>Paragraph C.9</u>

Cable trays for control and instrumentation cables may be filled above the side rails where the overfill has been limited to a maximum of 1 1/2 in. above the top of the side rail and where solid hat covers with a 2 in. raised flat center section are used to enclose the top of the cable tray.

Cable splices in raceways are prohibited with the exception of the cables which are spliced in trays at penetration types IX and IX-A. The penetration manufacturers standard practice for these penetration types directs that the connection between the penetration conductor and field run cable be made by splicing. Due to the limited space at the penetrations and the size of these large power cables, installation of the termination boxes to enclose the splices is impractical. The cable trays will be enclosed by top and bottom tray covers per Section 8.3.1.4.

The cable splices are made according to the cable manufacturer's procedures. The resulting splice is as good as the cable and in no way degrades the performance of the cable.

5. <u>Paragraph C.10</u>

Class lE cable and raceways shall be marked at intervals not exceeding 15 feet and shall be plainly visible.

TABLE 1.8-1 (Cont)

6. Paragraph C.12

   a. Power cables that supply power to the control, computer, or cable spreading room panels, limited to 120 V ac or 125 V dc, are enclosed in rigid metallic conduit or flexible conduit at the entrance to panels.

   b. Power cables serving facilities in or traversing the control, computer, or cable spreading rooms, limited to 480 V ac, 120 V ac, or 125 V dc, are enclosed in rigid metallic conduit or in flexible conduit at the entrance to panels or equipment.

   c. As noted above in items a and b, all power cables are totally enclosed in rigid metallic conduit or in flexible conduit or in enclosed raceways and are not exposed to free air. Any potential electrical fires caused by fault current in the power cables are not considered to be a credible hazard, since fires resulting from fault current would be contained in the conduit. In addition, these rooms are protected areas and are not subject to external energetic events such as floods, high energy pipe breaks, and missiles.

   d. The loss of the above cables, or the control, computer, or cable spreading rooms due to the design basis event fire, will not compromise the capability to achieve cold shutdown as outlined in Section 9.5.1 and in the Fire Protection Evaluation Report.

   e. The Beaver Valley Power Station - Unit 2 design utilizes a single cable spreading room.

7. Paragraph C.16 (Section 5.6.2 of IEEE-Standard 384-1974)

   The minimum 6 in. separation (or a barrier) applies to spacing between exposed terminals, contacts, and equipment of redundant Class lE circuits or Class lE and non-Class lE circuits for testing and maintenance purposes. Separation between redundant Class lE or Class lE and non-Class lE wire (bundles) is as discussed in the position in Paragraph C.6.

   Separation requirements for Westinghouse NSSS equipment are specifically addressed in Section 7.1.2.2.

TABLE 1.8-1 (Cont)

8. Paragraph C.1 (IEEE Standard 384-1974 - Section 3, ISOLATION DEVICES)

   The use of two independent Class lE overcurrent devices (breakers or fuses) in series, provides electrical separation between Class lE and non-Class lE circuits under the following conditions:

   1. Coordination is provided between each of the two series devices and the main Class 1E feeder breaker.

   2. These devices are included in a surveillance program during normal plant operation.

RG No. 1.76, Rev. 0
UFSAR Reference Section 3.3.2.1

DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (APRIL 1974)

All applicable Beaver Valley Power Station - Unit 2 structures, systems, or components important to safety will be designed to withstand, or will be enclosed in structures which will withstand, the six descriptive parameters given in Table I of Regulatory Guide 1.76 for the Region I location.

RG No. 1.77, Rev. 0
UFSAR Reference Section 15.4.8

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS (MAY 1974)

The assumptions used for evaluating the potential radiological consequences of a control rod ejection accident at Beaver Valley Power Station - Unit 2 are based on Regulatory Guide 1.183. Refer to the position on Regulatory Guide 1.183, later in this table.

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

TABLE 1.8-1 (Cont)

RG No. 1.78, Rev. 0
UFSAR Reference Sections 2.2.3, 6.4, 9.4

ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (JUNE 1974)

Assumptions for evaluating the habitability of the Beaver Valley Power Station - Unit 2 (BVPS-2) portion of the main control room during a postulated hazardous chemical release meet the intent of this regulatory guide with the following clarifications and alternative:

1. Of the various evaluation methods available, BVPS-2 evaluation has been performed by the methodology outlined in NUREG-0570, published in June 1979, which is similar to that presented in Appendix B of Regulatory Guide 1.78 but at a much greater level of detail and refinement.

2. Protection of the control room during a chlorine release is not required due to removal of on-site chlorine storage.

Paragraph C.9

The control room emergency ventilation system is used to pressurize the control room envelope. Periodic control room envelope unfiltered air inleakage tests are performed to confirm that the control room envelope is operable.

RG No. 1.79, Rev. 1
UFSAR Reference Sections 6.3, 14.2

PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (SEPTEMBER 1975)

The program for preoperational testing of the Emergency Core Cooling Systems is performed by following the guidance of this regulatory guide with the following clarification:

Paragraph C.1.c(2) recommends a test of the accumulator isolation valves with full accumulator pressure and zero RCS pressure. R.G. 1.79 clearly states the purpose for the subject test to be "to ensure that inadvertent valve closures do not prevent operation of the core flooding system if required." Administrative controls and technical specification requirements provide assurance that accumulator isolation valves will never be required to change position in performance of a safety function. These controls include power removal upon opening the valve. Alarms which reflash at regular intervals are provided to alert the operator when a valve is not fully open. Technical specifications require periodic verification of valve position, periodic verification of power removal, and plant shutdown if any valve is not opened.

Therefore, the "inadvertent valve closures" which are the source of concern for this test are eliminated unless multiple failures are postulated. These failures would include combinations of the following:

1. Failure to open the valve per technical and operating procedures.

2. Failure of the power removal circuit or failure to remove valve power once the valve is open.

3. Failure of one or more operators to heed the alarm indicating a valve which is not fully open or failure of the alarm.

4. Failure of one or more operators to heed reflash of the alarm.

5. Incorrect verification of valve position on a periodic basis in violation of technical specifications.

Since combinations of these failures would be required to allow an inadvertent accumulator isolation valve closure to occur and not be noticed and corrected, the situation would be highly unlikely. Protecting or testing for multiple failures is over and above philosophy and is not considered necessary. It should be noted that R.G. 1.79 was issued in 1975. This occurred prior to the initiation of the currently used "power removal techniques" found in the industry today.

<u>RG No. 1.80, Rev. 0</u>
UFSAR Reference Section 14.2

<u>PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (JUNE 1974)</u>

This regulatory guide has been withdrawn (April 1982). Regulatory Guide 1.68.3 ("Preoperational Testing of Instrument and Control Air Systems," Rev. 0, April 1982) now applies.

<u>RG No. 1.81, Rev. 1</u>
UFSAR Reference Sections 8.1, 8.3.1, 8.3.2

<u>SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS (JANUARY 1975)</u>

The design of the BVPS-2 onsite emergency and shutdown electric systems follows the guidance of this regulatory guide.

The safe shutdown basis for BVPS-2 is described in the conformance evaluation for Regulatory Guide 1.139 (UFSAR Section 1.8). None of the equipment required for compliance with this safe shutdown commitment for BVPS-2 is powered from electrical systems shared with BVPS-1.

TABLE 1.8-1 (Cont)

RG No. 1.82, Rev. 0
UFSAR Reference Section 6.2.2.2

SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS (JUNE 1974)

Design of the sumps for emergency core cooling and containment spray systems at Beaver Valley Power Station Unit 2 (BVPS-2) meets the intent of this regulatory guide with the following alternatives:

Paragraphs C.1 and C.2 require two separate sumps in containment to supply the redundant halves of the emergency core cooling system (ECCS) and containment spray systems (CSS). Beaver Valley Power Station - Unit 2 provides a single sump with physical separation by perforated plate between the two halves which supply the redundant ECCS and CSS.

Paragraphs C.3 and C.6 require an outer trash rack. The BVPS-2 strainer is constructed of perforated stainless steel plate. The strength of the plate plus the larger size and complex geometry of the strainer eliminate the need for a separate trash rack. Although the new design does include a trash rack, it simply protects the strainer elements as defense in depth, and is not credited with meeting RG 1.82.

Paragraph C.4 requires that the containment floor slope down away from the sump to minimize debris entering the sump. A portion of BVPS-2 containment floor slopes down toward the sump but a raised lip is provided which directs normal floor drainage to the segmented section of the containment sump and will prevent small debris from being swept directly into the sump due to the slope.

Paragraph C.7 requires a reactor coolant velocity through sump screens to be approximately 0.2 ft/sec which will allow fine debris to settle out. The BVPS-2 sump screen velocity is approximately 0.2 ft/sec and is considered satisfactory for the intent of Regulatory Guide 1.82.

RG No. 1.83, Rev. 1
UFSAR Reference Section 5.4.2.2.1

INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (JULY 1975)

The program for inservice inspection of steam generator tubes at Beaver Valley Power Station - Unit 2 meets the intent of this regulatory guide with the following alternative:

Inservice inspection of steam generator tubes is performed in accordance with a steam generator program required by Technical Specifications. Required program provisions, such as for condition monitoring assessments and tube inspections, are specifically described in the Technical Specifications.

TABLE 1.8-1 (Cont)

RG No. 1.84, Rev. 24
UFSAR Reference Section 5.2.1.2

DESIGN AND FABRICATION CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1 (JUNE 1986)

Utilization of code cases for design and fabrication for BVPS-2 follows the guidance of this regulatory guide.

All applicable ASME III Code Cases utilized for BVPS-2 are identified in the ASME Code Baseline Document.

RG No. 1.85, Rev. 24
UFSAR Reference Sections 4.5.1, 4.5.2, 5.2.1.2, 5.4.2.1

MATERIALS CODE CASE ACCEPTABILITY ASME SECTION III DIVISION 1 (JUNE 1986)

Utilization of materials code cases for BVPS-2 follows the guidance of this regulatory guide.

All applicable ASME-III Code Cases utilized for BVPS-2 are identified in the ASME Code Baseline Document.

RG No. 1.86, Rev. 0
UFSAR Reference Section 5.8 (ER-OLS)

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (JUNE 1974)

The guidance of this Regulatory Guide will be followed when termination of the Beaver Valley Power Station - Unit 2 operating license is desired.

RG No. 1.87, Rev. 1

GUIDANCE FOR CONSTRUCTION OF CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS (SUPPLEMENT TO ASME SECTION III CODE CASES 1592, 1593, 1594, 1595, AND 1596) (JUNE 1975)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

TABLE 1.8-1 (Cont)

RG No. 1.88
UFSAR Reference Sections 9.5.1, 17.2

COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

Application of Regulatory Guide 1.88 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.89, Rev. 1
UFSAR Reference Section 3.11

ENVIRONMENTAL QUALIFICATION OF CERTAIN ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS (JUNE 1984)

BVPS-2 electric equipment important to safety is qualified to meet or exceed the intent of IEEE Std. 323-1971 and Category II of NUREG 0588, Rev. 1. For BVPS-2, this includes both safety-related and certain post-accident equipment. When determined possible, the qualification of this equipment will be upgraded to meet the standards set forth in Category I of NUREG-0588, Rev. 1. In accordance with 10CFR50.49(k), BVPS-2 is not required to requalify electric equipment important to safety (except replacement equipment) in accordance with Regulatory Guide 1.89, Rev. 1. Replacement electric equipment important to safety will be qualified in accordance with the guidance provided in Paragraph C.6 of this regulatory guide. Qualification records for replacement electric equipment will meet the intent of Appendix E of this regulatory guide by meeting the applicable requirements of 10 CFR 50.49.

RG No. 1.90, Rev. 1

IN-SERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS (AUGUST 1977)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.91, Rev. 1
UFSAR Reference Section 2.2.3.1.1

EVALUATIONS OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANTS (FEBRUARY 1978)

Evaluation of the consequences of explosions postulated to occur on transportation routes near Beaver Valley Power Station - Unit 2 follows the guidance of Regulatory Guide 1.91.

TABLE 1.8-1 (Cont)

RG No. 1.92, Rev. 1
UFSAR Reference 3.7B.2, 3.7N.2

COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS (FEBRUARY 1976)

The combining of modal responses in the seismic response analysis for Beaver Valley Power Station - Unit 2 follow the guidance of Regulatory Guide 1.92 with the following clarifications:

> If there are no closely spaced modes, the responses are combined by using the square root sum of the squares (SRSS) method. When there are closely spaced modes, the responses are combined by using either the grouping method, as discussed in Paragraph 1.2.1 of the regulatory guide or by using the double sum method, as discussed in Paragraph 1.2.3 of the regulatory guide or by using a method similar to the ten percent method as discussed in Paragraph 1.2.2 but with the inclusion of coupling factors as in the double sum method.
>
> Responses from the orthogonal earthquake inputs are obtained by either the absolute addition of the worst horizontal plus vertical responses or the SRSS combination of the two horizontal direction responses and then the absolute addition of the vertical response.

RG No. 1.93, Rev. 0
UFSAR Reference Sections 8.1, 8.3

AVAILABILITY OF ELECTRIC POWER SOURCES (DECEMBER 1974)

BVPS-2 will follow the guidance of this regulatory guide for operation of the plant in the event of loss of electric power sources.

RG No. 1.94, Rev. 1
UFSAR Reference Sections 3.8.1, 3.8.3, 3.8.4, 17.1, 17.2

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS (APRIL 1976)

Quality assurance requirements for installation, inspection, and testing of structural concrete and structural steel during the construction phase at BVPS-2 meet the intent of this Regulatory Guide with the following clarifications and alternatives:

> The BVPS-2 Quality Assurance Program for structural concrete and steel follows WASH 1283, dated May 24, 1984, and WASH 1309, dated May 10, 1974.

The provisions of Articles CC4334 and CC4330 of the "Code for Concrete Reactor Vessels and Containments" (ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1975 Edition) were not applied to BVPS-2 since the BVPS-2 reactor containment purchase order had been placed prior to the 1975 edition of the Division 2 ASME Code.

Alternatives to ANSI N45.2.5-1974 are taken with respect to frequency of calibration of impact wrenches and bolt projection criteria. Impact and torque wrenches shall be checked at least once daily per shift, and at least one full thread of all bolts shall project beyond the nut of all tightened connections. These criteria comply with the recommendations of the Research Council on Riveted and Bolted Structural Joints.

Application of Regulatory Guide 1.94 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.95, Rev. 1
UFSAR Reference Sections 2.2.1, 2.2.3.1.2, 6.4.4.2, 9.4.1

PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE (JANUARY 1977)

Protection of the Beaver Valley Power Station - Unit 2 control room operators against an accidental chlorine release is not required because there is no on-site chlorine storage.

RG No. 1.96, Rev. 1
DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS (JUNE 1976)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.97, Rev. 2
UFSAR Reference Sections 6.2, 7.4, 7.5, 9.3.2, 11.5, 12.3

INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT (DECEMBER 1980)

The instrumentation provided to monitor Beaver Valley Power Station - Unit 2 during and after postulated accident conditions meets the intent of this regulatory guide with the following clarifications and alternatives:

TABLE 1.8-1 (Cont)

A plant-specific analysis has been conducted to identify the appropriate variables and to establish the appropriate design basis and qualification criteria in order to provide sufficient information to allow the operating staff to ascertain plant conditions during and following an accident. As a result of this analysis, the variables to be monitored have been selected and they have been included in one or more of five classifications (Types A through E) according to their usage and need, and they have been assigned design and qualification Categories 1, 2, or 3. The selection of some of these plant-specific variables and their classifications and categories are different than those of Regulatory Guide 1.97.

Type A variables are not all designated Category 1. Type A variables are designated Category 2 if they are employed in the emergency operating procedures for the sole purpose of providing preferred backup information.

Category 1 instrumentation information is not continuously displayed but it is immediately accessible to the operator. In addition, a historical record of at least one instrumentation channel for each process variable is maintained.

Since Category 3 instrumentation is not part of a safety-related system, it is not qualified so that it will provide information when exposed to a post-accident adverse environment. Category 3 instrumentation is subject to servicing, testing, and calibration programs that are specified to maintain their capability. However, these programs are not in accordance with Regulatory Guide 1.118 which applies to safety-related systems.

<u>RG No. 1.98, Rev. 0</u>
<u>ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFFGAS SYSTEM FAILURE IN A BOILING WATER REACTOR (MARCH 1976)</u>

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

<u>RG No. 1.99, Rev. 1</u>
UFSAR Reference Section 5.3.2.1

<u>EFFECTS OF RESIDUAL ELEMENTS ON PREDICTED RADIATION DAMAGE TO REACTOR VESSEL MATERIALS (APRIL 1977)</u>

The Beaver Valley Power Station - Unit 2 reactor vessel material meets the end-of-life reference criterion of Regulatory Guide 1.99. However, the procedures set forth in this regulatory guide are believed to be overconservative at the higher fluences, and the restriction of the end-of-life transition temperature to 200°F is believed to be technically unnecessary.

TABLE 1.8-1 (Cont)

RG No. 1.100, Rev. 1
UFSAR Reference Section 3.10

SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS (AUGUST 1977)

Since the Beaver Valley Power Station - Unit 2 construction permit application was docketed before October 27, 1972, Class I electrical equipment are seismically qualified in accordance with IEEE Standard 344-1971. In addition, WCAP-8373, "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974," August 1974, and letter NS-CE-692, dated July 10, 1975 from Westinghouse to D. B. Vassello (USNRC) provide justification and cover the test program that demonstrates that the tests meet the intent of IEEE Standard 344-1975 and Regulatory Guide 1.100.

RG No. 1.101, Rev. 2
UFSAR Reference Section 13.3

EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS (OCTOBER 1981)

Beaver Valley Power Station - Unit 2 will follow the guidance of the regulatory guide for emergency planning and preparedness.

RG No. 1.102, Rev. 1
UFSAR Reference Sections 2.4.14, 3.4.1

FLOOD PROTECTION FOR NUCLEAR POWER PLANTS (SEPTEMBER 1976)

The Beaver Valley Power Station - Unit 2 design follows the flood protection guidance of this regulatory guide.

RG No. 1.103, Rev. 1
POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS (OCTOBER 1976)

This regulatory guide, which has been withdrawn (June 1981), was not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.104, Rev. 0

OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS (FEBRUARY 1976)

This regulatory guide was withdrawn August 1979.

TABLE 1.8-1 (Cont)

RG No. 1.105, Rev. 1
UFSAR Reference Section 7.1.2.1.9, 7.5.2.3

INSTRUMENT SETPOINTS (NOVEMBER 1976)

The establishing of instrument setpoints in systems important to safety which monitor variables that have limiting safety system settings follows the guidance of this regulatory guide with the following alternative and clarification:

Paragraph C.5

Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices will not be supplied.

Paragraphs C.1 and C.6

The assumptions used in selecting the setpoint values including instrument inaccuracy and calibration uncertainty and the minimum margin with respect to the technical specification limit will be documented by Westinghouse. Drift rates and their relationships to testing intervals will not be documented by Westinghouse.

RG No. 1.106, Rev. 1

THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES (MARCH 1977)

Thermal overload protection for electric motors on motor-operated valves follows the guidance of this regulatory guide.

RG No. 1.107, Rev. 1
QUALIFICATIONS FOR CEMENT GROUTING FOR PRESTRESSING TENDONS IN CONTAINMENT STRUCTURES (FEBRUARY 1977)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.108, Rev. 1
UFSAR Reference Sections 8.1, 8.3.1, 14.2.12.54, 14.2.12.55

PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS (AUGUST 1977)

The periodic testing of diesel generator units used as on-site electric power systems at BVPS-2 will meet the intent of Regulatory Guide 1.108. All necessary periodic testing, testing frequencies, and reporting requirements are specified in the BVPS-2 Technical Specifications.

TABLE 1.8-1 (Cont)

Paragraph C.2.a(3) suggests a periodic 24-hour, full-load-carrying capability test consisting of 22 hours at the continuous diesel generator rating and 2 hours at the 2-hour rating. Such a test is appropriate only for initial qualification of the diesel generator by the vendor or during preoperational testing to demonstrate adequate design and construction. Since this test imposes more severe service than is required by plant design, periodic performance of this test would only serve to repeatedly demonstrate suitable design or sizing of the units and is beyond what is necessary to demonstrate operability or reliability. Testing on a periodic basis is unnecessary and is inconsistent with the goals of Generic Letter 84-15 by providing conditions which could increase diesel generator degradation and reduce reliability. The diesel generator may occasionally be run to demonstrate its capability to operate for prolonged periods (24 hours or longer) when it is determined that such operation is prudent.

The periodic complete loss of load test specified in Paragraph C.2.a(4) is intended to demonstrate that the diesel overspeed limits are not exceeded upon simultaneous loss of all loads which could be supplied by the diesel. Since it is not expected that all diesel loads would be operated simultaneously, such a test would not be representative of expected operating transients. Additionally, the safety analyses do not take credit for the ability of a diesel to assume its assigned load once it is lost for any reason. Therefore, if a full load rejection is assumed, any resulting consequences to the diesel itself have been considered in the safety analyses with acceptable results. A full loss of load test is performed initially by the vendor to demonstrate acceptable response, but subsequent periodic testing is not required.

Where applicable, during preoperational testing of diesel generator units, acceptable vendor testing may be substituted in whole or in part for operator testing listed in Regulatory Guide 1.108. Sufficient testing will be conducted during the operational life of the plant to demonstrate the operability, reliability, and functional capability of the diesel generator units.

<u>RG No. 1.109, Rev. 1</u>

UFSAR Reference Sections 2.3.5, 11.3, Appendix 11A, 13.3

<u>CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10 CFR PART 50, APPENDIX I (OCTOBER 1977)</u>

Calculation of annual doses to man from routine releases of reactor effluents at Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.110, Rev. 0

COST-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS (MARCH 1976)

Since the Beaver Valley Power Station - Unit 2 (BVPS-2) application for a construction permit was docketed on October 20, 1972, the cost-benefit analysis is optional and will not be developed for BVPS-2.

RG No. 1.111, Rev. 1
UFSAR Reference Section 2.3.5

METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASES FROM LIGHT-WATER-COOLED REACTORS (JULY 1977)

The methods for estimating atmospheric transport and dispersion of gaseous effluents in routine releases from Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide.

RG No. 1.112, Rev. 0-R
UFSAR Reference Sections 2.3.5, 11.1, 11.3.3, 12.2, 15.7

CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT-WATER-COOLED POWER REACTORS (MAY 1977)

The calculation of releases of radioactive materials in gaseous and liquid effluents at Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide.

RG No. 1.113, Rev. 1
UFSAR Reference Section 2.4.12

ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (APRIL 1977)

The mathematical models and dispersion coefficients selected to calculate aquatic dispersion of effluents from accidental and routine reactor releases at Beaver Valley Power Station - Unit 2 follows the guidance of Regulatory Guide 1.113.

