4.1 SUMMARY DESCRIPTION

Susquehanna Units 1 and 2 are General Electric BWR/4 Boiling Water Reactors. Each reactor contains 764 fuel assemblies and 185 control rods arranged in an upright cylindrical configuration. Light water acts as both moderator and coolant.

The reactor assembly consists of the reactor vessel, its internal components of the core, shroud, steam separator and dryer assemblies, and jet pumps. Also included in the reactor assembly are the control rods, control rod drive housings, and the control rod drives. Figure 3.9-3, shows the arrangement of reactor assembly components. Important design and performance characteristics are discussed in Sections 4.2, 4.3 and 4.4. Loading conditions for reactor assembly components are specified in Section 3.9.

4.1.1 REACTOR VESSEL

The reactor vessel design and description are covered in Section 5.3.

4.1.2 REACTOR INTERNAL COMPONENTS

The major reactor internal components are the core (fuel, channels, control rods, and instrumentation), the core support structure (including the shroud, top guide and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray spargers, and the jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion resistant alloys. All major internal components of the vessel can be removed except the jet pump diffusers, the jet pump risers, the shroud, the core spray lines, spargers, and the feedwater sparger. The removal of the steam dryers, shroud head and steam separators, fuel assemblies, in-core instrumentation, control rods, orificed fuel supports, and control rod guide tubes, can be accomplished on a routine basis.

4.1.2.1 Reactor Core

4.1.2.1.1 General

The design of the boiling water reactor core, including fuel, is based on the proper combination of many design variables and operating experience. These factors contribute to the achievement of high reliability.

A number of important features of the boiling water reactor core design are summarized in the following paragraphs:

(1) The BWR core mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The moderate pressure level characteristics of a direct cycle reactor (approximately 1050 psia) result in moderate cladding temperatures and stress levels.

(2) The low coolant saturation temperature, high heat transfer coefficients, and near-neutral water chemistry of the BWR are significant, advantageous factors in minimizing Zircaloy temperature and associated temperature-dependent corrosion and hydride buildup.

The relatively uniform fuel cladding temperatures throughout the core minimize migration of the hydrides to cold cladding zones and reduce thermal stresses.

(3) The basic thermal and mechanical criteria applied in the design have been proven by irradiation of statistically significant quantities of fuel. The design heat transfer rates and linear heat generation rates are similar to values proven in fuel assembly irradiation.

(4) The design power distribution used in sizing the core represents a worst expected state of operation.

(5) The Framatome critical power methodology for boiling water reactors (References 4.1-12, 4.1-29, 4.1-14, and 4.1-15) is applied to assure that more than 99.9% of the fuel rods are expected to avoid boiling transition for the most severe abnormal operational transient described in Chapter 15. The possibility of boiling transition occurring during normal reactor operation is insignificant.

(6) Because of the large negative moderator density coefficient of reactivity during normal power operation, the BWR has a number of inherent advantages. These are the uses of coolant flow for power maneuvering, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon, in order to follow load.

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability, and by calculations. No xenon instabilities have ever been observed in the test results. All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient of reactivity (Reference 4.1-1).

Important features of the reactor core arrangement are as follows:

(1) The bottom-entry cruciform control rods consist of B_4C in stainless steel tubes (i.e., B_4C rods) only or a combination of B_4C rods and solid hafnium rods surrounded by stainless steel. Control Rods are further described in subsections 4.1.3.2 and subsequent sections.

(2) The fixed in-core ion chambers provide continuous power range neutron flux monitoring. A probe tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range monitors are located in-core and are axially retractable. The in-core location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is further discussed in Subsection 7.7.1.6.

(3) As shown by experience, the operator, utilizing the in-core flux monitor system, can maintain the desired power distribution within a large core by proper control rod scheduling.

(4) The channels (Zircaloy-2, Zircaloy-4, or Z4B) provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods, and protect the fuel during handling operations.

(5) The core is designed to be subcritical at any time in its operating history with any one control rod fully withdrawn.

(6) The selected control rod pitch provides the ability to finely control the power distribution in the assemblies contained in the reactor core. The pitch also allows ample clearance below the pressure vessel between control rod drive mechanisms for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The Susquehanna SES Units utilize a conventional scatter loading with the lowest reactivity bundles placed in the peripheral region of the core. At periodic refueling intervals, each unit will enter an outage. During this time, fuel assemblies are identified to be discharged, new fuel is loaded and fuel assemblies in the reactor core may be "shuffled" to new locations. Therefore, the core loading patterns are both unit and cycle specific. The core configurations for each unit are discussed in Section 4.3.