RG No. 1.114, Rev. 1
UFSAR Reference Section 13.5.1

GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT (NOVEMBER 1976)

Operators at the controls of Beaver Valley Power Station - Unit 2 will carry out their responsibilities using the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.115, Rev. 1
UFSAR Reference Sections 3.5.1.3, 3.5.2, 9.1.2, 9.5.4, 9.5.8

PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES (JULY 1977)

Protection against low-trajectory turbine missiles at Beaver Valley Power Station - Unit 2 (BVPS-2) will meet the intent of this regulatory guide.

Paragraph C.4 and associated Standard Review Plan NUREG-0800, Section 3.5.1.3, Acceptance Criterion II.1 do not apply to the BVPS-2 plant design arrangement.

As an alternative, the latest technical advances in the design and analysis of Westinghouse low-pressure turbine rotors will be used to provide additional assurance of a low failure probability.

A not-to-be-exceeded ultrasonic inspection interval and inservice inspection program is provided in Section 10.2.3 to insure missile generation probability values well below the 10^{-5} limit. This missile generation probability, in combination with the probability of resulting damage to safety-related equipment, maintains the total probability of turbine missile damage below the regulatory guide limit of 10^{-7} per year.

RG No. 1.116
UFSAR Reference Section 17.2

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

Application of Regulatory Guide 1.116 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.117, Rev. 1
UFSAR Reference Sections 3.2, 3.5, 3.8

TORNADO DESIGN CLASSIFICATION (APRIL 1978)

The method used for identifying those structures, systems, and components at Beaver Valley Power Station Unit 2 that should be designed to withstand the effects of the design basis tornado follows the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.118, Rev. 2
UFSAR Reference Sections 7.5, 8.1, 8.3

PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS (JUNE 1978)

Periodic testing of electric power and protection systems at Beaver Valley Power Station - Unit 2 follows IEEE Standard 338-1977 and the guidance of Regulatory Guide 1.118 with the following clarifications:

> Equipment performing control functions, but activated from protection system sensors is not part of the safety system and is not tested for time response.

Status, annunciating, display, and monitoring functions, except those related to the safety parameter display system, are considered control functions, and reasonability checks, that is, comparison between or among similar such display functions, are made.

Response time testing of nuclear instrumentation system detectors is not required because they exhibit response times that are an insignificant fraction of the total system response time (that is, less than 5 percent).

Protection system sensors are demonstrated to be adequate by vendor testing, analysis, operating experience, or by suitable type testing.

RG No. 1.119, Rev. 0

SURVEILLANCE PROGRAM FOR NEW FUEL ASSEMBLY DESIGNS (JUNE 1976)

This regulatory guide was withdrawn June 1977.

RG No. 1.120, Rev. 1
UFSAR Reference Section 9.5.1

FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS (NOVEMBER 1977)

This regulatory guide provides guidelines acceptable to the NRC for implementing General Design Criterion 3 (GDC 3) of 10CFR50, Appendix A, in the development of a fire protection program. This guide is in an extended comment period and is not yet approved. Branch Technical Position (BTP) CMEB 9.5-1, which is part of the acceptance criteria of Standard Review Plan 9.5.1 (NUREG-0800), dated July 1981, also provides guidelines acceptable to the NRC for implementing GDC 3 in the development of a fire protection program. Regulatory Guide 1.120, Rev. 1, is not listed among the SRP 9.5.1 acceptance criteria. However, the guidelines of this regulatory guide have been incorporated into BTP CMEB 9.5-1. Therefore, the BVPS-2 position on this regulatory guide is covered by the evaluation of the BVPS-2 Fire Protection Program against the guidelines of BTP CMEB 9.5-1.

TABLE 1.8-1 (Cont)

This evaluation is provided in Section 9.5A.2.

RG No. 1.121, Rev. 0
UFSAR Reference Section 5.4.2

BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES (AUGUST 1976)

Bases for plugging degraded steam generator tubes at Beaver Valley Power Station - Unit 2 will meet the intent of this regulatory guide with the following alternatives:

Paragraph C.1

The term "Unacceptable defects" is interpreted to apply to those imperfections resulting from service induced mechanical or chemical degradation of the tube walls which have penetrated to a depth in excess of the plugging limit.

Paragraphs C.2.a(2) and C.2.a(4)

A 200-percent margin of safety will be used based on the following definition of tube failure. Tube failure is defined as plastic deformation of a crack to the extent that the sides of the crack open to a nonparallel, elliptical configuration.

Paragraph C.2.b

In cases where sufficient inspection data exist to establish degradation allowance, the rate used will be an average time-rate determined from the mean of the test data.

Where requirements for minimum wall are markedly different for different areas of the tube bundle, (for example, U-bend area versus straight length in Westinghouse designs), two plugging limits may be established to address the varying requirements in a manner which will not require unnecessary plugging of tubes.

Paragraphs C.3.d(l) and C.3.d(3)

The combined effect of these requirements would be to establish a maximum permissible primary-to-secondary leak rate which may be below the threshold of detection with current methods of measurement. The maximum acceptable length of a through-wall crack has been determined based on secondary pipe break accident loadings which are typically twice the magnitude of normal operating pressure loads. Westinghouse will use a leak rate associated with the crack size determined on the basis of accident loadings.

TABLE 1.8-1 (Cont)

Paragraph C.3.e(6)

Computer code names and references will be supplied rather than the actual codes.

Paragraph C.3.f(1)

A minimum acceptable tube wall thickness (plugging limit) will be established based on structural requirements and consideration of loadings, measurement accuracy, and, where applicable, a degradation allowance as discussed in this position and in accordance with the general intent of this regulatory guide. Analyses to determine the maximum acceptable number of tube failures during a postulated condition are normally done to entirely different bases and criteria and are not within the scope of this regulatory guide.

RG No. 1.122, Rev. 1
UFSAR Reference Section 3.7B.2.5

DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR-SUPPORTED EQUIPMENT OR COMPONENTS (FEBRUARY 1978)

The development of floor design response spectra for seismic design of floor-supported equipment or components at Beaver Valley Power Station - Unit 2 meets the intent of this regulatory guide with the following alternative and clarification:

The response spectra peak resonant period values are broadened +25 percent and -20 percent with vertical sides for use in the design basis.

When ASME III Code Case N-411 damping values are applied for piping analysis peak resonant period values are broadened plus and minus 15 percent with parallel sides. The use of this code case is limited to pipe stress reconciliation and support optimization.

RG No. 1.123, Rev. 1
UFSAR Reference Section 17.2

QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS (JULY 1977)

The original BVPS-2 Quality Assurance Program was described in Appendix A of the PSAR. This Quality Assurance Program followed the guidance of Regulatory Guide 1.123, Rev. 0. During procurement activities, the Quality Assurance program was upgraded to reflect changes in regulatory requirements, including Regulatory Guide 1.123, Rev. 1.

TABLE 1.8-1 (Cont)

Westinghouse topical reports applicable to specific time periods are WCAP-8370, "Westinghouse Quality Assurance Program," Rev. 7A (June 1, 1975 - September 30, 1977), Rev. 8A (October 1, 1977 - October 31, 1979), Rev. 9A (November 1, 1979 - present) and WCAP-7800, "Nuclear Fuel Division Quality Assurance Program Plan," Rev. 5 (applicable to the entire time period).

Application of Regulatory Guide 1.123 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.124, Rev. 1
UFSAR Reference Sections 3.9B.3.4.1, 5.4.14

SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS (JANUARY 1978)

The design limits and appropriate loading combinations associated with normal operation, postulated accidents and specified seismic events for the design of Class 1 linear-type component supports, as defined in Subsection NF of Section III of the ASME code, are not applicable to Beaver Valley Power Station - Unit 2 since the design and placement of the purchase order for these supports precedes the first issue of ASME III, Subsection NF. The only exception is the reactor pressure vessel leveling devices which were procured after July 1974. The design rules for the leveling devices follow ASME III, Subsection NF, as a guide and meet the intent of Regulatory Guide 1.124 with the following alternative:

Paragraph C.8

> Supports for the "active" components that are required only during an emergency or faulted plant condition and that are subjected to loading combinations described in Regulatory Positions C.6 and C.7 should be designed within the design limits described in Regulatory Position C.5 or other justifiable design limits. These limits should be defined by the design specification and stated in the PSAR, such that the function of the supported system will be maintained when they are subjected to the loading combinations described in Regulatory Positions 6 and 7.
>
> The design limits and loading combinations for the component supports are presented in Section 5.4.14.

RG No. 1.125, Rev. 1
UFSAR Reference Section 2.4

PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS (OCTOBER 1978)

The use of physical models for design and operation of hydraulic structures and systems for Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.126, Rev. 1
UFSAR Reference Section 4.2

AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION (MARCH 1978)

The analysis of fuel densification for Beaver Valley Power Station - Unit 2 meets the intent of this regulatory guide with the following clarification:

> The regulatory guide states clearly that, "The model presented in . . . this guide is not intended to supersede NRC-approved vendor models." Beaver Valley Power Station - Unit 2 uses the Westinghouse densification model presented in WCAP-8218 (Proprietary) which has been approved by the USNRC. Westinghouse's reports WCAP-8219 (Non-proprietary) and WCAP-8264, (Customer Version) are companions to the approved versions.

RG No. 1.127, Rev. 1
UFSAR Reference Section 2.4

INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS (MARCH 1978)

BVPS-2 does not use water-control structures, such as those described in this regulatory guide. Water-control structures along the Ohio River are operated and maintained by the responsible governmental authorities, such as the U.S. Army Corps of Engineers.

RG No. 1.128, Rev. 1
UFSAR Reference Sections 8.1.6, 8.3.2

INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (OCTOBER 1978)

The installation design and installation of large lead storage batteries at Beaver Valley Power Station Unit 2 follows IEEE Standard 484-1975 and the guidance of Regulatory Guide 1.128, with the following clarification:

> Hydrogen detectors are not required because proper ventilation has been provided in the battery rooms to ensure a sufficient air exchange rate. In addition, instrumentation exists so that a lack of sufficient flow will be indicated in the control room by an alarm system.

TABLE 1.8-1 (Cont)

RG No. 1.129, Rev. 1
UFSAR Reference Section 8.3.2

MAINTENANCE, TESTING, AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (FEBRUARY 1978)

Maintenance, testing, and replacement of large lead storage batteries at Beaver Valley Power Station Unit 2 will follow the guidance of this regulatory guide with the following alternatives:

Periodic testing will follow the guidance of IEEE Standard 450-1980.

The initial battery service test will be held within 2 years, the customary interval for the first outage; thereafter, the service tests will be held within intervals at 18 months.

RG No. 1.130, Rev. 1
UFSAR Reference Sections 3.9B.3.4.1, 5.4.14

SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS 1 PLATE-AND-SHELL-TYPE COMPONENT SUPPORTS (OCTOBER 1978)

The service limits and loading combinations for Class 1 plate-and-shell type component supports meet the intent of Regulatory Guide 1.130 with the following alternatives:

Paragraph C.3

Service limits for component supports designed by linear elastic analysis are limited by the critical buckling strength. The critical buckling strength is calculated using material properties at temperature. Conservative factors of safety for flat plates and for shells are maintained for each design and service limit. The allowable stress for Service Limit D does not exceed two-thirds of the critical buckling stress.

The safety margins of 3 for shells and 2 for flat plates required by Paragraph C.3 of Regulatory Guide 1.130 are unnecessarily conservative. The margins have been arbitrarily selected by the USNRC; they were influenced by the precedent set in Subsection NE of the ASME Code for shells (F.S. = 3), and by the AISC column factors in the case of plates (F.S. varies from 1.67 to 1.92). The USNRC has authorized a study of generic buckling criteria by a group of consultants. Initial reports indicate they are proposing establishment of critical buckling stresses based on lower bound values of test data from the aerospace industry (thin shells) without specific tolerance requirements on construction.

TABLE 1.8-1 (Cont)

Paragraph C.7

Support for active components that are required only during an emergency or faulted plant condition and that are subjected to loading combinations described in Paragraphs C.5 and C.6 are designed within the limits described in Paragraph C.4 or other justifiable design limits. These limits are defined in UFSAR Section 5.4.14. The function of the supported system is maintained when they are subjected to the loading combinations described in Paragraphs C.5 and C.6.

Paragraph C.7 implies that the lower stress limits associated with Levels A and B Service Limits must be used for any component support that serves a safety-related function during an emergency or faulted (LOCA) plant condition. This would seem to imply that a main coolant pump support, which is a passive element in the main coolant loop, would have to be designed to meet the Level A and B Limits during an emergency or faulted plant condition. This would require that a snubber providing restraint on a residual heat removal line would have to be designed to the Level A and B Service Limits during an emergency or faulted plant condition. If this is the intent, it is a severe departure from current practice. Only active components, such as valves, whose operation is required for safe shutdown during an emergency or faulted conditions have been required to meet design stress limits for these plant conditions. Level C and D Service Limits have been considered adequate to assure pressure boundary integrity under the more severe operating conditions.

RG No. 1.131, Rev. 0
UFSAR Reference Sections 3.11 and 8.3.3

QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (AUGUST 1977)

Qualification tests of electric cables, field splices, and connections for Beaver Valley Power Station - Unit 2 meet the intent of this regulatory guide, with the following alternatives and clarification:

a. All cables installed in trays at BVPS-2, either:

   1. have passed the vertical cable tray gas burner flame test delineated in Section 2.5.4.4 of IEEE-383-1974 or have been determined to be adequate for the hazard in accordance with NRC Generic Letter 86-10 or,

   2. additionally, the flame testing for cables specified after January 1978 was modified in accordance with Reg. Guide 1.131-77 or,

TABLE 1.8-1 (Cont)

3. for non-safety applications, are flame retardant and have passed equivalent industry flame testing as approved by engineering evaluation.

b. Single conductor 600 V control, coaxial, and triaxial cables installed after December 5, 2000, meet the vertical flame testing listed in item a above. Single conductor 600 V control, coaxial, and triaxial cables installed prior to December 5, 2000, are flame resistant and meet the vertical flame test provisions of the appropriate ICEA standards. These cables were not subjected to the gas burner vertical cable tray flame test of IEEE-383-1974, and as such, are restricted to installation in conduit and electrical enclosures (i.e., panels and junction boxes).

RG No. 1.132, Rev. 1
UFSAR Reference Section 2.5.4

SITE INVESTIGATIONS FOR FOUNDATIONS OF NUCLEAR POWER PLANTS (MARCH 1979)

Site investigations for foundations at Beaver Valley Power Station - Unit 2 meet the intent of Regulatory Guide 1.132 with the following alternatives and clarifications:

Paragraph B.5

Surveys of horizontal deviation are made in all boreholes that are used for crosshole seismic tests. This must be done in order to determine the true distance between the energy source and receiver. The suggestion that surveys of vertical deviation be performed appears to be an error.

Paragraph C.1 (Item 5)

Only typical time-distance plots are included on the geologic profiles. A typical time-distance plot is more appropriate since time-distance plots are not usually included on geologic profiles.

Paragraph C.2

Since boring logs do not typically include field and laboratory test results, the results of field permeability tests and borehole logging are presented in tables and figures especially designed for these tests.

Paragraph C.3

Measurement of water or drilling mud levels in borings is not required in all cases since water or drilling mud levels in some materials, such as clays, may give false information about ground-water levels. A sufficient number of observation wells are installed to monitor ground-water levels.

TABLE 1.8-1 (Cont)

Paragraph C.6

Continuous undisturbed samples will be taken in compressible or normally consolidated clays only if required for geotechnical analysis. The need for continuous undisturbed samples is a matter of engineering judgment and is evaluated for each case.

Appendix C

1. Earth Dams, Dikes, Levees, and Embankments

   Boring criteria for rock sites will be developed for each case since penetration to d_{max} at a rock site is excessive.

2. Deep Cuts and Canals

   Boring criteria for rock sites will be developed for each case. Detailed mapping is used for rock sites, supplemented by borings where appropriate.

3. Pipelines

   Boring spacing criteria for rock sites will be developed for each case since the suggested spacing would require an excessive number of borings for rock sites.

   The boring depth criteria for soil and rock sites will be developed for each case since a minimum depth of 5 diameters may be excessive, depending upon the site geology and pipe diameter.

4. Tunnels

   Boring spacing and depth criteria will be developed for each case. The suggested spacing would require an excessive number of borings if applied to rock sites and a minimum depth of 5 times the tunnel diameter may be excessive, depending upon the site geology.

RG No. 1.133, Rev. 1
UFSAR Reference Section 4.4, 14.2

LOOSE-PART DETECTION PROGRAM FOR THE PRIMARY SYSTEM OF LIGHT-WATER-COOLED REACTORS (MAY 1981)

The original design of Beaver Valley Power Station Unit 2 included a loose-part monitoring system that was intended to meet the intent of Regulatory Guide 1.133. Subsequent evaluation has determined that this system was not required. Therefore, various features may be eliminated, modified or maintained as an operating convenience.

TABLE 1.8-1 (Cont)

RG No. 1.134, Rev. 1

MEDICAL EVALUATION OF NUCLEAR POWER PLANT PERSONNEL REQUIRING OPERATOR LICENSES (MARCH 1979)

Medical evaluation of Beaver Valley Power Station Unit 2 (BVPS-2) personnel requiring operator licenses meets the intent of Regulatory Guide 1.134 by following acceptable alternative criteria. These alternative criteria meet the requirement of 10 CFR 55.10, "Contents of Applications," and 10 CFR 55.33, "Renewal of Licenses," that each initial or renewal operator or senior operator license application contain a report of medical examination by a licensed medical practitioner on the form prescribed in 10 CFR 55.60, "Examination Form." BVPS-2 also demonstrates compliance with the requirement of 10 CFR 55.11, "Requirements for the Approval of Applications," and 10 CFR 55.33 that the physical condition and general health of BVPS-2 operator applicants are not such as might cause operational errors endangering public health and safety.

RG No. 1.135, Rev. 0
UFSAR Reference Section 2.4

NORMAL WATER LEVEL AND DISCHARGE AT NUCLEAR POWER PLANTS (SEPTEMBER 1977)

This regulatory guide pertains to those facilities whose construction permit application was docketed after May 1, 1978. Because Beaver Valley Power Station - Unit 2 (BVPS-2) received its docket prior to this date, the regulatory guide is not applicable. However, the methods used for determining normal water levels and surface water discharges for BVPS-2 follows the guidance of this regulatory guide.

RG No. 1.136, Rev. 2
UFSAR Reference Section 3.8.1

MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS (ARTICLES CC-1000, -2000, AND -4000 THROUGH -6000 OF THE "CODE FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS") (JUNE 1981)

Regulatory Guide 1.136, Revision 2, June 1981, is not applicable to the Beaver Valley Power Station - Unit 2 (BVPS-2) design and construction since Revision 2 criteria are to be used in the evaluation of construction permit applications docketed after May 1981. However, materials, construction, and testing of the concrete containment for BVPS-2 meet the intent of Regulatory Guide 1.136, Revision 1, October 1978, with the following alternative:

> In Paragraph C.3, the chloride content for grout should not exceed 200 ppm. Reduced chloride content for grout with pH from 11.6 to 12.0 is unnecessary and will require continuous testing of pH value. All normal Portland cement grout has a pH in excess of 12.0. Therefore, up to 200 ppm of chloride should be acceptable.

TABLE 1.8-1 (Cont)

Regulatory Guide 1.136, Revision 2, refers to adoption of articles cc-1000, -2000, and -4000 through -6000 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1980 edition. Revision 2 of Regulatory Guide 1.136 replaces the following regulatory guides:

Regulatory Guide 1.10, Revision 1 - Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures.

Regulatory Guide 1.15, Revision 1 - Testing of Reinforcing Bars for Category I Concrete Structures.

Regulatory Guide 1.18, Revision 1 - Structural Acceptance Test for Concrete Primary Reactor Containments.

Regulatory Guide 1.19, Revision 1 - Nondestructive Examination of Primary Containment Liner Welds.

Regulatory Guide 1.55, Revision 0 - Concrete Placement in Category I Structures.

Regulatory Guide 1.103, Revision 1 - Post-tensioned Prestressing Systems for Concrete Reactor Vessels and Containments.

The applicability of the preceding regulatory guides to BVPS-2 is as stated in the respective positions on these regulatory guides.

<u>RG No. 1.137, Rev. 1</u>
UFSAR Reference Section 9.5.4

<u>FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS REV. 1 (OCTOBER 1979)</u>

Beaver Valley Power Station - Unit 2 (BVPS-2) meets the intent of the guidelines of Regulatory Guide 1.137 for fuel-oil systems for standby diesel generators with the following alternatives:

ANSI-N195-1976, which is the basis for Regulatory Guide 1.137, recommends the use of duplex strainers to allow continued operation of the system if a strainer becomes plugged. BVPS-2 meets the intent of this recommendation by providing redundant pumps with a simplex wye-type strainer in the discharge line of each pump.

Fuel specifications and periodic tests to verify fuel oil quality will be in accordance with the BVPS-2 technical specifications.

TABLE 1.8-1 (Cont)

RG No. 1.138, Rev. 0
UFSAR Reference Section 2.5.4

LABORATORY INVESTIGATIONS OF SOILS FOR ENGINEERING ANALYSIS AND DESIGN OF NUCLEAR POWER PLANTS (APRIL 1978)

Laboratory investigations of soils and rocks for engineering analysis and design of Beaver Valley Power Station - Unit 2 meets the intent of Regulatory Guide 1.138, with the following alternatives. Most of these were noted in a letter from S. B. Jacobs, Stone & Webster Engineering Corporation, to the Secretary of the USNRC, dated July 6, 1978:

Paragraph C.1.c

Standards used to calibrate laboratory test equipment are of a known higher accuracy than the test equipment, rather than four times more accurate than the working instrument. Calibration of the architect engineer's geotechnical laboratory equipment to higher standards than currently in use is not justified since soils and rocks are material whose properties vary widely within the same deposit or formation. In addition, certain physical properties of soils and rocks are greatly affected by sampling and by preparation for testing in the laboratory. The geotechnical engineer takes these natural variations and sampling/preparation effects into account, and exercises considerable judgment in assigning the material properties to be used in an analysis. (This is different from the case of manufactured materials, where dimensions and physical properties are maintained within a narrow range by the manufacturing process.)

Paragraph C.1.d

Index and classification tests are not performed on all soil and rock samples. Classification of soil and rock samples is performed by visual-manual techniques. Index and classification tests are performed on representative samples to confirm the visual-manual classifications.

Paragraph C.2

Moisture seals are not periodically checked and renewed as needed. Tube samples are inspected for obvious leakage when a tube has been selected for testing. Each sample is examined when it is extruded, and any evidence of drying in the tube is noted on the sample description log. This procedure is sufficient to evaluate whether drying has occurred. Samples that appear to have dried are not tested. Periodic inspection and replacement of moisture seals would not provide any better protection against testing samples whose water content has changed than the method used.

TABLE 1.8-1 (Cont)

The duration of storage is not specifically recorded for each test since this can be calculated from the boring logs, where the sampling date is given, and the laboratory test data sheets, where the date of testing is recorded. Therefore, it is unnecessary to make a separate record of storage time.