4.1.2.1.3 Fuel Assembly Description

The fuel assembly is composed of fuel and water rods (or interior water channels), structural components and a fuel channel. The mechanical design of the assembly is described in Section 4.2. The nuclear design of the assembly is described in Section 4.3. Thermal hydraulic design of the assembly is described in Section 4.4.

4.1.2.1.3.1 Fuel Rod

A fuel rod consists of UO_2 pellets and a Zircaloy-2 cladding tube. A fuel rod is made by stacking pellets into a Zircaloy-2 cladding tube which is evacuated and back-filled with helium, and sealed by welding Zircaloy end plugs in each end of the tube.

The BWR fuel rod is designed as a pressure vessel. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod.

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between fuel and clad. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment. Fuel rod design bases are discussed in more detail in Section 4.2.

4.1.2.1.3.2 Fuel Bundle

The fuel bundle has two important design features:

(1) The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.

(2) The unique structural design permits the removal and replacement, if required, of individual fuel rods.

Fuel bundles are designed to meet all the criteria for core performance and to provide ease of handling.

Selected fuel rods in each assembly may differ from the others in initial uranium enrichment, burnable poison content, and fuel rod length. The variation in enrichment and burnable poison distribution produces more uniform power production across the fuel assembly, and thus allows a significant reduction in the amount of heat transfer surface required to satisfy the design thermal limitations. The inclusion of part length fuel rods in the assembly improves the two phase pressure drop, enhances the inherent stability of the bundle, and improves the required shutdown margin of the core design.

Section 4.2 provides a more detailed description of the mechanical design aspects of the fuel bundles in use at Susquehanna.

4.1.2.1.4 Assembly Support and Control Rod Location

All peripheral fuel assemblies are supported by the core plate. Otherwise, individual fuel assemblies in the core rest on fuel support pieces mounted on top of the control rod guide tubes. Each guide tube, with its fuel support piece, bears the weight of four assemblies and is supported by a control rod drive penetration nozzle in the bottom head of the reactor vessel. The core plate provides lateral support and guidance at the top of each control rod guide tube.

The top guide, mounted inside the shroud, provides lateral support and guidance for each fuel assembly. The reactivity of the core is controlled by cruciform control rods which occupy alternate spaces between fuel assemblies. The position of each control rod is controlled by independent mechanical hydraulic drive systems. These systems insert and withdraw the control rod from the bottom of the core and can accurately position its associated control rod during normal operation and yet exert approximately ten times the force of gravity to insert the control rod during the scram mode of operation. Bottom entry allows optimum power shaping in the core, ease of refueling, and convenient drive maintenance.

4.1.2.2 Shroud

The shroud is a cylindrical, stainless steel structure which surrounds the core and provides a barrier to separate the upward flow through the core from the downward flow in the annulus, and also provides a floodable volume in the unlikely event of an accident which tends to drain the reactor pressure vessel. A flange at the top of the shroud mates with a flange on the shroud head and steam separators. The upper cylindrical wall of the shroud and the shroud head form the core discharge plenum. The jet pump diffusers penetrate the shroud support below the core elevation to introduce the coolant to the bottom head volume. The shroud support is designed

to support and locate the jet pumps, core support structure, peripheral fuel assemblies and to separate the inlet and outlet flows of the recirculation loops.

Mounted inside the upper shroud cylinder in the space between the top of the core and the upper shroud flange are the core spray spargers with spray nozzles for injection of cooling water. The core spray spargers and nozzles do not interfere with the installation or removal of fuel from the core.

4.1.2.3 Shroud Head and Steam Separators

The shroud head consists of a flange and dome onto which is welded an array of standpipes, with a steam separator located at the top of each standpipe. The shroud head mounts on the flange at the top of the cylinder and forms the cover of the core discharge plenum region. The joint between the shroud head and shroud flange does not require a gasket or other replacement sealing technique. The fixed axial flow-type steam separators have no moving parts and are made of stainless steel.

In each separator, the steam-water mixture rising from the standpipe impinges on vanes which give the mixture a spin to establish a vortex wherein the centrifugal forces separate the steam from the water. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water exits from the lower end of the separator and enters the pool that surrounds the standpipes to enter the downcomer annulus.