Paragraph C.3.a

Classification tests are not performed on every undisturbed test specimen of soil or rock. Visual-manual techniques are the primary means of classifying soil and rock samples. Classification tests are performed on representative samples as necessary to confirm the results of the visual-manual classifications.

Measurements and control tests are not performed to determine whether undisturbed samples have changed during shipment, storage, and handling. Undisturbed samples are visually inspected as they are opened and extruded, to determine whether there has been any change in sample length within the tube or if there are any signs of sample disturbance. Results of these inspections are reported on the sample description log. This procedure and examination of the laboratory test results for possible indications of sample disturbance, is sufficient to determine whether sample disturbance has occurred.

Paragraph C.3.b

A discussion of the validity of test results on scalped materials is not presented as part of the laboratory test data. The laboratory test results indicate which portion of the sample has been scalped, and a competent reviewer of the test results would understand the effect of scalping on the results of specific tests.

Paragraph C.4.a(2)

Results of tests with B-values less than 0.95 may be used for analysis in certain cases. For very stiff or hard clays, it may not be possible to achieve a B-value of 0.95. If results of such a test are used, the B-value is reported and the probable effect on results of the test could be evaluated by a competent reviewer.

Paragraph C.5.a

All soil and rock identifications and descriptions are not documented. The current practice of recording sample descriptions and determining index properties of representative samples is consistent with good engineering practice.

TABLE 1.8-1 (Cont)

Anomalous test data are not reported if they are caused by sample disturbance or equipment malfunction because such data do not reflect the true properties of the material in the field. However, records of such tests are maintained as part of the laboratory records.

Appendix B, Relative Density Test

The frequency of the vibratory table cannot be adjusted, because it depends upon the fixed frequency of the input current.

RG No. 1.139, Rev. 0
UFSAR Reference Sections 3.1.2.34, 5.4.7, Appendix 5A, 10.3, 10.4.9, 14.2

GUIDANCE FOR RESIDUAL HEAT REMOVAL (MAY 1978)

While the safe shutdown basis for Beaver Valley Power Station - Unit 2 (BVPS-2) is hot standby, the cold shutdown capability of the plant meets the intent of Regulatory Guide 1.139 and the Branch Technical Position RSB 5-1 design guidelines for Class 2 plants with the following clarifications:

Following a safe shutdown earthquake, assuming loss of onsite or offsite power and the most limiting single failure, the plant is capable of achieving residual heat removal (RHR) system initiation conditions (approximately 350°F and 400 psia) within 36 hours.

BVPS-2 designers employed an approach to safety grade cold shutdown by upgrading to safety grade, wherever feasible. It is recognized that achievement of cold shutdown, utilizing normally available non-safety grade systems and components, is desirable with postulation of loss-of-offsite power as the initiating event.

In order to provide adequate cooldown within 36 hours to initiate the RHR system, the steam generator power operated relief valves (PORVS) are qualified to full safety-grade design and an adequate supply of auxiliary feedwater is established. The primary plant demineralized water storage tank initially establishes the supply of auxiliary feedwater and is provided with backup from the demineralized water storage tank. Combined, these tanks have adequate storage capacity to remove residual heat for more than three days. Additionally, supply connections from the service water system are available should they be required.

Further upgrades for BVPS-2 include the addition of a safety grade atmospheric dump valve which provides backup to the steam generator PORVS and the upgrading of the pressurizer PORVS to safety grade.

TABLE 1.8-1 (Cont)

The RHR system is provided with two separate and independent systems. This redundancy provides the system with the capability to maintain its heat removal function even with a major single failure. Each RHR system train is isolated from the reactor coolant system (RCS) on the suction side by two motor operated valves (MOVs) in series. The suction isolation valves are independently interlocked to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Each MOV receives power via a separate motor control center and the two valves in series in each train receive their power from a different vital bus. The suction isolation valves are also independently interlocked so that any open valve will automatically close if RCS pressure increases above the RHR system design pressure.

Refer to the positions on Regulatory Guides 1.22 and 1.68 for testing and the position on Regulatory Guide 1.33 for operational procedures.

<u>RG No. 1.140, Rev. 1</u>
UFSAR Reference Sections 9.4, 11.1, 11.3, 14.2.12

<u>DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (OCTOBER 1979)</u>

Design, testing, and maintenance criteria for normal ventilation exhaust system air filtration and adsorption units for Beaver Valley Power Station - Unit 2 meet the intent of Regulatory Guide 1.140 with the following alternatives:

<u>Paragraph C.2.a</u>

1. Where a very small amount of dust is anticipated, prefilters instead of upstream HEPA filters are provided.

2. Heating coils are provided only where the relative humidity is expected to exceed 70 percent.

<u>Paragraph C.2.f</u>

1. The ductwork leak tests are performed in accordance with ANSI N510-1975 with the alternative that ASME Performance Test Code 19.5-1971 will be used in lieu of paragraph 6.3.1. The equipment and equipment arrangement based on the above ASME performance test code will provide test results equivalent to results obtained by equipment specified in paragraph 6.3.1 of ANSI N510-1975.

TABLE 1.8-1 (Cont)

2. The air leakage rate for ductwork will be established based on the air cleaning effectiveness provisions defined in paragraph 4.12.1 of ANSI N509-1980. Because the ductwork carrying contaminated air upstream of filters is under negative pressure and there is no leakage from the ductwork to surrounding space, and because any air leak from the ductwork under positive pressure is on the downstream side of filters (filtered air), the leakage for ductwork upstream of filters and downstream of fans is limited to 5 percent of rated flow at internal design pressure (leakage Class II, Table 4-3 of ANSI N509-1980).

3. For ductwork between the outlet of filters and the inlet of fans, the leakage is limited to 0.5 percent of rated flow at internal design pressure (leakage Class I, Table 4-3 of ANSI N509-1980).

4. The filter housing leak test will be performed in accordance with paragraph 4.12 of ANSI N509-1976 with the alternative that the housing maximum allowable leakage will be as specified in Table 4-3 of ANSI N509-1980. The leakage test procedure will be developed based on Section 6 of ANSI N510-1980.

Paragraph C.3.c

The component mounting frames meet the recommendations of ANSI N509-1976, paragraph 5.6.3, with an alternative for the tolerance provisions. The tolerances for mounting frames are sufficient to pass the bank leak test of paragraphs C.5.c and C.5.d of this Regulatory Guide.

Paragraph C.3.e

1. Stainless steel materials for filter housings are procured to ASTM material specifications such as A167 Type 304 and A267 Type 304 in addition to those listed in paragraph 4.3 of NSIC-65/ERDA 76-21.

2. Welding is performed in accordance with AWS D1.1 or ASME IX; therefore, the workmanship samples recommended in paragraph 7.3 of ANSI N509-1976 are not used to demonstrate welders' qualifications to perform production work.

Paragraph C.3.f

1. Welding procedures, welders, and welding operators are qualified in accordance with designer's welding specifications. These specifications are in general conformance with AWS D1.1 and ASME Section IX, which are recommended in paragraph 7.3 of ANSI N509-1976. Production weld visual acceptance criteria, which are based on AWS D1.1, are used in lieu of workmanship samples recommended in ANSI N509-1976 paragraph 7.3.

TABLE 1.8-1 (Cont)

2. Materials for ductwork are procured to ASTM material specifications, such as A276 Type 304, A500 Gr B, A575 Gr N1020, and A576 Gr 1020, in addition to those listed in paragraph 5.10.6 of ANSI N509-1976.

3. An alternative is taken to paragraph 5.10.3.5 of ANSI N509-1976. While ductwork, as a structure, has a resonant frequency above 25 Hz, this may not be true for the unsupported plate or sheet sections. ANSI N509-1980, which has been issued since the issuance of this Regulatory Guide, has deleted this provision. Tympanic vibration modes of the duct are not considered in design because the loads will be small, and minimum thickness of duct material is 20 gauge. This is more conservative then SMACNA provisions.

<u>Paragraph C.3.g</u>

The qualification of impregnated carbon will be in accordance with Table 5-1 of ANSI N509-1980. The test will be performed as specified in ASTM D3803-1979, paragraph 4.1, Method A.

<u>Paragraph C.3.i</u>

1. The type and application of protective coatings on internal surfaces is controlled in accordance with the designer's specifications, which specify high quality materials and application methods in accordance with the coating manufacturer's instructions. These practices are used in lieu of the recommendations in paragraphs 5.6.4 and 5.7.1 of ANSI N509-1976.

2. An alternative is taken to paragraph 5.7.2 of ANSI N509-1976. Copies of fan rating or test reports are not provided. However, certified fan performance curves are furnished.

3. An alternative is taken to balancing techniques defined in paragraph 5.7.3 of ANSI N509-1976. Displacement criteria following normal industrial practice are used.

4. The fan drawings follow the recommendations of paragraph 5.7.4 of ANSI N509-1976 with the alternative that all information about lubricants and lubrication is contained in operation and instruction manuals.

5. Where AMCA certified ratings are submitted, documentation developed in conjunction with the certification of fans is not furnished in accordance with paragraph 5.7.5 of ANSI N509-1976.

TABLE 1.8-1 (Cont)

Paragraph C.3.1

The following alternatives are taken to paragraph 5.9 of ANSI N509-1976:

1. Isolation dampers used in the contaminated air streams are neither designed nor constructed to the recommendations of ANSI B31.1. The system is designed to ensure that the leakage through the dampers is from the noncontaminated to contaminated portion of the system, and the flow is exhausted through the filters before being released to the atmosphere. Therefore, the uncontrolled release of radioactivity is precluded and the intent of Section 5.9 of ANSI N509-1976 is satisfied.

2. Butterfly valves, where used, are in accordance with the ASME III code.

3. One Class B damper of each size will be tested for leakage rate instead of testing every damper.

4. Welding is controlled in accordance with the designer's specifications using visual acceptance criteria, which are based on AWS D1.1 in lieu of the standards recommended in paragraphs 5.9 and 7.3 of ANSI N509-1976.

5. The minimum diameter of damper shafts that are 24 inches and under in length shall be 1/2 inch. The minimum diameter of damper shafts that are greater than 24 inches in length through 48 inches in length shall be 3/4 inch.

Paragraph C.5.a

A visual inspection of the atmosphere cleanup system and all associated components is not planned to be made before each in-place air flow distribution test, DOP test, or activated carbon adsorber section leak test, but will be performed after initial installation and on an as-needed basis in accordance with the provisions of Section 5 of ANSI N510-1975.

Paragraph C.5.b

The airflow capacity and distribution test procedure will be developed based on Section 8 of ANSI N510-1980 with the following alternatives:

1. To avoid damage to system components, an artificial resistance will be used in lieu of the provision of paragraph 8.3.1.1.

TABLE 1.8-1 (Cont)

2. Airflow measurements for the airflow capacity test will be performed in accordance with AABC National Standards for total system balance, Fourth Edition, 1982, instead of Section 9 of ACGIH, Industrial Ventilation, as specified in paragraph 8.3.1(3) of ANSI N510-1975. The above alternative will provide consistency with the airflow measurement method for balancing of the plant ventilation systems.

3. The airflow test will be performed in accordance with paragraphs 8.3.1.1 and 8.3.1.6 only. The test specified in paragraph 8.3.1.7 of ANSI N510-1980 duplicates the test under paragraph 8.3.1.1, because the actual values for pressure drop for both tests are approximately equal. The airflow test specified in paragraph 8.3.1.6 will be performed with the filter bank at 100 percent of design dirty-pressure drop. The system and equipment instrumentation and surveillance preclude inadvertent operation of the filter banks with the pressure or flow outside of the allowable limits.

4. Airflow distribution through prefilters and moisture separator banks is not specified, therefore, the provisions of paragraph 8.3.2.3 do not apply.

Paragraph C.5.c

Sealant is used in ductwork field repairs, in addition to rivets, by applying the sealant between the ductwork and a riveted patch. This method of repair is allowed only on non-seismic SMACNA Class ductwork.

The in-place DOP Test of HEPA filters will be performed in accordance with Section 10 of ANSI N510-1980. The air-aerosol mixing uniformity test will be performed in accordance with Section 9 of the above code.

In-place DOP testing will be performed on the Filtration Units initially and after an entire or partial bank changeout.

Paragraph C.5.d

The in-place test of the carbon adsorber will be performed in accordance with Section 12 of ANSI N510-1980.

In place testing of the carbon absorber units will be performed initially and after an entire or partial bank changeout.

TABLE 1.8-1 (Cont)

Paragraph C.6.b

Laboratory testing frequency for the activated carbon will coincide with scheduled reactor shutdowns for refueling.

The carbon samples not obtained from test cannisters will be obtained with slotted-tube sampler in accordance with ANSI N509-1980.

RG No. 1.141, Rev. 0
UFSAR Reference Section 6.2.4

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS (APRIL 1978)

Containment isolation provisions for fluid systems at Beaver Valley Power Station - Unit 2 follow the guidance of this regulatory guide.

RG No. 1.142, Rev. 1
UFSAR Reference Sections 3.8.3, 3.8.4

SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENT (OCTOBER 1981)

The design and analysis procedures for the Beaver Valley Power Station - Unit 2 safety-related concrete structures other than the reactor vessel and containment meet the intent of Regulatory Guide 1.142 with the following alternative:

The load combinations considered conform to the requirements of ACI Standard 318-7 and meet the general intent of ACI Standard 349-76. These combinations are shown in Sections 3.8.3.3 and 3.8.4.3.

RG No. 1.143, Rev. 1
UFSAR Reference Sections 3.8.4, 10.4.8, 11.2, 11.3, 11.4, 11.5

DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (OCTOBER 1979)

The Beaver Valley Power Station - Unit 2 design of radioactive waste management systems, structures, and components meets the intent of Regulatory Guide 1.143 with the following clarifications and alternatives:

Paragraph C.1.1.3

The steam generator blowdown flash tank, the steam generator blowdown demineralizers, and the steam generator blowdown demineralizer heat exchangers are located in the turbine building, a nonseismic structure.

TABLE 1.8-1 (Cont)

Leakage from steam generator blowdown system components located in the turbine building is collected in the turbine building sumps and monitored via grab samples prior to release to the environment. The turbine building drain release will be isolated from the storm drainage system and transferred to the liquid waste system when the activity concentration exceeds established site limits.

Paragraph C.1.2.3

The steam generator blowdown evaporator test tanks, the steam generator blowdown demineralizers, and the steam generator blowdown flash tank do not have curbs or elevated thresholds.

Test tanks receive distillate from the steam generator blowdown evaporators and provide facilities for storage and sampling prior to release to the environment. Calculations show that the tanks do not require shielding due to the low activity level of water expected in these tanks. A floor drain is provided in the general area of the test tanks to collect leakage and overflow in the auxiliary building sump for pumping to the liquid waste system. Refer to paragraph C.1.1.3 for the steam generator blowdown flash tank.

Paragraph C.1.2.5

The refueling water storage tank is not provided with a dike or retention pond. Tank overflow is directed to the liquid waste system via piping.

The refueling water storage tank is fabricated in accordance with the requirements of ASME III, Class 2. The atmospheric tank is tested full of water to ensure that there are no leaks, and undergoes 100-percent radiographic examination on the shell. Tank overflow is directed to the liquid waste system via the safeguards building sump piping. The radionuclide concentrations of the refueling water storage tank liquid will be determined following each refueling.

Paragraph C.2.1.1

The gaseous waste system equipment meets or exceeds the codes in Table 1. The gaseous waste delay beds, gaseous waste surge tank, waste gas chiller, overhead gas compressors and piping are designed and fabricated to ASME III. The valves are designed and fabricated to ANSI B31.1. The gaseous waste storage tanks are designed and fabricated to ASME VIII, Division 1.

All equipment manufacturing codes exceed those specified by Table 1 of Regulatory Guide 1.143.

TABLE 1.8-1 (Cont)

Paragraph C.2.1.3

The gaseous waste system piping within the gaseous waste storage tank vault structure is designated non-seismic and has been analyzed for dead-load and thermal stresses only.

Failure of the gaseous waste system piping within the gaseous waste storage tank vault will not adversely affect any other systems or components because the storage tank vault contains only gaseous waste system piping. The Offsite Dose Calculation Manual (ODCM) provides a limit on the quantity of radioactivity contained in each group of gaseous waste storage tanks. The basis of this requirement is to restrict the quantity of radioactivity so any release due to a leak or failure will not exceed the requirements of NRC Branch Technical Position ETSB 11-5, Rev. 0, dated July 1981, of 0.5 Rem at the exclusion area boundary. This accident is described in Section 15.7.1.

The storage tank system is used to store all the gas generated by either BVPS-1 or BVPS-2 when going to a cold shutdown condition. The system uses seven tanks, each of which has individual solenoid operated isolation valves, a pressure indicator, a high pressure alarm, and overpressure protection. The storage tanks were purchased and constructed to ASME VIII requirements and subsequently were seismically mounted in the gaseous waste storage tank vault structure. The seven tanks may all be in service at one time, or be operated in smaller groupings.

Paragraph C.4.3

The piping downstream of the overhead gas compressors is one-half inch.

The discharge piping for the overhead gas compressors is designed to not impose any unnecessary pressure loss in that portion of the gaseous waste system.

The steam generator blowdown heat exchanger condensate flow sensing lines are one-half inch. These sensing lines are sized to accommodate instrument requirements. Standard 300# orifice flanges are tapped with 1/2" ports. There is no concern for clogging the smaller diameter pipe because there are no resins present in this portion of the system.

Paragraph C.5.1.1

Beaver Valley Power Station - Unit 2 (BVPS-2) does not apply the Regulatory Guide 1.60 design ground response spectra. Instead, a BVPS spectrum is applied as described in Section 3.7B.1

TABLE 1.8-1 (Cont)

Paragraph C.5.2.1

Beaver Valley Power Station - Unit 2 does not use Regulatory Guide 1.60 spectra nor Regulatory Guide 1.61 damping values. Instead, refer to Sections 3.7B.1.1 and 3.7B.1.3.

Paragraph C.5.2.2

Beaver Valley Power Station - Unit 2 complies with this section, except that the spectra described in Sections 3.7B.1 and 3.7B.2 are used.

Paragraph C.5.2.3

Beaver Valley Power Station - Unit 2 uses the modal time-history technique to generate floor response spectra. Refer to Section 3.7B.2.

Paragraph C.5.2.4

Beaver Valley Power Station - Unit 2 uses ACI-318-71. This was the code in effect at the time of design. The differences between this code and ACI-318-77 are insignificant.

Paragraph C.6

Quality assurance programs used for the design, manufacture, construction, and inspection of the equipment used in the radwaste management systems are in accordance with a QA Category II classification and the codes and standards specified in the equipment purchase specifications.

RG No. 1.144
UFSAR Reference Section 17.2

AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS

Application of Regulatory Guide 1.144 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.145, Rev. 1
UFSAR Reference Sections 2.3, 13.3

ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENTS AT NUCLEAR POWER PLANTS (NOVEMBER 1982)

Atmospheric dispersion models used for potential accident consequence assessments at Beaver Valley Power Station - Unit 2 follow the guidance of this regulatory guide.

TABLE 1.8-1 (Cont)

RG No. 1.146
UFSAR Reference Section 17.2

QUALIFICATION OF QUALITY ASSURANCE PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS

Application of Regulatory Guide 1.146 during the operations phase of BVPS-2 is described in the FENOC Quality Assurance Program Manual.

RG No. 1.147, Rev. 5
UFSAR Reference Section 6.6

INSERVICE INSPECTION CODE CASE ACCEPTABILITY ASME SECTION XI DIVISION 1 (AUGUST 1986)

Utilization of inservice inspection code cases for BVPS-2 follows the guidance of this regulatory guide.

All applicable ASME XI Code Cases utilized for BVPS-2 are identified in the ASME Code Baseline Document.

RG No. 1.148, Rev. 0
UFSAR Reference Section 3.11

FUNCTIONAL SPECIFICATION FOR ACTIVE VALVE ASSEMBLIES IN SYSTEMS IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS (MARCH 1981)

BVPS-2 meets the intent of Regulatory Guide 1.148 for ensuring the operability of active valve assemblies in systems important to safety with the following clarifications and alternatives:

> Active valves are defined as those relied upon to perform a safety function (as well as a reactor shutdown function) during the transients or events considered in the respective operating condition categories.
>
> Active valves are subjected to analysis and tests to verify operability during a seismic event as described in UFSAR Section 3.9N.3.2 and 3.9B.3.2.2. Active valves also undergo pre-installation operational tests and periodic in-service operational tests to verify and assure their functional ability.
>
> The overall design process includes systems design, valve specifications, and quality assurance procedures. Many of the functional requirements are included in the systems design which dictates the type of valve required for the system application. Inclusion of all requirements in a specification to a valve manufacturer would provide information that is not required to manufacture the valves. Since the requirements are all integrated into the design process, a consolidation of all the requirements into a single document is considered unnecessary.

TABLE 1.8-1 (Cont)

RG No. 1.149, Rev. 0
UFSAR Reference Section 13.2

NUCLEAR POWER PLANT SIMULATORS FOR USE IN OPERATOR TRAINING (APRIL 1981)

The guidance of this regulatory guide is followed for simulators used in operator training for Beaver Valley Power Station - Unit 2.

RG No. 1.150, Rev. 1
UFSAR Reference Section 5.3.1, 5.3.3

ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATIONS (FEBRUARY 1983)

Ultrasonic testing of the reactor vessel welds during preservice and inservice examinations at BVPS-2 will follow the guidance of this regulatory guide as described in the Preservice Inspection Program, which was submitted to the NRC in Letter 2NRC-5-154, dated December 26, 1985, and the Inservice Inspection Program, which is scheduled to be submitted to the NRC in the last quarter of 1986.

RG No. 1.155, June 1988
UFSAR Reference Section 8.3.1.1.19

STATION BLACKOUT

BVPS utilizes the emergency diesel generators at each unit as an alternate AC (AAC) power source to operate systems necessary for coping with a station blackout. The design of the cross-tie circuit between BVPS-1 and BVPS-2 AAC power sources conforms with guidance provided by RG No. 1.155.

RG No. 1.163, September 1995
UFSAR Reference Section 6.2.6

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 180, BVPS Unit 2 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.''

TABLE 1.8-1 (Cont)

RG No. 1.183, Rev. 0

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (JULY 2000)

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. With the exception of the Waste Gas System Rupture, this regulatory guide is utilized to evaluate the potential radiological consequences of all of the BVPS-2 design basis accidents.

RG No. 1.194, Rev. 0

ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL HABITABILITY ASSESSMENTS AT NUCLEAR POWER PLANTS

Control Room X/Q values were calculated using the NRC-sponsored ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and in a manner consistent with RG 1.194.

1.9 STANDARD REVIEW PLAN CONFORMANCE EVALUATION

In accordance with 10 CFR 50.34(g), this section provided an evaluation of Beaver Valley Power Station - Unit 2 (BVPS-2) against the USNRC Standard Review Plan [NUREG-0800], dated July 1981. Therefore, this section 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.