For ease of removal, the shroud head is bolted to the shroud top flange by long shroud head bolts that extend above the separators for easy access during refueling. The shroud head is guided into position on the shroud via guide rods on the inside of the vessel and locating pins located on the shroud head. The objective of the shroud head bolt design is to provide direct access to the bolts during reactor refueling operations with underwater tool manipulation during the removal and installation of the assemblies.

4.1.2.4 Steam Dryer Assembly

The steam dryer assembly is mounted in the reactor vessel above the shroud head and forms the top and sides of the wet steam plenum. Vertical guide rods on the inside of the vessel provide alignment for the dryer assembly during installation. The dryer assembly is supported by pads extending from the vessel wall and is prevented from lifting during postulated transients by brackets welded to the reactor vessel top head. Steam from the separators flows upward into the dryer assembly. Moisture is removed by the dryer vanes and flows first through a system of troughs and pipes to the pool surrounding the separators and then into the downcomer annulus between the core shroud and reactor vessel wall. The steam leaving the top of the dryer assembly flows into vessel steam outlet nozzles which are located alongside the steam dryer assembly. The schematics of a typical steam dryer panel are shown in Figures 4.1-2 and 4.1-3.

4.1.3 REACTIVITY CONTROL SYSTEMS

4.1.3.1 Operation

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of

selected patterns of rods. The rods, which enter from the bottom of the near-cylindrical reactor core, are positioned in such a manner to counter-balance steam voids in the top of the core and which results in significant power flattening.

These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor "scram" (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms which allow either axial positioning for reactivity regulation or rapid scram insertion.

The design of the rod-to-drive connection permits each blade to be attached or detached from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Rods

For the original equipment and Duralife D160-C control rods the neutron absorber portion of the control rod is contained in the wings of the cruciform shaped control blades which are inserted in the bypass region between four fuel assemblies. The original equipment control blades contain boron-carbide (B_4C) powder filled stainless steel absorber tubes. Newer generation control blades contain a combination of B_4C filled tubes and solid hafnium rods. The boron-carbide absorber tubes are seal welded with end plugs on either end. Stainless steel balls are used to separate the tubes into individual compartments. The stainless steel balls are held in position by a slight crimp in the tube. The individual tubes act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction. The tubes or rods are held in a cruciform array by a stainless steel sheath extending the full length of the tubes.

A top handle aligns the tubes and provides structural rigidity at the top of the control rod. Rollers, housed in the handle, provide guidance for control rod insertion and withdrawal. A bottom casting is also used to provide structural rigidity and contains positioning rollers and a parachute-shaped velocity limiter. The handle and lower casting are welded into a single structure by means of a small cruciform post located in the center of the control rod.

Replacement Marathon control rods may use a modified handle assembly that eliminates pins and rollers present in the earlier design.

Marathon control blade wings are made up of an array of square tubes welded together. The tube arrays are welded to center tie rods to form the cruciform blade shape. The square tubes are loaded with either B_4C or Hafnium. The B_4C is contained in separate capsules to prevent migration within the tubes. The square tubes are sealed at each end to prevent the neutron poisons from washing out into the coolant. The blade handle and velocity limiter are equivalent to previous control blade designs, (Reference 4.1-24). The Marathon Ultra – HD Control Rod design was introduced in U2C18, (Reference 4.1-30).

Westinghouse CR 99 control rods, introduced in U1C20, are designed similar to the original equipment and newer style Marathon control rod, (Reference 4.1-31). Like the above GE control rods, the Westinghouse CR 99 control rods have a cruciform blade shape, a handle, a

B_4C loaded absorber zone and a velocity limiter. Different from the aforementioned GE control rods, the Westinghouse CR 99 control rod has horizontal absorber holes drilled into solid stainless steel wings and uses guide pads (buttons) or no guide pads, rather than upper pins and rollers, to guide control rod motion. Reference 4.1-31 provides additional discussion on the design of the CR 99 control rod.

The control rods can be positioned at 6-in. steps and have a nominal withdrawal and insertion speed of 3 in/sec.

The velocity limiter, an engineered safety feature (ESF), is a device which is an integral part of the control rod and protects against the low probability of a rod drop accident. It is designed to limit the free fall velocity and reactivity insertion rate of a control rod so that minimum fuel damage would occur. It is a one-way device, in that control rod scram time is not significantly affected.