The evaluation of BVPS-2 against each Standard Review Plan (SRP) includes an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for BVPS-2 and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, an evaluation is provided which discusses how the proposed alternative provided an acceptable method of complying with those USNRC rules or regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria.

Table 1.9-1 identifies each SRP against which BVPS-2 was evaluated. For those SRPs against which exceptions were taken, SRP conformance statements, which discuss and justify such alternatives to the SRP acceptance criteria, are provided in Table 1.9-2.

Tables for Section 1.9

(Tables in Section 1.9 are historical)

TABLE 1.9-1

STANDARD REVIEW PLAN CONFORMANCE (HISTORICAL)

CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

		Rev.	Conform	Exception
1.8	Interfaces for Standard Design	1	N/A	

CHAPTER 2: SITE CHARACTERISTICS

		Rev.	Conform	Exception
2.1.1	Site Location and Description	2	X	
2.1.2	Exclusion Area Authority and Control	2	X	
2.1.3	Population Distribution	2	X	
2.2.1 - 2.2.2	Identification of Potential Hazards in Site Vicinity	2	X	
2.2.3	Evaluation of Potential Accidents	2	X	
2.3.1	Regional Climatology	2	X	
2.3.2	Local Meteorology	2		X
2.3.3	Onsite Meteorological Measurements Programs	2	X	
	Appendix A	2	X	
2.3.4	Short-Term Diffusion Estimates	1	X	
2.3.5	Long-Term Diffusion Estimates	2	X	
2.4.1	Hydrologic Description	2	X	
	Appendix A	2	X	
2.4.2	Floods	2	X	
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2	X	
2.4.4	Potential Dam Failures (Seismically-Induced)	2	X	
2.4.5	Probable Maximum Surge and Seiche Flooding	2	X	
2.4.6	Probable Maximum Tsunami Flooding	2	X	
2.4.7	Ice Effects	2	X	
2.4.8	Cooling Water Canals and Reservoirs	2	X	
2.4.9	Channel Diversions	2	X	
2.4.10	Flood Protection Requirements	2	X	
2.4.11	Cooling Water Supply	2		X
2.4.12	Ground Water	2		X
	BTP HGEB 1	2	X	
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	2		X
2.4.14	Technical Specifications and Emergency Operation Requirements	2	X	
2.5.1	Basic Geologic and Seismic Information	2		X
2.5.2	Vibratory Ground Motion	1		X
2.5.3	Surface Faulting	2	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
2.5.4	Stability of Subsurface Materials and Foundations	2		X
2.5.5	Stability of Slopes	2	X	
	CHAPTER 3: DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS			
3.2.1	Seismic Classification	1	X	
3.2.2	System Quality Group Classification	1		X
	Appendix A (formerly BTP RSB 3-1)	1	X	
	Appendix B (formerly BTP RSB 3-2)	1	X	
	Appendix C	0	X	
	Appendix D	0	X	
3.3.1	Wind Loadings	2		X
3.3.2	Tornado Loadings	2	X	
3.4.1	Flood Protection	2	X	
3.4.2	Analysis Procedures	1	X	
3.5.1.1	Internally Generated Missiles (Outside Containment)	2	X	
3.5.1.2	Internally Generated Missiles (Inside Containment)	2	X	
3.5.1.3	Turbine Missiles	2		X
3.5.1.4	Missiles Generated by Natural Phenomena	2	X	
	BTP AAB 3-2	1	X	
	BTP ASB 3-2	2	X	
3.5.1.5	Site Proximity Missiles (Except Aircraft)	1	X	
3.5.1.6	Aircraft Hazards	1	X	
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2	X	
3.5.3	Barrier Design Procedures	1	X	
	Appendix A	0	X	
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1		X
	BTP ASB 3-1	1		X
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1		X
	BTP MEB 3-1	1		X
3.7.1	Seismic Design Parameters	1		X
3.7.2	Seismic System Analysis	1		X
3.7.3	Seismic Subsystem Analysis	1		X
3.7.4	Seismic Instrumentation	1	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
3.8.1	Concrete Containment	1		X
3.8.2	Steel Containment	1		X
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1		X
3.8.4	Other Seismic Category I Structures	1		X
3.8.5	Foundations	1		X
3.9.1	Special Topics for Mechanical Components	2		X
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	2		X
3.9.3	ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures	1		X
	Appendix A	0		X
3.9.4	Control Rod Drive Systems	1	X	
3.9.6	Inservice Testing of Pumps and Valves	2	X	
3.10	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2		X
3.11	Environmental Design of Mechanical and Electrical Equipment	2		X

CHAPTER 4: REACTOR

		Rev.	Conform	Exception
4.2	Fuel System Design	2	X	
4.3	Nuclear Design	2	X	
	BTP CPB 4.3-1	2	X	
4.4	Thermal and Hydraulic Design	1	X	
	Appendix	1	X	
4.5.1	Control Rod Drive Structural Materials	2		X
4.5.2	Reactor Internal and Core Support Materials	2		X
4.6	Functional Design of Control Rod Drive System	1	X	

CHAPTER 5: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

		Rev.	Conform	Exception
5.2.1.1	Compliance with Codes and Standard Rule 10 CFR 50.55a	2		X
5.2.1.2	Applicable Codes Cases	2		X
5.2.2	Overpressure Protection	1		X
	BTP RSB 5-2	0	X	
5.2.3	Reactor Coolant Pressure Boundary Materials	2		X
	BTP MTEB 5-7	2	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	1	X	
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	1		X
5.3.1	Reactor Vessel Materials	1		X
5.3.2	Pressure-Temperature Limits	1	X	
	BTP MTEB 5-2	1	X	
5.3.3	Reactor Vessel Integrity	1	X	
5.4	Preface	1	X	
5.4.1.1	Pump Flywheel Integrity (PWR)	1		X
5.4.2.1	Steam Generator Materials	2		X
	BTP MTEB 5-3	2	X	
5.4.2.2	Steam Generator Tube Inservice Inspection	1	X	
5.4.6	Reactor Core Isolation Cooling System (BWR)	2	N/A	
5.4.7	Residual Heat Removal (RHR) System	2		X
	BTP RSB 5-1	2	X	
5.4.8	Reactor Water Cleanup System (BWR)	2	N/A	
5.4.11	Pressurizer Relief Tank	2	X	
5.4.12	Reactor Coolant System High Point Vents	0		X
	CHAPTER 6: ENGINEERED SAFETY FEATURES			
6.1.1	Engineered Safety Features Materials	2		X
	BTP MTEB 6-1	2	X	
6.1.2	Protective Coating Systems (Paints) - Organic Materials	2	X	
6.2.1	Containment Functional Design	2	X	
6.2.1.1A	PWR Dry Containments, Including Sub-atmospheric Containments	2		X
6.2.1.1B	Ice Condenser Containments	2	N/A	
6.2.1.1C	Pressure-Suppression Type BWR Containments	4	N/A	
	Appendix I	1	N/A	
6.2.1.2	Subcompartment Analysis	2	X	
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	1		X
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2		X
	BTP CSB 6-1	2		X
6.2.2	Containment Heat Removal Systems	3	X	
6.2.3	Secondary Containment Functional Design	2	X	
	BTP CSB 6-3	2	X	
6.2.4	Containment Isolation System	2		X
	BTP CSB 6-4	2	X	
6.2.5	Combustible Gas Control in Containment	2	X	
	Appendix A	2	X	
6.2.6	Containment Leakage Testing	2	X	
6.2.7	Fracture Prevention of Containment Pressure Boundary	5		
6.3	Emergency Core Cooling System	1		X
	BTP RSB 6-1	1		X
6.4	Control Room Habitability Systems.	2		X
	Appendix A	2		X
6.5.1	ESF Atmosphere Cleanup Systems	2		X
6.5.2	Containment Spray as a Fission Product Cleanup System	1	X	
6.5.3	Fission Product Control Systems and Structures	2	X	
6.5.4	Ice Condenser as a Fission Product Cleanup System	2	X	
6.6	Inservice Inspection of Class 2 and 3 Components	1	X	
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	2	N/A	
	CHAPTER 7: INSTRUMENTATION AND CONTROLS			
7.1	Instrumentation and Controls - Introduction	2	X	
	Table 7-1 Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	2	X	
7.2	Reactor Trip System	2		X
	Appendix A	2	X	
7.3	Engineered Safety Features Systems	2		X
	Appendix A	2	X	
7.4	Safe Shutdown Systems	2		X
7.5	Information Systems Important to Safety	2		X

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
7.6	Interlock Systems Important to Safety	2		X
7.7	Control Systems	2	X	
Appendix 7A	Branch Technical Positions (ICSB)	2	X	
	BTP ICSB 1 (DOR)	2	X	
	BTP ICSB 3	2	X	
	BTP ICSB 4 (PSB)	2	X	
	BTP ICSB 5	2	X	
	BTP ICSB 9	2	X	
	BTP ICSB 12	2	X	
	BTP ICSB 13	2	X	
	BTP ICSB 14	2	X	
	BTP ICSB 16	2	X	
	BTP ICSB 19	2	X	
	BTP ICSB 20	2	X	
	BTP ICSB 21	2	X	
	BTP ICSB 22	2	X	
	BTP ICSB 25	2	X	
	BTP ICSB 26	2	X	
Appendix 7B	General Agenda, Station Site Visits	1	X	
	CHAPTER 8: ELECTRIC POWER			
8.1	Electric Power - Introduction	2	X	
	Table 8-1 Acceptance Criteria and Guidelines for Electric Power Systems	2	X	
8.2	Offsite Power System	2	X	
8.3.1	AC Power Systems (Onsite)	2	X	
8.3.2	DC Power Systems (Onsite)	2	X	
Appendix 8A	Branch Technical Positions (PSB)	2		X
	BTP ICSB 2 (PSB)	2	X	
	BTP ICSB 4 (PSB)	2	X	
	BTP ICSB 8 (PSB)	2	X	
	BTP ICSB 11 (PSB)	2	X	
	BTP ICSB 15 (PSB)	2	N/A	
	BTP ICSB 17 (PSB)	2	X	
	BTP ICSB 18 (PSB)	2		X
	BTP ICSB 21 (PSB)	2	X	
	BTP PSB 1	0		X
	BTP PSB 2	0	X	
Appendix 8B	General Agenda, Station Site Visits	0	X	
	CHAPTER 9: AUXILIARY SYSTEMS			
9.1.1	New Fuel Storage	2		X

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
9.1.2	Spent Fuel Storage	3		X
9.1.3	Spent Fuel Pool Cooling and Cleanup System	1	X	
9.1.4	Light Load Handling System (Related to Refueling)	2		X
	BTP ASB 9-1	2		X
9.1.5	Overhead Heavy Load Handling Systems	0		X
9.2.1	Station Service Water System	2	X	
9.2.2	Reactor Auxiliary Cooling Water Systems	1		X
9.2.3	Demineralized Water Makeup Systems	2	X	
9.2.4	Potable and Sanitary Water System	2	X	
9.2.5	Ultimate Heat Sink	2	X	
	BTP ASB 9-2	2	X	
9.2.6	Condensate Storage Facilities	2	X	
9.3.1	Compressed Air System	1	X	
9.3.2	Process and Post-Accident Sampling Systems	2	X	
9.3.3	Equipment and Floor Drainage System	2	X	
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	2		X
9.3.5	Standby Liquid Control System (BWR)	2	N/A	
9.4.1	Control Room Area Ventilation System	2		X
9.4.2	Spent Fuel Pool Area Ventilation System	2		X
9.4.3	Auxiliary and Radwaste Area Ventilation System	2		X
9.4.4	Turbine Area Ventilation System	2	N/A	
9.4.5	Engineered Safety Feature Ventilation System	2		X
9.5.1	Fire Protection Program	3		X
	BTP CMEB 9.5-1	2		X
9.5.2	Communications Systems	2	X	
9.5.3	Lighting Systems	2	X	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2		X
9.5.5	Emergency Diesel Engine Cooling Water System	2	X	
9.5.6	Emergency Diesel Engine Starting System	2		X
9.5.7	Emergency Diesel Engine Lubrication System	2	X	
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
	CHAPTER 10: STEAM AND POWER CONVERSION SYSTEM			
10.2	Turbine Generator	2	X	
10.2.3	Turbine Disk Integrity	1	X	
10.3	Main Steam Supply System	2	X	
10.3.6	Steam and Feedwater System Materials	2	X	
10.4.1	Main Condensers	2	X	
10.4.2	Main Condenser Evacuation System	2	X	
10.4.3	Turbine Gland Sealing System	2		X
10.4.4	Turbine Bypass System	2	X	
10.4.5	Circulating Water System	2	X	
10.4.6	Condensate Cleanup System	2		X
10.4.7	Condensate and Feedwater System	2		X
	BTP ASB 10-2	2		X
10.4.8	Steam Generator Blowdown System (PWR)	2		X
10.4.9	Auxiliary Feedwater System (PWR)	2	X	
	BTP ASB 10-1	2	X	
	CHAPTER 11: RADIOACTIVE WASTE MANAGEMENT			
11.1	Source Terms	2		X
11.2	Liquid Waste Management Systems	2		X
11.3	Gaseous Waste Management Systems	2		X
	BTP ETSB 11-5	0	X	
11.4	Solid Waste Management Systems	2		X
	BTP ETSB 11-3	2	X	
	Appendix 11.4-A	0	X	
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	3		X
	Appendix 11.5-A	1	X	
	CHAPTER 12: RADIATION PROTECTION			
12.1	Assuring That Occupational Radiation Exposures are as Low as is Reasonably Achievable	2		X
12.2	Radiation Sources	2	X	
12.3-12.4	Radiation Protection Design Features	2		X
12.5	Operational Radiation Protection Program	2		X
	CHAPTER 13: CONDUCT OF OPERATIONS			
13.1.1	Management and Technical Support Organization	2	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
1.3.1.2- 13.1.3	Operating Organization	2	X	
13.2.1	Reactor Operating Training	0	X	
13.2.2	Training for Non-Licensed Plant Staff	0		X
13.3	Emergency Planning	2	X	
13.4	Operational Review	2		X
13.5.1	Administration Procedures	0	X	
13.5.2	Operating and Maintenance Procedures	0	X	
13.6	Physical Security	2	X	
	CHAPTER 14: INITIAL TEST PROGRAM			
14.1	Initial Plant Test Programs - PSAR	2	N/A	
14.2	Initial Plant Test Programs - FSAR	2	X	
14.3	Standard Plant Designs, Initial Test Program Final Design Approval (FDA)	1	N/A	
	CHAPTER 15: ACCIDENT ANALYSIS			
15.0	Introduction	2	N/A	
	15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM			
15.1.1-15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	1	X	
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	2		X
	Appendix A	2	X	
	15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM			
15.2.1-15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulatory Failure (Closed)	1	X	
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	1		X
15.2.7	Loss of Normal Feedwater Flow	1		X
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	1		X

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
	15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE			
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions	1	X	
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2		X
	15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES			
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	2	X	
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2	X	
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2	X	
15.4.4-15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	1	X	
15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)	1	X	
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1	X	
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	2		X
	Appendix A	1	X	
15.4.9	Spectrum of Rod Drop Accidents (BWR)	2	N/A	
	Appendix A	2	N/A	
	15.5 INCREASE IN REACTOR COOLANT INVENTORY			
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	1	X	

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
	15.6 DECREASE IN REACTOR COOLANT INVENTORY			
15.6.1	Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve	1		X
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2	X	
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	2	X	
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	2	N/A	
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	2		X
	Appendix A	1		X
	Appendix B	1		X
	Appendix C	2		X
	Appendix D	1		X
	15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT			
15.7.1	Waste Gas System Failure	1	X	
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	1	N/A	
15.7.3	Postulated Radioactive Release Due to Liquid-Containing Tank Failures	2	X	
15.7.4	Radiological Consequences of Fuel Handling Accidents	1	X	
15.7.5	Spent Fuel Cask Drop Accidents	2	X	
	15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM			
15.8	Anticipated Transients Without Scram	1	X	
	Appendix	1	X	
	CHAPTER 16: TECHNICAL SPECIFICATIONS			
16.0	Technical Specifications	1		X
	CHAPTER 17: QUALITY ASSURANCE			
17.1	Quality Assurance During the Design and Construction Phases	2	X	
17.2	Quality Assurance During the Operations Phase	2		X

TABLE 1.9-1 (Cont)

		Rev.	Conform	Exception
	CHAPTER 18: HUMAN FACTORS ENGINEERING			
18.0	Human Factors Engineering/Standard Review Plan Development	0	N/A	X

TABLE 1.9-2
SRP CONFORMANCE STATEMENTS (HISTORICAL)

SRP NO. 2.3.2

TITLE: LOCAL METEOROLOGY DIFFERENCES FROM THE SRP:

5- and 50-mile detailed topographic maps are not provided.

REMARKS:

During a telephone communication between S. A. Vigeant (SWEC) and J. Levine (USNRC) on January 13, 1982, the USNRC indicated that topographic cross sections are sufficient for USNRC review and detailed maps do not necessarily need to be provided.

TABLE 1.9-2 (Cont)

SRP NO. 2.4.11

TITLE: COOLING WATER SUPPLY

DIFFERENCES FROM THE SRP:

A means of assuring that sediment accumulation is adequately controlled is not discussed in FSAR Section 2.4.11.

REMARKS:

Administrative controls assure that sediment is removed periodically.

TABLE 1.9-2 (Cont)

SRP NO. 2.4.12

TITLE: GROUND WATER

DIFFERENCES FROM THE SRP:

The information requested by SRP 2.4.12 is provided in FSAR Section 2.4.13. A discussion of accident effects on ground water is also provided.

REMARKS:

The discussion of ground water presented in FSAR Section 2.4.13 is in accordance with the content requirements of Regulatory Guide 1.70, Rev. 3. The discussion of ground water in FSAR Section 2.4.13 is in full conformance with SRPs 2.4.12 and 2.4.13.

TABLE 1.9-2 (Cont)

SRP NO. 2.4.13

TITLE: ACCIDENTAL RELEASES OF LIQUID EFFLUENTS IN GROUND AND SURFACE WATERS

DIFFERENCES FROM THE SRP:

The information requested by SRP 2.4.13 on dilution and travel times of accidental releases in surface waters is provided in FSAR Section 2.4.12.

REMARKS:

The discussion of accidental releases of liquid effluents in ground and surface waters is in accordance with the content requirements of Regulatory Guide 1.70, Rev. 3. The discussions of accidental releases to ground water in FSAR Section 2.4.13 and accidental releases to surface water in FSAR Section 2.4.12 are in full conformance with SRP 2.4.13.

TABLE 1.9-2 (Cont)

SRP NO. 2.5.1

TITLE: BASIC GEOLOGIC AND SEISMIC INFORMATION

DIFFERENCES FROM THE SRP:

Gravity and aeromagnetic maps are not provided.

REMARKS:

At the present time a gravity map of Pennsylvania that includes the site region is not available.

Aeromagnetic maps are available but show only broad regional trends which provide no additional site-specific information.

TABLE 1.9-2 (Cont)

SRP NO. 2.5.2

TITLE: VIBRATORY GROUND MOTION

DIFFERENCES FROM THE SRP:

SRP 2.5.2 suggests that a calculation to determine the probability of exceeding the accelerating level of the OBE during the life of the plant may be helpful. Regulatory Guide 1.70 requires the calculation. This determination is not provided.

REMARKS:

A determination of the probability of exceeding the OBE during the operating life of the plant was not considered necessary because the OBE acceleration level was taken as one-half the SSE in accordance with Appendix A to 10 CFR 100.

TABLE 1.9-2 (Cont)

SRP NO. 2.5.4

TITLE: STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

DIFFERENCES FROM THE SRP:

The SRP suggests that static and dynamic properties of in situ and backfill materials be supported by laboratory test data.

Laboratory testing was not performed on samples of the in situ sands and gravels within the main plant area since it was not possible to obtain undisturbed samples from the boring investigations due to the gravel content of the soils.

Grain size analyses, in-place density tests, and compaction tests were performed on compacted backfill material as part of the site quality assurance program. However, laboratory determinations of engineering strength properties were not made.

REMARKS:

Since suitable undisturbed soil samples could not be obtained, the static and dynamic properties of the in situ sands and gravels at the site were determined from accepted, conservative empirical correlations of engineering properties to subsurface conditions determined by test borings, geophysical testing, and field testing. For instance, shear modulus was evaluated by the equation presented by Hardin and Drenevich (1972) and was compared with that computed from in situ measurements of shear wave velocity. The permeability of the in situ sands was determined from field permeability tests. Soil unit weights were evaluated from the results of in-place density tests made at the foundation level of the reactor containment. The friction angle was based upon correlations with relative density determined from standard penetration test blow count data obtained from test borings. The susceptibility of the soils at the site was based upon the observed behavior of similar sites during previous earthquakes (DLC 1976) and by using test data from dynamic laboratory tests performed to be more susceptible to liquefaction than the soils at the site (DLC 1972).

Similar methods were used to evaluate the engineering properties of the material used for compacted structural or select granular fill.

Liquefaction analyses are generally performed with the ground-water level assumed coincident with that corresponding to the 25-year flood, which for BVPS-2 is at el 690. With the exception of an area beneath the reactor containment (FSAR Figure 2.5.4-19), structural or select granular fill was placed above el 690 and as such would not be subject to liquefaction. Consequently, tests to evaluate its susceptibility to liquefaction were not performed.

TABLE 1.9-2 (Cont)

REFERENCES:

Duquesne Light Company (DLC) 1972. Preliminary Safety Analysis Report - Beaver Valley Power Station - Unit 2, Section 2.6.5.2. Prepared by SWEC, Boston, Mass.

Duquesne Light Company (DLC) 1976. Report on the Soil Densification Program - Beaver Valley Power Station - Unit 2.

Harding, B.O. and Drenevich, V.P. 1972. Shear Modulus and Damping in Soils, Design Equations. Journal of Soil Mechanics and Foundation Division, Vol. 98, SM-7. ASCE.

TABLE 1.9-2 (Cont)

SRP NO. 3.2.2

TITLE: SYSTEM QUALITY GROUP CLASSIFICATION

DIFFERENCES FROM THE SRP:

BVPS-2 uses the safety classification system provided in ANSI N18.2 as an acceptable alternative to the Regulatory Guide 1.26 quality group classification system. FSAR Section 3.2.2 provides a cross reference between the ANS safety classifications end the quality groups defined in Regulatory Guide 1.26.

REMARKS:

The ANS classification system implemented for BVPS-2 has been endorsed by industry as an alternative accepted by the USNRC for many other plants and meets the requirements of GDC 1 and 10 CFR 50.55a.

TABLE 1.9-2 (Cont)

SRP NO. 3.3.1

TITLE: WIND LOADINGS

DIFFERENCES FROM THE SRP:

ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," was not used as a basis for developing wind loads.

REMARKS:

ASCE Paper No. 3269, "Wind Force on Structures," was used as a basis for developing wind loads; this provides sufficient guidance for satisfying the requirements of GDC 2.