Control rods are cooled by the core leakage (bypass) flow. The core leakage flow is made up of recirculation flow that leaks through the several leakage flow paths, which are:

(1) The area between fuel channel and fuel assembly nosepiece;
(2) The area between fuel assembly nosepiece and fuel support piece;
(3) The area between fuel support piece and core plate;
(4) The area between core plate and shroud; and
(5) The bypass flow holes in the fuel assembly nosepiece.

Further details of the control blade design are provided in Section 4.2.

4.1.4 ANALYSIS TECHNIQUES

4.1.4.1 Reactor Internal Components

The following computer codes were used for initial design of the reactor internal components. Code descriptions are provided for historical purposes only.

Computer codes used for the analysis of the internal components are listed as follows:

(1) MASS
(2) SNAP (MULTISHELL)
(3) GASP
(4) NOHEAT
(5) FINITE
(6) DYSEA
(7) SHELL 5
(8) HEATER
(9) FAP-71
(10) CREEP-PLAST

Detailed descriptions of these programs are given in the following sections:

4.1.4.1.1 MASS (Mechanical Analysis of Space Structure)

4.1.4.1.1.1 Program Description

The program, proprietary of the General Electric Company, is an outgrowth of the PAPA (Plate and Panel Analysis) program originally developed by L. Beitch in the early 1960s. The program is based on the principle of the finite element method. Governing matrix equations are formed in terms of joint displacements using a "stiffness-influence-coefficient" concept originally proposed by L. Beitch (Reference 4.1-2). The program offers curved beam, plate, and shell elements. It can handle mechanical and thermal loads in a static analysis and predict natural frequencies and mode shapes in a dynamic analysis.

4.1.4.1.1.2 Program Version and Computer

The GE Nuclear Energy Division is using a past revision of MASS. This revision is identified as revision "0" in the computer production library. The program operates on the Honeywell 6000 computer.

4.1.4.1.1.3 History of Use

Since its development in the early 60s, the program has been successfully applied to a wide variety of jet-engine structural problems, many of which involve extremely complex geometries. The use of the program in the Nuclear Energy Division also started shortly after its development.

4.1.4.1.1.4 Extent of Application

Besides the Jet Engine and Nuclear Energy Divisions, the Missile and Space Division, the Appliance Division, and the Turbine Division of General Electric have also applied the program to a wide range of engineering problems. The Nuclear Energy Division (NED) uses it mainly for piping and reactor internals analyses.

4.1.4.1.2 SNAP (MULTISHELL)

4.1.4.1.2.1 Program Description

The SNAP Program, which is also called MULTISHELL, is the General Electric Code which determines the loads, deformations, and stresses of axisymmetric shells of revolution (cylinders, cones, discs, toroids, and rings) for axisymmetric thermal boundary and surface load conditions. Thin shell theory is inherent in the solution of E. Peissner's differential equations for each shell's influence coefficients. Surface loading capability includes pressure, average temperature, and linear-through-wall gradients; the latter two may be linearly varied over the shell meridian. The theoretical limitations of this program are the same as those of classical theory.

4.1.4.1.2.2 Program Version and Computer

The current version maintained by the General Electric Jet Engine Division at Evandale, Ohio is being used on the Honeywell 6000 computer in GE/NED.

4.1.4.1.2.3 History of Use

The initial version of the Shell Analysis Program was completed by the Jet Engine Division in 1961. Since then, a considerable amount of modification and addition has been made to accommodate its broadening area of application. Its application in the Nuclear Energy Division has a history longer than ten years.

4.1.4.1.2.4 Extent of Application

The program has been used to analyze jet engine, space vehicle and nuclear reactor components. Because of its efficiency and economy, in addition to reliability, it has been one of the main shell analysis programs in the Nuclear Energy Division of General Electric.

4.1.4.1.3 GASP

4.1.4.1.3.1 Program Description

GASP is a finite element program for the stress analysis of axisymmetric or plane two-dimensional geometries. The element representations can be either quadrilateral or triangular. Axisymmetric or plane structural loads can be input at nodal points. Displacements, temperatures, pressure loads, and axial inertia can be accommodated. Effective plastic stress and strain distributions can be calculated using a bilinear stress-strain relationship by means of an iterative convergence procedure.

4.1.4.1.3.2 Program Version and Computer

The GE version, originally obtained from the developer, Professor E. L. Wilson, operates on the Honeywell 6000 computer.

4.1.4.1.3.3 History of Use

The program was developed by E. L. Wilson in 1965 (Reference 4.1-3). The present version in GE/NED has been in operation since 1967.