TABLE 1.9-2 (Cont)

SRP NO. 3.5.1.3

TITLE: TURBINE MISSILES

DIFFERENCES FROM THE SRP:

Acceptance Criterion II.1 does not apply to the BVPS-2 plant design arrangement. However, protection against low-trajectory turbine missiles at Beaver Valley Power Station - Unit 2 (BVPS-2) does meet the overall risk acceptance guidelines recommended by this SRP.

REMARKS:

The latest technical advances in the design and analysis of Westinghouse low-pressure turbine rotors is used to provide assurance of turbine disc integrity. Improved disc integrity, overspeed protection, and a comprehensive inspection program provides sufficient missile protection to meet the overall risk acceptance guidelines and maintain acceptably low probabilities.

TABLE 1.9-2 (Cont)

SRP NO. 3.6.1

TITLE: PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

DIFFERENCES FROM THE SRP (BRANCH TECHNICAL POSITION ASB 3-l):

1. BTP ASB-1, paragraph B.1.a(l) suggests that essential equipment be separated from the main steam and feedwater piping and be protected from the effects of a one square foot break of these lines, even though these lines are designed to the "break exclusion" criteria of MEB 3-1. BVPS-2 has determined the resultant environment (pressure, temperature, and humidity) due to the one square foot break and has evaluated the effects on safety-related equipment. Dynamic effects (e.g. pipe whip, jet impingement, and pressurization), structural effects, and flooding have not been considered for the arbitrary one square foot break. Additionally, essential components are not enclosed within structures or compartments as suggested in BTP ASB-1, paragraph B.1.b.

2. Item B.2.c recommends that high energy piping between the containment isolation valves be designed to meet the "break exclusion" criteria of MEB 3-1. On BVPS-2, the only high energy piping in containment penetration areas designed for "break exclusion" are the main steam and feedwater lines as well as the high energy branch lines contained within the main steam valve house. In addition, non-nuclear safety class piping is contained within the break exclusion zone.

3. Item B.2.d requires the break excluded piping outboard of the isolation valves to the first restraint be maintained as the same piping classification as the inboard piping. On BVPS-2, the piping class outboard of the isolation valve is designated non-nuclear safety, QA Category II.

4. Item B.3.b(3) limits the exclusion of single active component failures to certain dual-purpose moderate energy essential systems. However, BVPS-2 employs this criterion to dual-purpose systems in general (i.e., high and moderate energy systems). For example, after a postulated pipe rupture in a CVCS pump discharge line, a single active failure of the idle pump in the redundant train to start is not postulated.

TABLE 1.9-2 (Cont)

REMARKS:

1. These differences are due to the addition of later NRC criteria than were utilized during the plant design stage, i.e.; the arbitrary one square foot break. BVPS-2 has met the intent of BTP-ASB 3-1, B.la. and lb., by evaluating the environmental effects of the arbitrary break on safety-related equipment. This degree of conformance is allowed by BTP-ASB 3-1, paragraph B.lc.

2. The only other high energy piping systems which penetrate the reactor containment are the steam generator blowdown and chemical and volume control systems. It is not necessary to break exclude either of these two systems in containment penetration areas because:

    1. Containment isolation capability is not required to mitigate the effects of a break in either system.

    2. These pipe breaks can all be isolated in consideration of single active failures.

    Refer to Difference and Remark 3 for an explanation of the NNS piping.

3. The piping outboard of the isolation valve, although QA Category II, is analyzed to Seismic Category I criteria from the isolation valve to the terminal anchor (6-way restraint) which is located outside the break exclusion area. The actual break excluded piping is the Class 2 portion from the penetration to the isolation valve and then the non-nuclear safety class high energy piping and branch lines to the main steam valve house wall. Although non-nuclear safety class piping which consists of ANSI B31.1 material has been included in the break exclusion zone, it has been analyzed in accordance with the criteria of MB 3-1, Item B1.b for ASME III Class 2 material. All piping within the extended break exclusion zone is subjected to the augmented ISI criteria of MEB 3-1, Item B1.b.(7). Also refer to differences from SRP Section 3.6.2.

TABLE 1.9-2 (Cont)

4. The BVPS-2 position is in accordance with current industry practice as defined in American National Standard ANSI/ANS 51.1 - 1983 (Appendix B, Section B5 "Application of the Single Failure Criterion"), and ANSI/ANS 58.9 - 1981 (Section 4.5) which makes no distinction between high and moderate energy dual purpose safety-related fluid systems.

TABLE 1.9-2 (Cont)

SRP NO. 3.6.2

TITLE: DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

DIFFERENCES FROM THE SRP:

1. BTP MEB 3-1, paragraph B.1.b, states that breaks need not be postulated in those portions of Class 2 piping from the containment wall to and including the inboard or outboard isolation valve.

   The BVPS-2 criteria state that breaks need not be postulated in those portions of high energy lines designated as break exclusion zones:

   a. Between the containment penetration and the outboard isolation valve. These portions of Class 2 piping are designed to ASME III, Subarticle NE-1120.

   b. Between the containment isolation valves and the Main Steam Valve House (MSVH)/Service Building wall. This includes all MSS high-energy branch lines within the MSVH. All non-nuclear safety (NNS) portions of the break excluded piping are designed in accordance with ANSI B31.1.

2. BTP MEB 3-1, paragraph B.1.c(l), states that breaks should be postulated...at intermediate locations where the maximum stress range as calculated by Eq. (10) and either Eq. (12) or (13) exceeds 2.4 Sm.

   The BVPS-2 criteria states that breaks should be postulated at any intermediate location where the maximum stress range as calculated by Eq. (10) and either Eq. (12) or (13) exceeds 3.0 Sm.

3. SRP 3.6.2 paragraph III.2.C.(4) states that for the jet thrust force T=KpA to be acceptable, K values should not be less than 1.26 for steam, saturated water, or steam-water mixtures, or 2.0 for subcooled, nonflashing water. The BVPS-2 criteria states that values of K or K for jet impingement evaluation may be reduced below the 1.26 and 2.00 limits where justified by consideration of pressure drop due to frictional effects.

4. BTP MB 3-1, paragraph B.1.c(l)(d), states that if stresses and usage factors at two intermediate locations in a Class 1 piping or branch run do not exceed the threshold values for postulating

TABLE 1.9-2 (Cont)

break then the...two highest stress locations based on Equation (10) should be selected.

The BVPS-2 criteria states that intermediate breaks are postulated only at those locations where the stresses and/or usage factors exceed the threshold values specified in BTP MEB 3-1, paragraph B.1.c(l)(b) and B.1.c(l)(c).

5. BTP MEB 3-1, paragraph B.1.c(2)(b)(ii), states that if the stresses in a Class 2 or 3 piping or branch run do not exceed the threshold value for postulating breaks then intermediate breaks are postulated...at not less than two separated locations chosen on the basis of highest stress.

   The BVPS-2 criteria states that intermediate breaks are postulated only at those locations where the stresses exceed the threshold values specified in paragraph B.1.c(2)(b)(ii) for the following system:

   1. Reactor Coolant System (not including the primary loop)
   2. Hydrogenated Drain System
   3. Residual Heat Removal System
   4. Safety Injection System
   5. Main Steam System
   6. Main Feedwater System
   7. Auxiliary Feedwater System
   8. Steam Generator Blowdown System
   9. Chemical and Volume Control System
   10. Gaseous Nitrogen System
   11. Auxiliary Steam System

6. SRP 3.6.2, Section III.3(c) states that "...the jet (from a ruptured high energy pipe) is time and distance invarient...".

   The BVPS-2 criteria allows the use of a 10 diameter limit for certain piping systems in establishing the range of jet effects.

7. SRP 3.6.2 and associated sections of BTP MEB 3-1 specify criteria for postulation of piping ruptures.

TABLE 1.9-2 (Cont)

BVPS-2 has invoked the current GDC-4 and an earlier schedular exemption granted to delete these pipe ruptures from the design basis.

8. SRP 3.6.2 and associated sections of BTP MEB 3-1 specify criteria for postulation of high energy line piping ruptures.

   BVPS-2 has submitted analyses under NUREG-1061 (Vol. 3) qualifying certain ASME III Class 1 and 2 lines for exemption from pipe rupture postulation.

   REMARKS:

1. The BVPS-2 criteria provide an acceptable alternative method of complying with GDC 4 by following the guidance of NUREG-75/087 (November 1975). In addition:

   a. The NNS piping has been seismically designed as described in Section 3.2.1.2.

   b. The NNS piping has been designed to the additional design criteria per paragraphs B.1.b(l)(d), B.1.b(l)(e) and B.1.b(2) - B.1.b(7).

   c. The NNS piping except for the 32" main steam piping is of seamless construction.

   d. The 32" NNS main steam piping is electric fusion welded pipe manufactured to ASTM A-155, KC-70. Class 1 requirements. In addition, the plate used in this manufacturing process was ultrasonically examined in accordance with ASTM A-577-69.

   e. The weld material used in NNS pipe spool fabrication and installation conforms and is certified to ASME Code, Section II requirements.

2. BVPS-2 Code Class 1 piping has been designed to the requirements of the ASME Code, Section III (ASME III), 1971 Edition through the Winter 1972 Addenda. In this version of ASME III (ref. paragraph NB-3653) Equation 10 includes a stress term for a linear through wall temperature gradient referred to as 1ΔT.1. This stress term is not included in Equation 10 specified in versions of ASME III later than the 1977 Edition through Summer 1979 Addenda. Applying 2.4 Sm as the threshold level in determining intermediate break locations is not considered appropriate when stress intensity ranges are calculated using the more conservative version of Equation 10 specified in the ASME III edition and Addenda applicable for BVPS-2 Code Class 1 piping. When using this more conservative version of Equation 10 it is more appropriate to apply 3.0 Sm as the threshold level.

3. American National Standard ANSI/ANS-58.2-1980, titled "Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture" provide an acceptable alternative method for

TABLE 1.9-2 (Cont)

the calculation of jet force (Appendix D) and fluid thrust force (Appendix B) based upon inclusion of the effects of pipe friction.

4. The deviations were requested by DLC in DLC letter no. 2NRC-5-042, docket no. 50-412 and granted in NRC letter dated May 27, 1985 for Docket No. 50-412

5. Same as Remark 4.

6. NUREG/CR-2913, 1983 establishes that piping containing steam or subcooled, flashing water at pressures between 870 and 2,465 psia and with no greater than 70°C subcooling may be evaluated for a jet impingement effective range equal to ten times the nominal pipe diameter.

7. SRP 3.6.2, paragraph II.1 and associated sections of Branch Technical Position MEB 3-1 require conformance with the previous version of General Design Criterion 4. The current version of GDC-4 (Federal Register, Volume 51, No. 70, Friday, April 11, 1986) permits exclusion of such breaks from the design basis "when analysis demonstrates the probability of rupturing such piping is extremely low under design basis conditions." Prior to the formal revision of GDC-4, BVPS-2 was granted a schedular exemption (Reference NRC letter dated October 11, 1985, Knighton to Carey) based upon review of such analyses. This exemption is maintained under the current GDC-4.

8. A combination of conventional methodology (Standard Review Plan Section 3.6.2) along with an alternative leak-before-break (WHIPJET) approach is used for the provision of protection from the mechanistic effects of postulated pipe rupture. WHIPJET demonstrates that the fluid leakage from a postulated defect at the highest stress location (in terms of normal plus safe shutdown earthquake loads) concurrent with minimum material properties in a high energy piping line can be detected well before the rupture of the pipe. WHIPJET is consistent with the procedural recommendations and analytical criteria found in NUREG-1061, Volume 3.

   Portions of the reactor coolant system (RCS), residual heat removal system (RHR), and safety injection system (SIS) have been exempted from consideration of pipe whip and jet impingement effects by the application of WHIPJET. Specific lines covered by this alternative approach are listed by line number in Table 3.6B-4.

TABLE 1.9-2 (Cont)

SRP NO. 3.7.1

TITLE: SEISMIC DESIGN PARAMETERS

DIFFERENCES FROM THE SRP:

1. Design response spectra were not derived using Regulatory Guide 1.60 guidelines.

REMARKS:

1. The design response spectra are unique to BVPS-2 and were derived in accordance with Section 2.5 and question 3.15 of the BVPS-2 PSAR. That is, the shape and the magnitude of values in the design response spectra were specified by the USNRC in question 3.15.

2. Refer to Table 1.8-1 for conformance to Regulatory Guide 1.61.

TABLE 1.9-2 (Cont)

SRP NO. 3.7.2

TITLE: SEISMIC SYSTEM ANALYSIS

DIFFERENCES FROM THE SRP:

1. The BVPS-2 design is based on two-dimensional earthquake effects.

2. BVPS-2 design floor response spectra are developed considering one component of earthquake motion at a time.

3. BVPS-2 design responses are not developed by enveloping the results from a half-space analysis and a finite element analysis for soil-structure interaction.

REMARKS:

1. The guidelines to base designs on three dimensional earthquake effects were not in effect when BVPS-2 was designed and its construction permit issued. The BVPS-2 PSAR commits to using the two dimensional earthquake criteria.

2. Design response spectra are developed based on the response only in the direction of input motion. Coupling between three-directions of input motion is not a BVPS-2 design requirement; BVPS-2 uses a two-direction of input-motion criterion with no consideration of statistical independence between the different input time histories. Coupling between orthogonal directions of response is not considered since the structures are reasonably symmetric, and BVPS-2 is a soil site for which the structural responses are due largely to soil response and not to actual structural distortion responses.

3. The BVPS-2 design is in accordance with the previous SRP (NUREG-75/087, SRP 3.7.2). BVPS-2 was designed using a half-space approach and justified by using a finite element approach to demonstrate the conservatism of the half-space method for soil-structure interaction (refer to FSAR Section 3.7B-2 for further discussion of this issue).

TABLE 1.9-2 (Cont)

SRP NO. 3.7.3

TITLE: SEISMIC SUBSYSTEM ANALYSIS

DIFFERENCES FROM THE SRP:

1. SRP 3.7.3, paragraph II.2.g, suggests that closely spaced modes be combined in accordance with Regulatory Guide 1.92. Westinghouse combines closely spaced modes in accordance with the methods described in FSAR Section 3.7N.2.7.

2. Paragraph II.2.h requires adjacent Non-Category I systems to be analyzed according to the same seismic criteria as applicable to the Category I system, if it is not feasible or practical to isolate the Category I system. Such non-Category I piping, which is not attached to Category I piping systems, is not qualified for the seismic event associated with the upset plant condition.

3. Paragraph II.2.1 addresses buried conduit and tunnels. The FSAR does not address the issue in this section.

REMARKS:

1. The Westinghouse methods for combining closely spaced modes represents an alternative accepted by the USNRC for recent plants and meets the requirements of GDC 2 and Appendix A to 10 CFR 100.

2. For Non-Category I piping which is not attached to Category I piping systems, the Non-Category piping is qualified only for the seismic event associated with the faulted plant condition. This will ensure the structural integrity of the Non-Category I piping, thus precluding damage to Category I components/systems and is in agreement with the BVPS-2 position on Regulatory Guide 1.29.

3. Tunnels are addressed in FSAR Section 3.7.2. Conduit will be addressed later via an FSAR amendment to show compliance.

TABLE 1.9-2 (Cont)

SRP NO. 3.8.1

TITLE: CONCRETE CONTAINMENT

DIFFERENCES FROM THE SRP:

1. BVPS-2 does not use the ACI/ASME (ACI 359) Code as a basis for load equations, design allowables, materials, quality control, or special construction techniques.

2. BVPS-2 takes some alternatives to Regulatory Guides 1.10, 1.19, 1.55, and 1.94.

REMARKS:

1. The BVPS-2 design uses ACI 318-71, with the alternative noted in Section 3.8.1.2.1.1, as a basis for the concrete portions of the containment and ASME Section III, Division I (1971 edition with Addenda through winter 1972) as a guide for liner portions instead of the ACI/ASME (ACI 359) Code. The ACI/ASME Code was not in effect at the time BVPS-2 was designed and its construction permit issued.

2. Refer to Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 3.8.2

TITLE: STEEL CONTAINMENT

DIFFERENCES FROM THE SRP:

The structural acceptance criteria used for BVPS-2 differ from that in this SRP and Regulatory Guide 1.57.

REMARKS:

The structural acceptance criteria (that is, design limits and loading combinations) used for BVPS-2 is consistent with NUREG-75/087, which satisfies the requirements of the GDC. Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.57.

TABLE 1.9-2 (Cont)

SRP NO. 3.8.3

TITLE: CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

DIFFERENCES FROM THE SRP:

1. The ACI-349 Code was not used for BVPS-2.

2. The ASME Boiler and Pressure Vessel Code, Section III, was not used for BVPS-2.

3. BVPS-2 takes exception to one of the requirements of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

4. BVPS-2 takes some alternatives to Regulatory Guides 1.10, 1.15, 1.55, 1.94 and 1.142.

REMARKS:

1. The BVPS-2 design uses ACI 318-71, with the alternative noted in Section 3.8.1.2.1.1, as a basis for the concrete internal structures; the ACI 349 Code was not in effect at the time BVPS-2 was designed and its construction permit issued.

2. The ASME Boiler and Pressure Vessel Code is not applicable to the internal structures of BVPS-2.

3. Refer to Section 3.8.1.2.1.3 for this exception.

4. Refer to Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 3.8.4

TITLE: OTHER SEISMIC CATEGORY I STRUCTURES

DIFFERENCES FROM THE SRP:

1. BVPS-2 takes some alternatives to Regulatory Guides 1.10, 1.55, 1.69, 1.94, 1.115, 1.142, and 1.143.

2. BVPS-2 does not use the ACI 349 Code as a basis for reinforced concrete design and analysis procedures.

3. BVPS-2 takes exception to one of the requirements of the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.

4. Loads, load combinations, and structural acceptance criteria are not in complete agreement with SRP 3.8.4.

5. Appendices A, B, C, and D to SRP 3.8.4 are not addressed in Section 3.8.4.

REMARKS:

1. Refer to Section 1.8 for BVPS-2 positions on regulatory guides.

2. Reinforced concrete design and analysis procedures conform to the ACI 318-71 Code, with the alternative noted in Section 3.8.1.2.1.1. The ACI 349 Code was not issued at the time the BVPS-2 Construction Permit was issued nor by the time the majority of the reinforced concrete was designed.

3. Refer to Section 3.8.1.2.1.3 for this exception.

4. Certain load combinations have not been postulated. These combinations would not govern the design of the structures.

5. Appendix A: BVPS-2 has no safety-related concrete masonry walls.

   Appendix B: The Appendix, "Structural Design Audits," contains no specific acceptance criteria.

   Appendix C: Items comprising "The Design Report" were discussed during the NRC Structural Design Audit.

   Appendix D: The spent fuel racks are addressed in Section 9.1.2.

TABLE 1.9-2 (Cont)

SRP NO. 3.8.5

TITLE: FOUNDATIONS

DIFFERENCES FROM THE SRP:

1. BVPS-2 takes some alternatives to Regulatory Guides 1.10, 1.19, 1.55, 1.94, 1.136, and 1.142.

2. BVPS-2 does not use the ACI 349 Code as a basis for reinforced concrete design and analysis procedures.

3. Overturning moments do not consider the combination of the three components of the earthquake.

REMARKS:

1. Refer to Section 1.8 for BVPS-2 positions on regulatory guides.

2. Reinforced concrete design and analysis procedures conform to the ACI 318-71 Code, with the alternative noted in Section 3.8.1.2.1.1. The ACI 349 Code was not issued at the time the BVPS-2 construction permit was issued nor by the time the majority of the reinforced concrete was designed.

3. BVPS-2 is committed to two-dimension earthquake criteria. Refer to Section 3.7B.2 for a description of the analytical techniques used to determine earthquake forces.

TABLE 1.9-2 (Cont)

SRP NO. 3.9.1

TITLE: SPECIAL TOPICS FOR MECHANICAL COMPONENTS

DIFFERENCES FROM THE SRP:

SRP 3.9.1, paragraph II.2, requests a considerable amount of information for all the computer codes used in the design and analysis of seismic Category I components. Westinghouse only provides a brief description of the computer codes used by Westinghouse for component design and analysis in FSAR Section 3.9N.1.2. Additional information required by this SRP for the computer codes referred to in FSAR Section 3.9N.1.2 is provided by reference to WCAP-8252 and -8929. Computer codes used by Westinghouse vendors are not included in the FSAR.

REMARKS:

The information requested by the SRP for referenced Westinghouse computer codes is provided in WCAP-8252 and -8929 and meets the relevant requirements of GDC 1 and Appendix B to 10 CFR 50. Both of these documents have been submitted to the USNRC for review.

Vendor computer codes are not included in the FSAR because of the large number of codes used and the proprietary nature of this vendor information. Westinghouse ensures the acceptability of vendor computer codes through quality assurance audits at vendor facilities and the technical review of various design documents submitted by vendors to Westinghouse.

TABLE 1.9-2 (Cont)

SRP NO. 3.9.2

TITLE: DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

DIFFERENCES FROM THE SRP:

1. SRP 3.9N.2, paragraph II.2.e, defines criteria for combining closely spaced modes. The Westinghouse method for combining closely spaced modes is provided in FSAR Section 3.7N.2.7.

2. Differences in methods exist of combining individual orthogonal stresses resulting from the dynamic responses of the equipment and the modeling techniques employed in the seismic qualification of mechanical and electrical equipment.

3. BVPS-2 employs alternatives to Regulatory Guides 1.20, 1.61. 1.68, and 1.92.

REMARKS:

1. Refer to the justification provided for SRP 3.7.3.

2. FSAR Section 3.9B.2.2 was written to a previous revision of the SRP. Based on previous successfully submitted SARs the general intent should be satisfied by the present 3.9.2.2 and other referenced sections. Technically acceptable alternate methods are identified in FSAR Section 3.7B.3.

3. Refer to Section 1.8 of the FSAR for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 3.9.3

TITLE: ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

DIFFERENCES FROM THE SRP:

1. Design of ASME Code Class 1, 2, and 3 Components do not include specific provisions for addressing functional capability as outlined in SRP 3.9.3.

2. SRP 3.9.3, Appendix A.1.3.3, defines the design basis pipe break (DBPB) as an emergency condition. For ASME Code Class 1, 2, and 3 components and component supports, the DBPB is a faulted condition. (Refer to the loading combination tables in Section 3.9.)

3. SRP 3.9.3, paragraph II.3, requests deformation limits to be included in the FSAR for component supports when the support affects the operability of the component. This requirement would only apply to active pump supports. Active valves are supported in the piping system. No information on deformation limits is provided in the FSAR for supports of active pumps.

4. BVPS-2 provides acceptable alternatives to various portions of Regulatory Guides 1.48, 1.67, 1.124, and 1.130.

5. A fatigue evaluation of shock arrestors is requested by SRP 3.9.3, paragraph II.3.b.l unless it can be demonstrated that the number of load cycles which the snubber will experience during normal plant operating conditions is small (less than 2500), or motion during normal plant operating conditions does not exceed snubber dead band. This fatigue evaluation is not done.

6. SRP 3.9.3, paragraph II.3.b.2 discusses the need to consider snubber end fitting clearance and lost motion when calculating snubber reaction loads and stress which are based on a linear analysis of the system or component. Piping analysis and component analysis do not consider snubber end fitting clearance and lost motion.

7. BVPS-2 does not use service stress limits A, B, C, and D as described in Appendix A to SRP 3.9.3.

REMARKS:

1. Conformance with ASME III requirements and experience with components experiencing significant earthquakes and fluid transient loads (NUREGs/CR-2137 and CR-1665) provides adequate assurance of functional capability.

TABLE 1.9-2 (Cont)

2. The DBPB is a faulted condition event consistent with the criteria defined in ANSI N18.2. Additionally, the stress limits and analysis methods for faulted conditions defined in the ASME Code and Section 3.9N of the FSAR are sufficiently conservative to ensure the structural integrity and operability of components when subjected to faulted condition loads including the DBPB.

3. BVPS-2 is committed to a pump operability program which does not permit permanent deformation of component supports for active pumps. Although specific deformation limits are not quantified, they are addressed in a qualitative manner in FSAR Sections 3.9N.3.2 and 3.9B.3.2.

4. Refer to Section 1.8 for BVPS-2 positions on regulatory guides.

5. Normal plant surveillance of snubbers will detect and report any lack of function or deterioration of performance due to system vibration. Any defective snubbers will be replaced.

6. Snubber end fitting clearance and lost motion are not considered significant in piping analysis or component analysis.

7. The ASME III Code editions applicable to BVPS-2 components (refer to the ASME Code Baseline Document) do not include the use of service stress limits A, B, C, and D.

TABLE 1.9-2 (Cont)

SRP NO. 3.10

TITLE: SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

DIFFERENCES FROM THE SRP:

1. The SRP requests that acceptable load combination and methods for defining dynamic responses for mechanical and electrical equipment be defined in SRP Section 3.9.3. There are differences in the SRP guidelines and the Westinghouse and balance of plant (BOP) methods of load combination and methods for combining dynamic responses for mechanical equipment (refer to Section 3.9.3 remarks). For electrical equipment, the only dynamic loads considered in testing are seismic loads. These seismic loads are not combined by test with other dynamic loads.

2. BVPS-2 takes some alternatives to Regulatory Guides 1.61, 1.89, 1.92, 1.100, and 1.122.

3. Seismic qualification of mechanical equipment is not addressed in FSAR Section 3.10.

4. Section 3.10 of the FSAR does not cover operability of active pumps and valves.

5. Paragraph II.l.a.(2) of the SRP suggests that equipment should be tested in the operational condition and that loadings simulating normal plant conditions should be superimposed on seismic and dynamic loads. This includes flow induced loads and degraded flow conditions. For the tests performed, operational conditions are included where practical, simulated in some manner, or addressed by analysis. Flow loads are not superimposed on seismic loads for valve operability tests.

6. Paragraph II.l.a.(8) suggests that fixture design for seismic tests should simulate actual service mounting and should not cause any extraneous dynamic coupling to the test item.

7. If the dynamic testing of a pump or valve is impractical, static testing of the assembly can be performed in accordance with paragraph II.1.a(l0). End loadings are not applied, and all dynamic amplification effects are not included in the static deflection tests for active valves.

8. Paragraph 3.10.II.l.a.(14)(b)iii requires that an analysis be performed to determine the pressure differential and impact energy on a valve disc during a LOCA, and to verify the design adequacy of the disc. Westinghouse specified the loading conditions applicable to each valve. Westinghouse performs design verification on vendor supplied valves to assure that the valves are designed properly and meet the

TABLE 1.9-2 (Cont)

stress acceptance criteria specified both in the Equipment Specifications and the ASME Code Standards. The valve suppliers use the guidance provided in NB-3552(C) of the ASME Code to determine if any analysis is warranted.

9. Paragraph II.1.a.(14)(b)viii suggests the use of Regulatory Guide 1.92 for combination of multimodal and multidirectional responses in analyses. Westinghouse uses the methods defined in FSAR Section 3.7 for combining closely spaced modes.

10. Paragraph II.5.b suggests that a list of systems necessary to perform the functions outlined in the SRP Section 3.10 be included in the FSAR Section 3.10. This list is not included in FSAR Section 3.10N.

11. Paragraph II.5.b(2) suggests that a description of the results of any in-plant tests used to confirm qualification of equipment be included in the FSAR.

12. Paragraph II.5.c suggests a Seismic Qualification Report. Westinghouse does not maintain such a report for BVPS-2.

REMARKS:

1. Seismic loads are the only dynamic loads considered for electrical equipment. Interface requirements dictate that electrical equipment be protected from such dynamic loads as jet impingement and pipe whip. The exceptions to the above statement are line mounted electrical components (such as RTDs and valve accessories) in ASME Class 1 systems. This line mounted equipment is subjected to blowdown loads from a LOCA. However, these blowdown loads are enveloped by the seismic loads and are not combined with the seismic loads because of the low probability of the simultaneous occurrence of the two events. The second exception is the electrically operated pressurizer PORV which is subject to transient flow loads. These loads are not combined with seismic loads during test. However, the design load combination includes consideration of transient flow load effects (pressure loads resulting from LOCA) combined with seismic loads. This combined effect is evaluated by analysis for the pressurizer PORVS. In addition, seismic testing in accordance with WCAP-8587 has been conducted on this device.

2. In accordance with the SRP, since the BVPS-2 construction permit application was docketed before October 27, 1972, Class I electrical equipment will be seismically qualified in accordance with IEEE 344-1971. Although not required (due to Beaver Valley's docket date being before October 27, 1972), IEEE 344-1975 was employed for seismic qualification of Seismic Category I electrical equipment when feasible. Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

3. Seismic qualification of mechanical equipment is addressed in FSAR Sections 3.7 and 3.9 in accordance with SRPS 3.7 and 3.9.

4. The BVPS-2 operability program is covered in FSAR Sections 3.9N.3.2. and 3.9B.3.2. The latest version of the SRP has included operability under Section 3.10 and has deleted it from Section 3.9.

5. Full operational testing conditions are not included in testing because performing such a test is impractical. For example, when static deflection tests on valves are performed, the P across the valve disc is simulated. However, the test is not performed with the valve in a flow loop. As stated above, operational conditions are addressed other than by test.

   The active valve operability program defined in FSAR Sections 3.9N.3.2 and 3.9B.3.2 outlines the program for demonstrating operability under all required plant conditions. This program of conservative design, analysis, and test provides adequate assurance that safety related equipment will perform the required safety functions under the appropriate plant conditions.

6. Seismic qualification testing configurations are designed to represent the typical plant installation for the tested component. Interface requirements are defined based on the test configuration and other design requirements. Installation is then completed in accordance with the component interface requirements. Any dynamic coupling effects that result from mounting the component in accordance with these interface criteria would have been adequately considered during the test program.

7. Conservative restrictions are placed on the allowable piping loads transmitted to the valve or pump body such that these loads cannot cause detrimental deflections of the active component. This restriction of allowable piping loads combined with the static deflection testing performed on active valves provides adequate assurance of valve performance and obviates the need to apply end loadings during the static deflection tests for active valves.

   The operability program for active valves addresses dynamic amplification effects by increasing the g loadings utilized in static deflection tests and analyses when dynamic equipment response is a concern. In most cases the equipment is rigid and does not display dynamic amplification characteristics.

8. In conformance with NA-3254 of the ASME Code Section III, Westinghouse utilizes the mechanism of Equipment Design Specifications to address the equipment boundaries of jurisdiction. All loading conditions which are expected to be applicable to the hardware are therefore covered explicitly in the Equipment Specifications. By so doing, Westinghouse conforms to NA-3254. In most cases, the vendor invokes NB-3552(c) of

TABLE 1.9-2 (Cont)

the ASME Code states that any transient effects due to maloperation or accident whose number of occurrences does not exceed 5 need not be considered.

9. The Westinghouse methods for combining closely spaced modes has been previously justified and accepted by the USNRC.

10. The information requested is included in FSAR Chapters 6 and 7, Safety Related Mechanical and Electrical Systems.

11. The test results for Westinghouse supplied equipment are referenced in the FSAR. Actual test results are not included in the FSAR.

12. Seismic qualification of equipment is documented in test reports, analysis reports, and calculation notes contained in Westinghouse files. The documentation maintained by Westinghouse satisfies existing regulatory requirements and, therefore, it is not considered necessary to prepare an additional Seismic Qualification Report.

TABLE 1.9-2 (Cont)

SRP NO. 3.11

TITLE: ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

DIFFERENCES FROM THE SRP:

1. SRP 3.11 suggests that all mechanical and electrical systems and equipment necessary to perform the functions listed in paragraph 1 of Section I of the SRP 3.11 should be listed in Section 3.11 of the FSAR. Section 3.11 does not include the entire list.

2. BVPS-2 meets or exceeds the guidelines of NUREG-0588, Rev. 1, for a Category II plant in lieu of IEEE Standard 323-1974 which is endorsed by Regulatory Guide 1.89.

3. Qualification of safety related mechanical equipment consists of seismic and operability type programs as addressed in FSAR Sections 3.7 and 3.9.

REMARKS:

1. The information requested by SRP 3.11 is located in different parts of the FSAR as listed below:

    a. Safety related Mechanical and Electrical Systems - FSAR Chapters 6 and 7.

    b. Active Pumps and Valves - FSAR Tables 3.9N-9 and 3.9N-10.

    c. Plant Specific Class 1E components list - FSAR Table 3.11-1.

    d. Other safety related mechanical components - FSAR Sections 3.2N and 3.9N.

2. The environmental qualification of Class I electrical equipment is in accordance with the USNRC Commissioners' memorandum and order CLI-80-21.

3. Preparation and submittal of information pertaining to environmental qualification of mechanical equipment is pending USNRC rule making.

TABLE 1.9-2 (Cont)

SRP NO. 4.5.1

TITLE: CONTROL ROD DRIVE STRUCTURAL MATERIALS

DIFFERENCES FROM THE SRP:

Cleaning and cleanliness controls differ slightly from SRP 4.5.1. Alternatives are also taken to Regulatory Guides 1.31 and 1.44.

REMARKS:

FSAR Section 1.8 provides a discussion of cleaning and cleanliness controls used at BVPS-2. Although these controls differ from Regulatory Guide 1.37 guidelines, they are in accordance with the requirements of GDC 1.

For the BVPS-2 position on Regulatory Guides 1.31 and 1.44, refer to FSAR Section 1.8.

TABLE 1.9-2 (Cont)

SRP No. 4.5.2

TITLE: REACTOR INTERNAL AND CORE SUPPORT MATERIALS

DIFFERENCE FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.31 and 1.44.

REMARKS:

For the BVPS-2 position on these Regulatory Guides, refer to FSAR Section 1.8.

TABLE 1.9-2 (Cont)

SRP NO. 5.2.1.1

TITLE: COMPLIANCE WITH THE CODES AND STANDARD RULE, 10 CFR 50.55a

DIFFERENCES FROM THE SRP:

BVPS-2 uses the safety classification system provided in ANSI N18.2 as an acceptable alternative to the Regulatory Guide 1.26 quality group classification system.

REMARKS:

The ANS classification system implemented for BVPS-2 has been endorsed by industry as an alternative accepted by the USNRC for many other plants. This classification system meets the relevant requirements of GDC 1 and 10 CFR 50.55a.

TABLE 1.9-2 (Cont)

SRP NO. 5.2.1.2

TITLE: APPLICABLE CODE CASES

DIFFERENCES FROM THE SRP:

Code cases other than those endorsed by Regulatory Guides 1.84, 1.85, and 1.147 may be used at BVPS-2.

REMARKS:

Code cases not endorsed by the regulatory guides will be limited to those for which USNRC acceptance has been obtained, otherwise assured, or as discussed in FSAR Section 1.8.

TABLE 1.9-2 (Cont)

SRP NO. 5.2.2

TITLE: OVERPRESSURE PROTECTION

DIFFERENCES FROM THE SRP:

NUREG-0737, Items II.D.1 and II.D.3, are not discussed in FSAR Section 5.2.2.

REMARKS:

This material is discussed in Section 1.10.

TABLE 1.9-2 (Cont)

SRP NO. 5.2.3

TITLE: REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.31, 1.36, 1.37, 1.43, 1.44, 1.50 and 1.71.

REMARKS:

The BVPS-2 position on these Regulatory Guides is discussed in FSAR Section 1.8.

TABLE 1.9-2 (Cont)

SRP NO. 5.2.5

TITLE: REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

DIFFERENCES FROM THE SRP:

BVPS-2 does not fully follow the guidelines of Regulatory Guide 1.45, Position C.6 in that the containment sump level detection systems, which meet the detection sensitivity requirements of normal plant operation (that is, 1 gpm in 1 hour), are not designed to meet seismic requirements.

REMARKS:

Seismically qualified redundant level instrumentation, powered from Class lE electrical buses, is provided to monitor the containment sump level for excessive leakage following seismic events. The instrumentation is qualified for seismic loads up to SSE loads. Although a sensitivity of 1 gpm in 1 hour is not achieved, narrow range instruments are provided with an accuracy of ±1.7 inches. The airborne particulate radiation monitoring system is seismically designed for SSE loads and detects a leakage rate of one gpm in less than one hour. The combination of these two types of instruments satisfies the relevant requirements of GDCS 2 and 30.

TABLE 1.9-2 (Cont)

SRP NO. 5.3.1

TITLE: REACTOR VESSEL MATERIALS

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.31, 1.37, 1.43, 1.44, 1.50, and 1.65.

REMARKS:

The BVPS-2 positions on these regulatory guides are discussed in FSAR Section 1.8.

TABLE 1.9-2 (Cont)

SRP NO. 5.4.1.1

TITLE: PUMP FLYWHEEL INTEGRITY (PWR)

DIFFERENCES FROM THE SRP:

The reactor coolant pump flywheel integrity is not completely addressed in FSAR Section 5.4.1.1.

REMARKS:

The pump flywheel integrity is addressed in FSAR Sections 5.4.11 through 5.4.1.5 and Section 1.8.

TABLE 1.9-2 (Cont)

SRP NO. 5.4.2.1

TITLE: STEAM GENERATOR MATERIALS

DIFFERENCES FROM THE SRP:

1. Specific details of the secondary side chemistry program are not discussed in FSAR Section 5.4.2.1.

2. The SRP states that the steam generators should be visually inspected to be in a “metal clean” surface condition before start-up.

3. The SRP states that access for tooling to remove sludge by lancing from the tube support plates should be provided.

REMARKS:

1. The secondary side water chemistry program at BVPS-2 will follow the guidance of Westinghouse document SIP 5-4.

2. The lay-up procedure for steam generators during the time after removal from site storage for installation in containment and after installation is described in the Westinghouse document "NSSS Component Receiving and Storage Guidelines" specifically procedure SGE-32001, "Environmental Protection of Steam Generators During Site Installation and the Period Initially Preceding Water Fill." (During on-site storage the steam generator is sealed and pressurized in a nitrogen atmosphere to inhibit corrosion.)

   Procedure SGE-32001 states that "light rust film may be present and shall not, by itself, be cause for further cleaning." Adherence to SGE-32001 will minimize the potential for a significant amount of corrosion to occur in the steam generator during the remainder of site construction. Procedure SGE-32001 is stringent enough to provide adequate protection from significant corrosion during this layup period. A small amount of corrosion product (oxide) retards further corrosion by acting as a barrier to continued corrosion attack.

3. Historically, tube lancing from tube support plates has not been performed. Sludge filters down through flow slots and flow holes in the support plates to accumulate at low flow areas on the top of the tube sheet. There are two techniques used to remove sludge from the top of the tube sheet. During normal operation, utilization of the steam generator blowdown system will remove a portion of the accumulating sludge. During shutdowns, sludge lancing can be employed to remove sludge not removed by normal blowdown. Removal of sludge from the tubesheet is therefore more efficient than sludge removal from the tube support plates would be.

TABLE 1.9-2 (Cont)

SRP NO. 5.4.7

TITLE: RESIDUAL HEAT REMOVAL (RHR) SYSTEM

DIFFERENCES FROM THE SRP:

1. Task action plan items II.E.3.2 and II.E.3.3 of NUREG-0660 are not discussed.

2. Task action plan item III.D.1.1 of NUREG-0737 is not discussed in FSAR Section 5.4.7.

REMARKS:

1. Since Office of Nuclear Reactor Regulation (NRR) studies regarding these items are incomplete, they need not be addressed.

2. Item III.D.1.1 has no basis in the requirements specified in this SRP. It is, however, discussed in FSAR Section 1.10.

TABLE 1.9-2 (Cont)

SRP NO. 5.4.12

TITLE: REACTOR COOLANT SYSTEM HIGH POINT VENTS

DIFFERENCES FROM THE SRP:

1. The reactor coolant system high point vents are not discussed in FSAR Section 5.4.12.

2. The one-inch pipe LOCA is larger than a leak.

REMARKS:

1. The vents are discussed in FSAR Section 5.4.15, in accordance with the format of Regulatory Guide 1.70.

2. This line is necessary for letdown in the cold shutdown scenario, as discussed in FSAR Appendix 5A.

TABLE 1.9-2 (Cont)

SRP NO. 6.1.1

TITLE: ENGINEERED SAFETY FEATURES MATERIALS

DIFFERENCES FROM THE SRP:

1. The recommendations of ASME Section III, Appendix D, Article D-1000 was not required.

2. Moisture control on low hydrogen welding materials used for shop welding fabrication was not required to conform to both ASME Section III and AWSD1.1.

3. BVPS-2 takes some alternatives to Regulatory Guides 1.44, 1.54, and 1.71.

REMARKS:

1. Welding preheat was specified in accordance with ASME Sections III and IX with the additional provisions of Regulatory Guide 1.50 as discussed in FSAR Section 1.8.

   It was determined that either subsequent post-weld heat treatment or final nondestructive examination in accordance with ASME Section III provides adequate assurance of weld quality.

2. The requirements of either ASME, Section III, or AWSD.1.1, and the applicable code, Section II, SFA filler material specifications were met for shop fabrications.

3. Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 6.2.1.1.A

TITLE: PWR DRY CONTAINMENTS, INCLUDING SUBATMOSPHERIC CONTAINMENTS

DIFFERENCES FROM THE SRP:

1. SRP 6.2.1.1.A, paragraph II.g, states that instrumentation capable of operating in the post-accident environment should be provided to monitor containment atmosphere temperature and sump water temperature.

2. SRP 6.2.1.1.A, paragraph II.F states, that adequate margin above the maximum expected external pressure should be provided. This margin was not addressed in the FSAR.

REMARKS:

1. These two types of post-accident monitors are not required in order to perform a diagnosis as specified by the emergency response guidelines. Attachments 4 and 5 of NUREG-0737, Item II.F.1 are not addressed in FSAR Section 6.2.1.1; they are addressed in Section 7.5.

2. Conservatism in the analysis provides for this margin.

TABLE 1.9-2 (Cont)

SRP NO. 6.2.1.3

TITLE: MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

DIFFERENCES FROM THE SRP:

SRP 6.2.1.3, paragraph II.B.3e, identifies an acceptable Westinghouse mass and energy release model (Ref. 18 of SRP 6.2.1) which was not used for BVPS-2.

REMARKS:

The Westinghouse mass and energy release model for containment design is described in FSAR Sections 6.2.1.3.3, 6.2.1.3.4, 6.2.1.3.5, and 6.2.1.3.6. This model provides sufficient conservatism and satisfies GDC 50 requirements.

TABLE 1.9-2 (Cont)

SRP NO. 6.2.1.5

TITLE: MINIMUM CONTAINMENT PRESSURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM PERFORMANCE CAPABILITY STUDIES

DIFFERENCES FROM THE SRP:

BTP 6-1, Section B.3b, recommends conservative condensing heat transfer coefficients which differ from the Westinghouse model.

REMARKS:

The Westinghouse evaluation model for ECCS minimum containment pressure as presented in Appendix A of WCAP-8339 (1974) satisfies the governing requirements and has been approved by the USNRC staff.

TABLE 1.9-2 (Cont)

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TABLE 1.9-2 (Cont)

SRP No. 6.2.4

TITLE: CONTAINMENT ISOLATION SYSTEM

DIFFERENCES FROM THE SRP:

Containment isolation arrangements which differ from the arrangements specified in the General Design Criteria of SRP 6.2.4 are found in the following:

1. Emergency core cooling system safety injection lines.

2. Recirculation spray pump suction lines.

3. Containment vacuum pump and hydrogen recombine suction and discharge lines.

4. Containment depressurization.

5. Containment leakage monitoring.

6. Fuel transfer tube.

7. Main steam, steam generator blowdown, and feedwater penetrations.

8. Auxiliary feedwater.

9. Personnel air lock and equipment hatch.

10. Reactor coolant pump seal injection.

REMARKS:

The containment isolation system (CIS) meets the intent of NUREG-0737, Item II.E.4.2 with regard to containment isolation dependability. The CIS is activated automatically when monitored system variables exceed pre-established set points. It is neither necessary nor desirable that every containment isolation valve close simultaneously upon a containment isolation signal. The plant design allows selected valves in the engineered safety features systems, which are essential to mitigate the effects of an accident, to remain in or move to their open position. These pre-selected valves are remote and manually-operated from the main control room. Containment isolation arrangements which differ in some manner from the specific arrangements allowed by GDC 54, 55, 56, and 57 are described in Section 6.2.4.2.

TABLE 1.9-2 (Cont)

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TABLE 1.9-2 (Cont)

SRP NO. 6.3

TITLE: EMERGENCY CORE COOLING SYSTEM

DIFFERENCES FROM THE SRP:

1. Task action plan item III.D.1.1 of NUREG-0737 is not discussed in FSAR Section 6.3.

2. Item II.K.3.10 of NUREG-0737 is not discussed in FSAR Section 6.3.

REMARKS:

1. Item III.D.1.1 has no basis in the underlying requirements for this SRP. It is, however, discussed in FSAR Section 1.10.

2. Item II.K.3.10 is addressed in Section 7.2

TABLE 1.9-2 (Cont)

SRP NO. 6.4

TITLE: CONTROL ROOM HABITABILITY SYSTEMS

DIFFERENCES FROM THE SRP:

1. BVPS-2 takes some alternatives to Regulatory Guides 1.78 and 1.95.

2. BVPS-2 takes some alternatives to Regulatory Guide 1.52.

3. BVPS-2 takes some alternatives to the x/Q methodology referenced in SRP 6.4.

REMARKS:

1. Refer to FSAR Section 1.8 for BVPS-2 positions on Regulatory Guides 1.78 and 1.95. The only toxic gas BVPS-2 will monitor is chlorine, based upon the results of the Control Room Habitability Study, described in FSAR Section 2.2.3.1.2.

2. Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.52.

3. Refer to UFSAR Section 2.3.4.3 for the x/Q methodology used in analyses performed after 1991.

TABLE 1.9-2 (Cont)

SRP NO. 6.5.1

TITLE: ESF ATMOSPHERE CLEANUP SYSTEMS

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guide 1.52.

REMARKS:

Refer to FSAR Section 1.8 for the BVPS-2 position of Regulatory Guide 1.52.

TABLE 1.9-2 (Cont)

SRP NO. 7.2

TITLE: REACTOR TRIP SYSTEM

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.53, 1.62, 1.75, 1.105, and 1.118.

REMARKS:

Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 7.3

TITLE: ENGINEERED SAFETY FEATURES SYSTEMS

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.53, 1.62, 1.75, 1.105, and 1.118.

REMARKS:

Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 7.4

TITLE: SAFE SHUTDOWN SYSTEMS

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.53, 1.62, 1.75, 1.105, and 1.118.

REMARKS:

Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 7.5

TITLE: INFORMATION SYSTEMS IMPORTANT TO SAFETY

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.53, 1.75, 1.97, and 1.118.

REMARKS:

Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 7.6

TITLE: INTERLOCK SYSTEMS IMPORTANT TO SAFETY

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.53, 1.62, 1.75, 1.105, and 1.118.

REMARKS:

Refer to FSAR Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 8A

TITLE: BRANCH TECHNICAL POSITIONS (PSB)

DIFFERENCES FROM THE SRP:

Branch Technical Position PSB-1, paragraph B.1:

Two separate time delays for the second level of undervoltage protection are not provided. The Class 1E distribution system is not separated from offsite on safety injection signal and the load shedding feature is not reinstated once the transfer onto the diesel generator units has occurred.

REMARKS:

Two levels of undervoltage protection are provided. The first level detects the loss of power, isolates the emergency system from the normal system, starts the diesel generator, and begins load sequencing. The second level undervoltage scheme detects a degraded voltage condition and, if this condition does not improve within one minute, isolates the emergency system from the normal system, starts the diesel generator, and begins sequencing.

Both of these levels of protection utilize sequential logic from different potential transformers and relays at both the 480 volt and 4,160 volt systems.

The two schemes described satisfy all the safety functions required to assure the protection and safe shutdown of the station.

Branch Technical Position, ICSB 18:

Redundant position indication in the main control room is not provided for the hydrogen recombiner isolation valves 2HCS*MOV110A, B, 2HCS*MOV113A, B. All of these valves have their power supplies normally disconnected to avoid spurious opening during conditions when it is essential that they remain closed.

REMARKS:

The hydrogen control system (HCS) valves are controlled from remote electrical control stations in the plant and are closed during normal and post-accident operation. The only time these valves are intended to be open are for testing purposes during normal plant operation or if both hydrogen recombiners were to fail post-accident after the recombiners have been placed in operation. To ensure that spurious opening of the subject valves does not cause failure of the hydrogen

TABLE 1.9-2 (Cont)

control system to perform its safety function in a post-accident condition, an administrative procedure will be implemented to require that the valve breakers are maintained de-energized during normal and post-accident modes of operation.

TABLE 1.9-2 (Cont)

SRP NO. 9.1.1

TITLE: NEW FUEL STORAGE

DIFFERENCES FROM THE SRP:

The BVPS-2 design does not follow all the recommendations of ANS 57.1 and ANS 57.3.

REMARKS:

The ANS standards had not been issued at the time the BVPS-2 design was established.

TABLE 1.9-2 (Cont)

SRP No. 9.1.2

TITLE: SPENT FUEL STORAGE

DIFFERENCES FROM THE SRP:

The BVPS-2 design does not follow all the recommendations of ANS 57.2.

REMARKS:

The ANS standard had not been issued at the time the BVPS-2 design was established.

TABLE 1.9-2 (Cont)

SRP NO. 9.1.4

TITLE: LIGHT LOAD HANDLING SYSTEM (RELATED TO REFUELING)

DIFFERENCES FROM THE SRP:

1. Regulatory Guide 1.13, paragraph C.3, states that an interlock should be provided to prevent cranes from passing over stored spent fuel when fuel handling is not in progress. The BVPS-2 motor-driven platform crane has, instead, an adjustable interlock which will stop hoist motion if a preset weight, based on the weight of a single fuel assembly, is exceeded.

2. The maximum potential kinetic energy capable of being developed by any load handled above stored spent fuel may, if dropped, exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel storage racks.

3. The BVPS-2 design does not follow all the recommendations of ANS-57.1.

REMARKS:

1. Refer to Section 1.8 for the BVPS-2 position on Regulatory Guide 1.13.

2. An analysis has been performed to demonstrate that although the kinetic energy of a dropped tool exceeds that of a fuel assembly and its associated handling tool, there is no adverse safety impact.

3. The ANS standard had not been issued at the time the BVPS-2 design was established.

TABLE 1.9-2 (Cont)

SRP NO. 9.1.5

TITLE: OVERHEAD HEAVY LOAD HANDLING SYSTEMS

DIFFERENCES FROM THE SRP:

1. SRP 9.1.5 suggests using NUREG-0554, "Single Failure Proof Cranes." None of the cranes or monorails at BVPS-2 were designed to be single failure proof.

2. FSAR Section 9.1.5 does not follow the guidance of NUREG-0612 in the following areas.

    a. The design of special lifting devices at BVPS-2 did not follow ANSI N14.6-1978.

    b. Lifting devices of standard industry design should meet the ANSI B30.9-1971 standard.

    c. NUREG-0612 calls for safe load paths to be marked on floors and for safe load paths to follow structural floor members.

    d. NUREG-0612 states that if physical separation of safe shutdown equipment or a load drop analysis for a load falling through a floor is not provided, then cranes must be upgraded to single failure proof.

3. The BVPS-2 design does not follow all the recommendations of ANS-57.1 and ANS-57.2

REMARKS:

1. NUREG-0554 was issued in 1979, which is after all BVPS-2 cranes and monorails were designed. Also, the NUREG does not make any specific references to being applicable to General Design Criteria 2, 4, 5, or 61. Duquesne Light Company (DLC) submitted a report, "Control of Heavy Loads," to the USNRC in 1982, which addresses the design criteria and operating procedures for the load handling systems at BVPS-2, which are two major concerns of NUREG-0554.

2. Administrative procedures to limit crane operations within the safe load paths are utilized to address items 2c and 2d. NUREG-0612 was issued in 1980, which is after all BVPS-2 cranes and monorails were designed. Also, the NUREG does not make any specific references to being applicable to General Design Criteria 2, 4, 5 or 61. The DLC, "Control of Heavy Loads," report examined BVPS-2 compliance with NUREG-0612.

3. The ANS standards had not been issued at the time the BVPS-2 design was established.

TABLE 1.9-2 (Cont)

SRP NO. 9.2.2

TITLE: REACTOR AUXILIARY COOLING WATER SYSTEMS

DIFFERENCES FROM THE SRP:

1. BVPS-2 Reactor coolant pumps have not been demonstrated by testing to withstand a 20-minute loss of cooling water.

2. Primary component cooling water flow to the reactor coolant pumps is monitored by instrumentation which was designed and procured to standards other than IEEE 279.

REMARKS:

1. GDC 44 is cited by the SRP as a basis for the 20-minute acceptance criterion. GDC 44 is concerned with Provisions for transfer of heat from structures, systems and components important to safety to an ultimate heat sink and is unrelated to the length of time a reactor coolant pump can operate following loss of cooling water. Pumps similar to those used at BVPS-2 have been demonstrated to function for at least 10 minutes without cooling water.

2. Flow instrumentation is described in FSAR Section 9.2.2.1 and is normally supplied with power from an emergency bus.

TABLE 1.9-2 (Cont)

SRP NO. 9.3.4

TITLE: CHEMICAL AND VOLUME CONTROL SYSTEM (PWR) (INCLUDING BORON RECOVERY SYSTEM)

DIFFERENCES FROM THE SRP:

Task action plan item III.D.1.1 of NUREG-0737 is not discussed in FSAR Section 9.3.4.

REMARKS:

Item III.D.1.1 has no basis in the requirements specified in this SRP. It is, however, discussed in FSAR Section 1.10.

TABLE 1.9-2 (Cont)

SRP NO. 9.4.1

TITLE: CONTROL ROOM AREA VENTILATION SYSTEM

DIFFERENCES FROM THE SRP:

1. BVPS-2 takes some alternatives to Regulatory Guides 1.78 and 1.95.

2. BVPS-2 takes some alternatives to Regulatory Guides 1.52 and 1.140.

REMARKS:

1. Refer to FSAR Section 1.8 for BVPS-2 positions on Regulatory Guides 1.78 and 1.95. The only toxic gas BVPS-2 will monitor is chlorine, based upon the results of the Control Room Habitability Study, described in FSAR Section 2.2.3.1.2.

2. Refer to FSAR Section 1.8 for BVPS-2 positions on Regulatory Guides 1.52 and 1.140.

TABLE 1.9-2 (Cont)

SRP NO. 9.4.2

TITLE: SPENT FUEL POOL AREA VENTILATION SYSTEM

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.52 and 1.140.

REMARKS:

Refer to regulatory guide positions in Section 1.8 of the FSAR.

TABLE 1.9-2 (Cont)

SRP NO. 9.4.3

TITLE: AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guide 1.140.

REMARKS:

Refer to regulatory guide positions in Section 1.8 of the FSAR.

TABLE 1.9-2 (Cont)

SRP NO. 9.4.5

TITLE: ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guides 1.52 and 1.140.

REMARKS:

Refer to regulatory guide positions in Section 1.8 of the FSAR.

TABLE 1.9-2 (Cont)

SRP NO. 9.5.1

TITLE: FIRE PROTECTION PROGRAM

DIFFERENCES FROM THE SRP:

BVPS-2 takes alternatives to the guidance provided in Branch Technical Position CMEB 9.5-1. These alternatives and the associated justifications are provided in Appendix 9.5A (Section 9.5A.2).

TABLE 1.9-2 (Cont)

SRP 9.5.4

TITLE: EMERGENCY DIESEL ENGINE FUEL OIL STORAGE AND TRANSFER SYSTEM

DIFFERENCES FROM THE SRP:

1. BVPS-2 takes an alternative to Regulatory Guide 1.137.

2. The fill lines to the diesel generator fuel oil tanks are not tornado protected.

3. There are no tank design features which minimize the creation of turbulence of the accumulated residual sediments.

REMARKS:

1. Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.137.

2. Alternate methods of filling the fuel oil tank are provided through the sample connection or the vent flame arrestor connection which are located in a missile protected area.

3. The admission of deleterious material is prohibited by strainers in the fuel oil tank fill line. Strainers are also provided on the fuel oil transfer pump discharge, and a duplex filter is provided on the emergency diesel generator set.

TABLE 1.9-2 (Cont)

SRP NO. 9.5.6

TITLE: EMERGENCY DIESEL ENGINE STARTING SYSTEM

DIFFERENCES FROM THE SRP:

BVPS-2 does not rely on air dryers in the diesel generator air starting system to assure adequate dehydration.

REMARKS:

Accumulated moisture will be removed from the starting system by blowdown of the air receivers at regular intervals. Blowdown will be ensured by implementing administrative controls such as operating surveillance tests and will satisfy the relevant requirements of GDC 17.

TABLE 1.9-2 (Cont)

SRP NO. 10.4.3

TITLE: TURBINE GLAND SEALING SYSTEM

DIFFERENCES FROM THE SRP:

BVPS-2 takes some alternatives to Regulatory Guide 1.26.

REMARKS:

Refer to Section 1.8 for BVPS-2 positions on regulatory guides.

TABLE 1.9-2 (Cont)

SRP NO. 10.4.6

TITLE: CONDENSATE CLEANUP SYSTEM

DIFFERENCES FROM THE SRP:

Specific details of the secondary side chemistry program are not discussed in FSAR Section 10.4.6.

REMARKS:

The secondary side water chemistry program at BVPS-2 will follow the guidance of Westinghouse document SIP 5-4.

TABLE 1.9-2 (Cont)

SRP NO. 10.4.7

TITLE: CONDENSATE AND FEEDWATER SYSTEM

DIFFERENCES FROM THE SRP:

FSAR Section 10.4.7 does not specify provisions for start-up feedwater hammer testing per BTB ASB 10-2, Item 3.

REMARKS:

The system is adequately protected against damaging feedwater hammer by the feeding J-tube design, coupled with the elbow arrangement at the feedwater piping connection. Therefore, no testing is necessary.

TABLE 1.9-2 (Cont)

SRP NO. 10.4.8

TITLE: STEAM GENERATOR BLOWDOWN SYSTEM (PWR)

DIFFERENCES FROM THE SRP:

The system does not follow the guidance of Regulatory Guide 1.143, paragraph C.1 with respect to the turbine building seismic criteria.

REMARKS:

Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.143. Although the system differs from the SRP acceptance criteria, its design meets the requirements of GDC 1, 2, and 14.

TABLE 1.9-2 (Cont)

SRP NO. 11.1

TITLE: SOURCE TERMS

DIFFERENCES FROM THE SRP:

1. SRP 11.1, paragraph II.a suggests that a cost-benefit analysis be performed in accordance with Regulatory Guide 1.110 for the radioactive waste management systems and equipment.

2. BVPS-2 takes some alternatives to Regulatory Guide 1.140.

REMARKS:

1. BVPS-2 received a construction permit prior to June 4, 1976, the effective date of the cost-benefit analysis requirement. The radwaste systems and equipment described in the FSAR satisfy the regulatory position in Docket RM-50-2, which is reproduced in the Annex to Appendix I to 10 CFR 50.

2. Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.140.

TABLE 1.9-2 (Cont)

SRP NO. 11.2

TITLE: LIQUID WASTE MANAGEMENT SYSTEMS

DIFFERENCES FROM THE SRP:

1. BVPS-2 systems that contain provisions to control leakage and facilitate operation and maintenance meet Regulatory Guide 1.143 with some alternatives.

2a. SRP 11.2, paragraph II.1.a, provides only the "per unit" limits for determining compliance with 10 CFR 50, Appendix I.

2b. SRP 11.2, paragraph II.1.b, suggests that the applicant consider refinements to liquid radwaste treatment systems and performed a cost-benefit analysis in accordance with Regulatory Guide 1.110: liquid radwaste treatment systems should include all items of the system sequentially and in order of diminishing cost-benefit return, which can (for a favorable cost-benefit ratio) effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. This cost-benefit analysis has not been performed.

REMARKS:

1. Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.143.

2a. Analysis for BVPS-2 has been performed on a "per site" basis and resultant doses are within the RM-50-2 design limits and, therefore, is in compliance with 10 CFR 50, Appendix I.

2b. The cost-benefit analysis is not applicable for applicants for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25-30.

    BVPS-2 received its construction permit prior to the requirement for cost-benefit analyses of radwaste treatment systems. Duquesne Light Company was given the option by the USNRC of performing the cost-benefit analyses, but chose not to perform the analyses due to the stage of system design.

TABLE 1.9-2 (Cont)

SRP NO. 11.3

TITLE: GASEOUS WASTE MANAGEMENT SYSTEMS

DIFFERENCES FROM THE SRP:

1. SRP 11.3, paragraph II.B.1.c, requires the gaseous radwaste treatment systems to include all items of reasonably demonstrated technology that when added to the system sequentially and in order of diminishing cost-benefit return, can (for a favorable cost-benefit ratio) effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. This cost-benefit analysis has not been performed.

2. SRP 11.3, paragraph II.B.4, requires system designs that contain provisions to control leakage and to facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143. BVPS-2 meets Regulatory Guide 1.143 with the following exceptions:

    a. Paragraph C.2.1.1

       The gaseous waste system meets or exceeds the codes in Table 1. The gaseous waste delay beds, gaseous waste surge tank, and the overhead compressors have been purchased as ASME III. The waste gas chillers, piping, and valves are ANSI B31.1.

    b. Paragraph C.4.3

       The piping downstream of the overhead gas compressor discharge lines are 1/2 inch, whereas the guide suggests that process lines should not be less than 3/4 inch (nominal).

3. SRP 11.3, paragraph II.B.6.b, requires two independent gas analyzers continuously operating and providing two independent measurements verifying that hydrogen and/or oxygen are not present in potentially explosive concentrations. The two oxygen analyzers do not measure independent locations.

4. The SRP provides only the "per unit" limits for determining compliance with 10 CFR 50, Appendix I.

5. The design case concentrations for gaseous effluents from the ventilation vent exceed the limits of 10 CFR 20, Appendix B, Table II, Column 1, for five nuclides assuming the design basis ventilation rate is used.

TABLE 1.9-2 (Cont)

REMARKS:

1. BVPS-2 received a construction permit prior to June 4, 1976, the date that the requirement for the cost-benefit analysis became effective. The radwaste systems and equipment described in the FSAR satisfy the regulatory position in Docket RM-50-2 which is reproduced in the Annex to Appendix I.

2. The FSAR addresses ETSB 11-1.

    a. The waste gas chillers are a pipe-within-a-pipe heat exchanger designed and fabricated in accordance with ANSI B.31.1. The charcoal delay bed vessels and the gaseous waste surge tank are designed and fabricated to ASME III, a design which is superior to the specified ASME VIII. The overhead gas compressors are designed and fabricated to ASME III, a design which is superior to the specified manufacturer's standard.

    b. The overhead gas compressor prefilters provide a clean gas downstream where the lines are 1/2 inch. Pressure drop is not a problem in the 1/2 inch lines downstream of the overhead gas compressors.

3. The two oxygen analyzers are downstream of the overhead gas compressors with the most likely leakage point being the suction of the compressors.

4. Analysis for BVPS-2 has been performed on a "per site" basis and resultant doses are within the RM-50-2 design limits. It is, therefore, in compliance with 10 CFR 50, Appendix I.

5. Controlled releases will be imposed so as not to exceed the concentration limits of 10 CFR 20, Appendix B, Table II, Column 1.

TABLE 1.9-2 (Cont)

SRP NO. 11.4

TITLE: SOLID WASTE MANAGEMENT SYSTEMS

DIFFERENCES FROM THE SRP:

1. SRP 11.4, paragraphs II.2 and II.3, suggest a description of the methods for solidification, the solidifying agent used, and a process control program to ensure a solid matrix. A process control program has not been submitted by the system vendor but has been discussed for BVPS-2.

2. Regulatory Guide 1.143 suggest 0-15 psig tanks to be designed and fabricated to ASME III, Class 3 or, API 650. The spent resin holding tank and the decanting tank are designed and fabricated to ASME VIII, Division 1.

REMARKS:

1. The FSAR states that a process control program is needed to verify the absence of free liquid in the containers. The required program will be written prior to the completion of the environmental radiological Technical Specification.

2. The deviation of the design and fabrication of the spent resin holding tank and the decanting tank from the requirements of Regulatory Guide 1.143 is due to the geometry of the two tanks. The fabrication codes specified (ASME III, Class 3 or API 650) do not specifically cover tanks of this type, that is, vessel-type nonflat bottom atmospheric tanks. As such, these tanks have been constructed in accordance with ASME VIII, Division 1, which is more stringent than API 650, thus indicating that the SRP requirements are more than adequately addressed. Exception to ASME VIII, Division 1 requirements are:

   a. The vessels are not stamped.

   b. Radiography of welds is not required.

   c. A water fill leak test in accordance with API-650 is substituted for a hydrostatic test at 1.5 times design pressure.

   These tanks were designed and fabricated prior to issuance of NUREG-0800.

TABLE 1.9-2 (Cont)

SRP NO. 11.5

TITLE: PROCESS AND EFFLUENT RADIOLOGICAL MONITORING INSTRUMENTATION AND SAMPLING SYSTEMS

DIFFERENCES FROM THE SRP:

1. Regulatory Guide 1.97 suggests post-accident sampling for primary coolant activity. There are no plans to install equipment to continuously monitor reactor coolant following an accident.

2. BVPS-2 takes some alternatives to Regulatory Guide 1.21.

3. Table 2 of the SRP requires continuous monitoring of the turbine building drain effluents. BVPS-2 does not continuously monitor these effluents.

REMARKS:

1. To meet the intent of Regulatory Guide 1.97, reactor coolant can be analyzed by obtaining a grab sample via the post-accident sampling facility (PASS). The PASS does not operate at all times after an accident.

   Information provided by this Regulatory Guide 1.97 variable would not be vital to the control room operator for safe shutdown of the plant.

2. Refer to FSAR Section 1.8 for the BVPS-2 position on Regulatory Guide 1.21.

3. To obtain gross activity of turbine building drain effluents, grab sampling will be performed as explained in Section 11.2.2

TABLE 1.9-2 (Cont)

SRP NO. 12.1

TITLE: ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

DIFFERENCES FROM THE SRP:

1. 10 CFR 19.12 is not discussed.

2. NUREG-0761 guidelines are not specifically discussed.

REMARKS:

1. The SRP specifies in part that the FSAR is acceptable if sufficient information identified in Section 12.1 of Regulatory Guide 1.70 is provided to meet the relevant requirements of 10 CFR 19 and 20. Regulatory Guide 1.70 does not identify any necessary discussion of 10 CFR 19. The FSAR does, however, discuss 10 CFR 19.12 in Section 13.2.2.1.

2. The Radiation Protection Plan for BVPS-2 will be an extension of the plan currently in effect for the operation of BVPS-1. Having been developed prior to the current guidance (draft NUREG-0761, March 1981), the plan was not designed to follow this guidance document. The plan does meet the intent of draft NUREG-0761 by satisfying the requirements of 10 CFR 20 as described in FSAR Sections 12.5, 13.1, and 13.2.

TABLE 1.9-2 (Cont)

This page has been intentionally deleted from the FSAR.

TABLE 1.9-2 (Cont)

SRP NO. 12.3 - 12.4

TITLE: RADIATION PROTECTION DESIGN FEATURES

DIFFERENCES FROM THE SRP:

FSAR Sections 12.3 and 12.4 do not include:

1. A description of the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of 70.24(b) of 10 CFR Part 70 and Regulatory Guide 8.12 for the new fuel storage area.

2. A description of intermediate range area monitors used to measure radiation exposure rates in locations contiguous to containment building penetrations and hatches, and in vital access areas as described in Regulatory Guide 1.97, Rev. 2.

REMARKS:

1. An application will be filed for an exemption from the installation of the criticality monitor required in accordance with 10 CFR Part 70.24(b) and Regulatory Guide 8.12.

2. General Design Criterion 19 is the only underlying requirement of Regulatory Guide 1.97 which is specified in SRP 12.3-12.4, and is unrelated to the area radiation monitors. However, Regulatory Guide 1.97 indicates that the function of the radiation monitors in areas adjacent to containment penetrations is to indicate a breach of the primary containment building. Because of the high radiation levels due to radiation streaming through containment penetration and systems carrying post-LOCA fluid outside containment, these radiation monitors are not expected to perform the intended function.

   To ensure that personnel exposures do not exceed the guidelines of NUREG-0737, administrative controls require the use of a two-person team. One person performs the vital post-LOCA function and the other surveys the area with portable survey instruments. Additionally, the area monitors listed in FSAR Table 12.3-10 are mounted in selected areas of the plant to provide an indication of the radiation levels up to 10 Rem/hr.

TABLE 1.9-2 (Cont)

SRP NO. 12.5

TITLE: OPERATIONAL RADIATION PROTECTION PROGRAM

DIFFERENCES FROM THE SRP:

1. NUREG-0761 guidelines are not specifically discussed.
2. Regulatory Guide 8.28 is not discussed
3. Regulatory Guide 8.3 is not discussed.

REMARKS:

1. Refer to the SRP 12.1 conformance statement.
2. Regulatory Guide 8.28, "Audible Alarm Dosimeters," is not applicable to BVPS-2. The established Radiation Protection Program does not rely upon the use of audible alarm dosimeters to satisfy regulatory requirements.
3. Regulatory Guide 8.3, "Film Badge Performance Criteria," is not applicable to BVPS-2. The established Radiation Protection Program does not rely upon the use of film badges to satisfy regulatory requirements.

TABLE 1.9-2 (Cont)

SRP NO. 13.2.2

TITLE: TRAINING FOR NON-LICENSED PLANT STAFF

DIFFERENCES FROM THE SRP:

An annual evacuation drill for all employees is not discussed.

REMARKS:

The BVPS-2 Station Orientation Training Program, which is discussed in FSAR Section 13.2.2.1.1, meets the relevant requirements of 10 CFR 50.34(b) and 10 CFR 50, Appendix E, by providing all employees with adequate training to ensure that they are familiar with their emergency response duties.

TABLE 1.9-2 (Cont)

SRP NO. 13.4

TITLE: OPERATIONAL REVIEW

DIFFERENCES FROM THE SRP:

ANSI/ANS-3.l-1978 and Regulatory Guide 1.8 are not addressed with respect to the qualification requirements of personnel assigned to the Offsite Review Committee (ORC).

REMARKS:

The independent operational review of BVPS-2 will be performed by the same ORC that has been performing this review for BVPS-1. Qualification requirements of personnel assigned to the ORC meet those described in ANSI N18.1-1971. Meeting these qualification requirements satisfies the relevant requirements of 10 CFR 50.40(b) by ensuring that the ORC personnel are technically qualified to perform an operational review.

TABLE 1.9-2 (Cont)

SRP NO. 15.1.5

TITLE: STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

DIFFERENCES FROM THE SRP:

Items II.E.1.2, II.K.3.5, and II.K.3.25 of NUREG-0737 are not discussed in FSAR Section 15.1.5.

REMARKS:

Item II.E.1.2 is addressed in Sections 10.4.9 and 7.3, Item II.K.3.5 is addressed in Section 1.10, and Item II.K.3.25 is addressed in Section 5.4.1.

TABLE 1.9-2 (Cont)

SRP NO. 15.2.6

TITLE: LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

DIFFERENCES FROM THE SRP:

Items II.E.1.1 and II.E.1.2 of NUREG-0737 are not discussed in FSAR Section 15.2.6.

REMARKS:

Item II.E.1.1 is addressed in Section 10.4.9; Item II.E.1.2 is addressed in Sections 10.4.9 and 7.3.

TABLE 1.9-2 (Cont)

SRP NO. 15.2.7

TITLE: LOSS OF NORMAL FEEDWATER FLOW

DIFFERENCES FROM THE SRP:

Items II.E.1.1 and II.E.1.2 of NUREG-0737 are not discussed in FSAR Section 15.2.7.

REMARKS:

Items II.E.1.1 and II.E.1.2 are addressed in Section 10.4.9.

TABLE 1.9-2 (Cont)

SRP NO. 15.2.8

TITLE: FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)

DIFFERENCES FROM THE SRP:

Items II.E.1.1, II.E.1.2, II.K.3.5, and II.K.3.25 of NUREG-0737 are not discussed in FSAR Section 15.2.8.

REMARKS:

Item II.E.1.1 is addressed in Section 10.4.9, Item II.E.1.2 is addressed in Sections 10.4.9 and 7.3, Item II.K.3.5 is addressed in Section 1.10, and Item II.K.3.25 is addressed in Section 5.4.1.

TABLE 1.9-2 (Cont)

SRP NO. 15.3.3 - 15.3.4

TITLE: REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

DIFFERENCES FROM THE SRP:

Item II.K.3.5 of NUREG-0737 is not discussed in FSAR Section 15.3.

REMARKS:

Item II.K.3.5 is addressed in Section 1.10.

TABLE 1.9-2 (Cont)

SRP NO. 15.4.8

TITLE: SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

DIFFERENCES FROM THE SRP:

The SRP implies that the stresses should be evaluated to emergency conditions defined in ASME III. Westinghouse considers this a faulted condition as stated in ANSI N18.2. Faulted condition stress limits are applied for this accident.

REMARKS:

System overpressurization due to a rod ejection transient was evaluated in WCAP-7588 Rev. 1-A and received USNRC acceptance in the Topical Report Evaluation.

TABLE 1.9-2 (Cont)

SRP NO. 15.6.1

TITLE: INADVERTENT OPENING OF A PWR PRESSURIZER RELIEF VALVE OR A BWR RELIEF VALVE

DIFFERENCES FROM THE SRP:

Items II.K.3.1, II.K.3.5, and II.K.3.25 of NUREG-0737 are not discussed in FSAR Section 15.6.1.

REMARKS:

Items II.K.3.1 and II.K.3.5 are addressed in Section 1.10 and Item II.K.3.25 is addressed in Section 5.4.1.

TABLE 1.9-2 (Cont)

SRP NO. 15.6.5

TITLE: LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

DIFFERENCES FROM THE SRP:

1. The parameters presented below were used to calculate the radiological consequences. These values are consistent with NUREG-75/087 and Regulatory Guide 1.4, Revision 2.

Iodine available for release from containment	25%
Iodine composition	
Elemental	91%
Particulate	5%
Methyl	4%
Iodine spray removal	10 hr^{-1}
Maximum elemental iodine decontamination factor (NaOH)	100

2. No modifications have been made to the small break model in accordance with Items II.K.3.30 and II.K.3.31 of NUREG-0737.

3. The value assumed for ECCS leakage in radiological calculations (i.e., 2.0×10^{-2} gpm) is twice the expected leakage. The expected leakage was used as a basis since the BVPS-2 Technical Specifications (like the BVPS-1 version) contains no limit on ECCS leakage.

REMARKS:

1. The radiological consequences calculated using the model which is consistent with NUREG-75/087 and Regulatory Guide 1.4, Revision 2, are within the requirements of 10 CFR 100.

2. The BVPS-2 positions on the PWR applicable items from the U.S. Nuclear Regulatory Commission's Clarification of TMI Action Plan Requirements for applicants for an operating license, NUREG-0737, Enclosure 2, dated November 1980 are discussed in FSAR Section 1.10.

3. The BVPS-2 Technical Specifications are modeled after the BVPS-1 Technical Specifications in accordance with NRC letter dated September 18, 1984.

TABLE 1.9-2 (Cont)

SRP NO. 16.0, Rev. 1

TITLE: TECHNICAL SPECIFICATIONS

DIFFERENCES FROM THE SRP:

Technical Specifications are not provided.

REMARKS:

BVPS-2 Technical Specifications are currently being developed and will be provided within the schedule suggested by the USNRC staff. A working copy of the Technical Specifications is continuously available for the USNRC Resident Inspector's use.

TABLE 1.9-2 (Cont)

SRP NO. 17.2

TITLE: QUALITY ASSURANCE DURING THE OPERATIONS PHASE

DIFFERENCES FROM THE SRP:

The description of the Duquesne Light Company Operations Quality Assurance (QA) program does not provide discussions of those areas which are considered to be excessively detailed and, therefore, not appropriate as input to the FSAR.

REMARKS:

FSAR Section 17.2 follows the format and content guidelines of Regulatory Guide 1.70 in describing the operations QA program. Although the description does not address the more detailed SRP acceptance criteria, it provides sufficient information to conclude that the program satisfies the requirements of 10 CFR 50, Appendix B. Many of the SRP criteria are of a detailed nature not suitable for discussion in the FSAR. Methods of meeting these criteria or acceptable alternatives are documented in the Operations QA Program, Operations Quality Control Procedures, Quality Assurance Instructions, and detailed implementing procedures, all of which have been successfully implemented since early 1976 for BVPS-1.

1.10 NUREG 0737 ACTION ITEMS

Table 1.10-1 presents the BVPS-2 positions on the PWR applicable items from the U.S. Nuclear Regulatory Commission's Clarification of TMI Action Plan Requirements for applicants for an operating license, NUREG-0737, Enclosure 2, dated November 1980. In addition, this table identifies the location of each individual item addressed in the UFSAR.

Tables for Section 1.10

TABLE 1.10-1

NUREG 0737 CONFORMANCE

Item and Title	Position	UFSAR Reference
I.A.1.1 Shift Technical Advisor	BVPS-2 will meet the intent of this item.	13.2.2.5
I.A.1.2 Shift Supervisor Administrative Duties	BVPS-2 will meet the intent of this item.	13.5.1
I.A.1.3 Shift Manning	BVPS-2 meets the intent of this item. Minimum shift crew composition is specified by 10 CFR 50.54. Controls on working hours are specified by technical specifications.	13.5.1
I.A.2.1 Immediate Upgrade of RO and SRO Training and Qualifications	BVPS-2 will meet the intent of this item.	
I.A.2.3 Administration of Training Programs	BVPS-2 will meet the item by having licensed operation instructors enrolled in the operator's retraining program.	
I.A.3.1 Revise Scope and Criteria for Licensing Examinations	BVPS is building its own simulator and once complete will make it available for the simulator examination portion of NRC licensing examination.	
I.B.1.2 Evaluation of Organization and Management	BVPS-2 will meet the intent of this item.	13.4.4
I.C.1 Short-Term Accident Analysis and Procedures Revision	Following USNRC review and approval of the Westinghouse Owners Group revised emergency procedure guidelines, BVPS-2 will revise its emergency procedures, as necessary, to incorporate these recommendations.	13.5.2
I.C.2 Shift and Relief Turnover Procedures	BVPS-2 Administrative Procedures will meet the intent of this item.	13.5.1
I.C.3 Shift Supervisor Responsibility	BVPS-2 Administrative Procedures will comply with the intent of this item.	13.5.1
I.C.4 Control Room Access	BVPS-2 Administrative Procedures will comply with the intent of this item.	13.5.1

TABLE 1.10-1 (Cont)

Item and Title	Position	UFSAR Reference
I.C.5 Procedures for Feedback of Operating Experience	BVPS-2 Administrative Procedures will comply with the intent of this item.	13.5.1
I.C.6 Procedures for Verification of Correct Performance of Operating Activities	BVPS-2 Administrative Procedures will comply with the intent of this item.	13.5.1
I.C.7 NSSS Vendor Review of Procedures	BVPS-2 remains available to correct any deficiencies found in its low-power, power ascension, and emergency procedures, as necessary, if the NRC opts to require a NSSS vendor review. However, it may not be necessary considering BVPS-2 response to I.C.1.	*
I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOLS	BVPS-2 remains available to correct any deficiencies found in its emergency procedures if the USNRC opts to conduct a pilot monitoring program. The PMP may not be necessary considering the BVPS-2 response to Item I.C.1.	*
I.D.1 Control Room Design Review	A control room design review has been performed for BVPS-2 to meet this item.	18.0
I.D.2 Plant Safety Parameter Display Console	BVPS-2 will install a safety parameter display system.	7.5.6
I.G.1 Training During Low-Power Testing	The training objectives required by NUREG-0737, Item I.G.1 will be satisfied and operator training will be provided on a simulator which adequately represents BVPS-2 performance with regard to natural circulation. Test data which could be used to update existing simulator training will be obtained using the approved guidelines of the revised Westinghouse Low Power Test Program (W NS-EPR-2465 dated July 8, 1981) as indicated below: 1. During hot functional testing with the reactor coolant pumps supplying heat input to the secondary side, a loss of AC power will be simulated to the auxiliary feed pumps, controls and area ventilation. This will demonstrate that the plant can be stabilized utilizing manual control and the steam driven auxiliary feedwater pump. Informational data will be taken for simulator update as necessary, with no acceptance criteria applied. 2. The existing initial startup test, (Section 14.2.12.8.13), “Pressurizer Heater and Spray Capability” test will be revised to include a section with one RCP in operation	

Item and Title	Position	UFSAR Reference
	(not to be in loops with pressurizer surge line or spray line). Pressure will be reduced by turning off pressurizer heaters and noting depressurization rate. The heaters will then be re-established and pressure further reduced by use of auxiliary spray. The effects of changes in charging flow and steam flow on margin to saturation temperature will be observed. Test data will be recorded and will be available for simulator update as necessary, with no acceptance criteria applied.	
	3. BVPS-2 does not plan to conduct this natural circulation test since results of testing previously performed at North Anna-2 are sufficient to demonstrate the adequacy of BVPS-2 design features related to natural circulation. BVPS-1 has experienced loss of AC power and has satisfactorily demonstrated natural circulation during this transient.	
	4. The existing initial startup test (Section 14.2.12.6.5), “Verification of Plant Performance Following Turbine Trip Coincident with Loss of Offsite Power at Load” is satisfactory for obtaining the necessary plant conditions. Using this test, the plant will be brought to stable conditions using batteries and emergency diesels.	
II.B.1 Reactor Coolant System Vents	BVPS-2 meets the intent of this item.	5.4.13 5.4.15
II.B.2 Plant Shielding	BVPS-2 will meet the intent of this item.	12.3.2
II.B.3 Post-Accident Sampling	The requirement for a post-accident sampling system was eliminated by License Amendment No. 123.	
II.B.4 Training for Mitigating Core Damage	BVPS-2 will meet the intent of this item.	13.2.2.6

TABLE 1.10-1 (Cont)

Item and Title	Position	UFSAR Reference
II.D.1 Testing Requirements for Reactor Coolant System Relief and Safety Valves	BVPS-2 meets the intent of this item with the exception of ATWS testing.	3.9N.3 5.4.13
II.D.3 Direct Indication of Relief and Safety Valve Position	BVPS-2 meets the intent of this item.	7.5
II.E.1.1 Auxiliary Feedwater System Evaluation	BVPS-2 meets the intent of this item.	10.4.9.3 10A.1
II.E.1.2 Auxiliary Feedwater System Automatic Initiation and Flow Indication	BVPS-2 meets the intent of this item.	7.3 10.4.9
II.E.3.1 Emergency Power Supply for Pressurizer Heaters	BVPS-2 meets the intent of this item.	8.3.1.1.3 8.3.2
II.E.4.1 Dedicated Hydrogen Penetrations (Containment Design)	BVPS-2 meets the intent of this item.	6.2.5.1
II.E.4.2 Containment Isolation Dependability	BVPS-2 meets the intent of this item.	6.2.4 7.3
II.F.1 Additional Accident Monitoring Instrumentation	Attachment (1) BVPS-2 will meet the intent of this item.	11.5.1 11.5.2.4.2
	Attachment (2) BVPS-2 will meet the intent of this item.	11.5.2.4.2 11.5.1
	Attachment (3) BVPS-2 will meet the intent of this item	12.3
	Attachment (4) BVPS-2 meets the intent of this item.	7.5
	Attachment (5) BVPS-2 meets the intent of this item.	7.5
	Attachment (6) BVPS-2 meets the intent of this item.	*

TABLE 1.10-1 (Cont)

Item and Title	Position	UFSAR Reference
II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	BVPS-2 will meet the intent of this item. A detailed description of plant design satisfying this item will be provided in an amendment to BVPS-2 UFSAR.	7.5 7.7.2
II.G.1 Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	BVPS-2 meets the intent of this item.	8.3.1.1.3 8.3.1.2
II.K.1.5 Review ESF Valves	BVPS-2 plant operating procedures will meet the intent of this item.	13.5.2.1
II.K.1.10 Operability Status	BVPS-2 meets the intent of this item.	13.5.2.1
II.K.1.17 Trip per Low-Level B/S	BVPS-2 meets the intent of this item.	7.1 7.2
II.K.2.13 Thermal Mechanical Report	Westinghouse (in support of the Westinghouse Owners Group) has developed methods and performed analyses for a spectrum of small LOCAs. The methods employed the NOTRUMP computer program to generate the thermal/hydraulic transients. The results of the analyses are provided in WCAP-10019.	*
II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients	BVPS-2 meets the intent of this item.	5.4.7.2.3
II.K.2.19 Sequential Auxiliary Feedwater Flow Analysis	This item is not applicable to BVPS-2. The concerns expressed in this item do not apply to NSSS with inverted U-tubes such as the one utilized in BVPS-2.	*
II.K.3.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System	As a result of the evaluation documented in WCAP-9804, February 1981, it has been determined that an automatic pressurizer power-operated relief valve isolation system is not required for BVPS-2.	*

TABLE 1.10-1 (Cont)

Item and Title	Position	UFSAR Reference
II.K.3.2 Report on Overall Safety Effects of Power-Operated Relief Valve Isolation System	A generic report was submitted by the Westinghouse Owners Group which responds to this item (WCAP-9804). This report identifies a significant reduction in the PORV LOCA probability as a result of post-TMI modifications and the calculations compare favorably with the operational data for Westinghouse plants (included as an appendix to the report).	
II.K.3.3 Reporting SV and RV Failures and Challenges	Failures of PORVs or safety valves to close when required will be reported as appropriate in accordance with 10 CFR 50.72 and 10 CFR 50.73.	16
II.K.3.5 Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident	Westinghouse has performed an analysis of delayed reactor coolant pump trip during small-break LOCAs. This analysis is documented in WCAP-9584 and WCAP-9585, August 1979. In addition, Westinghouse has performed test predictions of LOFT experiments L3-1 and L3-6. The results of these predictions are documented in letters OG-49, dated, March 3, 1981; OG-50, dated March 23, 1981; and OG-60, dated, June 15, 1981. Based on: 1) the Westinghouse analysis, 2) the excellent prediction of the LOFT experiment L3-6 results using the Westinghouse analytical model, and 3) Westinghouse simulator data related to operator response time, the Westinghouse & BVPS-2 position is that the automatic reactor coolant pump trip is not necessary, since sufficient time is available for manual tripping of all the pumps.	*
II.K.3.7 Evaluation of PORV Opening Probability	This item is applicable to B&W plants only and therefore, does not apply to BVPS-2.	*
II.K.3.9 Proportional Integral Derivative Controller Modification	BVPS-2 controller (PID) derivative action setting is zero, thereby eliminating it from consideration. Therefore, II.K.3.9 is not applicable to BVPS-2.	*
II.K.3.10 Proposed Anticipatory Trip Modification	BVPS-2 meets the intent of this item.	7.2.1.1.2
II.K.3.12 Confirm Anticipatory Trip	BVPS-2 design meets the intent of this item.	7.2.1.1.2
II.K.3.17 Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specification Changes	BVPS-2 will meet the intent of this item.	13.5.2.1

TABLE 1.10-1 (Cont)

Item and Title	Position	UFSAR Reference
II.K.3.25 Effect of Loss of Alternating Current Power on Pump Seals	BVPS-2 meets the intent of this item.	5.4.1.3.1
II.K.3.30 Revised Small Break Loss-of-Coolant Accident Methods to Show Compliance with 10CFR Part 50, App. K	BVPS-2 position on the small-break LOCA analysis model currently approved by the NRC for use on BVPS-2 is conservative and in conformance with Appendix K to 10CFR Part 50. However (as documented in Anderson 1980), Westinghouse believes that improvement in the realism of small-break calculations is a worth-while effort and is currently developing a revised small-break LOCA analysis model to address NRC concerns (e.g. NUREG-0611, NUREG-0623, etc). Review and approval of the revised W small-break model by the NRC is scheduled for December 1, 1983.	15.6.5
II.K.3.31 Plant Specific Calculations to Show Compliance with 10 CFR Part 50.46	A small-break loss-of-coolant accident analysis specific to BVPS-2 is provided using the present Westinghouse small-break evaluation model. This is in conformance with 10CFR Part 50, Appendix K.	15.6.5
III.A.1.1 Emergency Preparedness, Short Term	BVPS Emergency Preparedness Plan (EPP) is in compliance with standards of 10CFR50 Appendix E and meets the intent of this item.	13.3
III.A.1.2 Upgrade Emergency Support Facilities	BVPS-2 will meet the intent of this item.	13.3
III.A.2 Emergency Preparedness	BVPS-2 Emergency Preparedness Plan will comply with the intent of this item.	13.3
III.D.1.1 Primary Coolant Sources Outside Containment	BVPS-2 plant operating procedures will meet the intent of this item.	13.5.2.1
III.D.3.3 Inplant I Radiation Monitoring	BVPS-2 will meet the intent of this item.	12.3
III.D.3.4 Control Room Habilitability	BVPS-2 meets the intent of this item.	6.4

NOTE:

*Statement stands alone, no UFSAR Section reference

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1.11 ASME CODE CASES

Table 1.11-1 is provided as a convenient means for locating discussions of code cases located in the FSAR or other docketed licensing basis documents.

Tables for Section 1.11

TABLE 1.11-1

ASME CODE CASE REFERENCES

Code Case	**Location of Reference to Code Case**
General	UFSAR Table 1.8, Regulatory Guides 1.84, 1.85 and 1.147 UFSAR Section 5.4.2.1.1
1177	UFSAR Table 1.8, Regulatory Guide 1.57 UFSAR Section 3.8.1.2.3
1330-3	UFSAR Table 1.8, Regulatory Guide 1.57 UFSAR Sections 3.8.1.2.3 and 3.8.2.7
1332-6	UFSAR Table 3.8-8
1355	Response to NRC question 210.44
1401-1	Response to NRC question 210.44
1423-2	UFSAR Table 5.2-2 Response to NRC question 210.44
1484	Response to NRC question 210.44
1493-1	Response to NRC question 210.44
1501	Response to NRC question 210.44
1526	Response to NRC question 210.44
1528	UFSAR Section 5.2.1.2 Response to NRC question 210.44
1552	Response to NRC question 210.44
1553-1	Response to NRC question 210.44
1605	UFSAR Table 1.8, Regulatory Guide 1.65
1606	UFSAR Table 3.9B-10
1607	UFSAR Table 3.9B-10
1618	UFSAR Table 5.2-4
1635	UFSAR Table 3.9B-10
1636	UFSAR Table 3.9B-10
1644	Response to NRC question 210.35
1649	Response to NRC question 210.44
N-3-10 (1335-10)	Response to NRC question 210.44
N-242-1	Response to NRC question 210.44
N-318	UFSAR Sections 3A.3, 3A.3.22.1, 3A.3.23.1 and 3A.3.25
N-392	UFSAR Sections 3A.3, 3A.3.21.1, 3A.3.24.1 and 3A.3.25
N-411	UFSAR Table 1.8, Regulatory Guides 1.61 and 1.122 UFSAR Section 3.7B.3.1.2 UFSAR Table 3.7B-1

1.12 EQUIVALENT MATERIALS

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 FENOC Quality Assurance Program. The Site Design Control program under which the 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. 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).