4.1.4.1.3.4 Extent of Application

The application of GASP in GE/NED is mainly for elastic analysis of axisymmetric and plane structures under thermal and pressure loads. The GE version has been extensively tested and used by engineers in GE.

4.1.4.1.4 NOHEAT

4.1.4.1.4.1 Program Description

The NOHEAT program is a two-dimensional and axisymmetric transient nonlinear temperature analysis program. An unconditionally stable numerical integration scheme is combined with an iteration procedure to compute temperature distribution within the body subjected to arbitrary time- and temperature-dependent boundary conditions.

This program utilizes the finite element method. Included in the analysis are the three basic forms of heat transfer, conduction, radiation, and convection, as well as internal heat

generation. In addition, cooling pipe boundary conditions are also treated. The output includes temperature of all the nodal points for the time instants by the user. The program can handle multitransient temperature input.

4.1.4.1.4.2 Program Version and Computer

The current version of the program is an improvement of the program originally developed by I. Farhoomand and Professor E. L. Wilson of University of California at Berkeley (Reference 4.1-4). The program operates on the Honeywell 6000 computer.

4.1.4.1.4.3 History of Use

The program was developed in 1971 and installed in General Electric Honeywell computer by one of its original developers, I. Farhoomand, in 1972. A number of heat transfer problems related to the reactor pedestal have been satisfactorily solved using the program.

4.1.4.1.4.4 Extent of Application

The program using finite element formulation is compatible with the finite element stress-analysis computer program GASP. Such compatibility simplified the connection of the two analyses and minimizes human error.

4.1.4.1.5 FINITE

4.1.4.1.5.1 Program Description

FINITE is a general-purpose finite element computer program for elastic stress analysis of two-dimensional structural problems including (1) plane stress, (2) plane strain, and (3) axisymmetric structures. It has provision for thermal, mechanical and body force loads. The materials of the structure may be homogeneous or nonhomogeneous and isotropic or orthotopic. The development of the FINITE program is based on the GASP program. (See Subsection 4.1.4.1.3.)

4.1.4.1.5.2 Program Version and Computer

The present version of the program at GE/NED was obtained from the developer
J. E. McConnelee of GE/Gas Turbine

Department in 1969 (Reference 4.1-5). The NED version is used on the Honeywell 6000 computer.

4.1.4.1.5.3 History of Use

Since its completion in 1969, the program has been widely used in the Gas Turbine and the Jet Engine Departments of the General Electric Company for the analysis of turbine components.

4.1.4.1.5.4 Extent of Usage

The program is used at GE/NED in the analysis of axisymmetric or nearly axisymmetric BWR internals.

4.1.4.1.6 DYSEA

4.1.4.1.6.1 Program Description

The DYSEA (Dynamic and Seismic Analysis) program is a GE proprietary program developed specifically for seismic and dynamic analysis of RPV and internals/building system. It calculates the dynamic response of linear structural system by either temporal modal superposition or response spectrum method. Fluid-structure interaction effect in the RPV is taken into account by way of hydrodynamic mass.

Program DYSEA was based on program SAPIV with added capability to handle the hydrodynamic mass effect. Structural stiffness and mass matrices are formulated similar to SAPIV. Solution is obtained in time domain by calculating the dynamic response mode by mode. Time integration is performed by using Newmark's b-method. Response spectrum solution is also available as an option.

4.1.4.1.6.2 Program Version and Computer

The DYSEA version now operating on the Honeywell 6000 computer of GE, Nuclear Energy Systems Division, was developed at GE by modifying the SAPIV program. Capability was added to handle the hydrodynamic mass effect due to fluid-structure interaction in the reactor. It can handle 3-Dimensional dynamic problem with beam, trusses, and springs. Both acceleration time histories and response spectra may be used as input.

4.1.4.1.6.3 History of Use

The DYSEA program was developed in the summer of 1976. It has been adopted as a standard production program since 1977 and it has been used extensively in all dynamic and seismic analysis of the RPV and internals/ building system.

4.1.4.1.6.4 Extent of Application

The current version of DYSEA has been used in all dynamic and seismic analysis since its development. Results from test problems were found to be in close agreement with those obtained from either verified programs or analytic solutions.

4.1.4.1.7 SHELL 5

4.1.4.1.7.1 Program Description

SHELL 5 is a finite shell element program used to analyze smoothly curved thin shell structures with any distribution of elastic material properties, boundary constraints, and mechanical thermal and displacement loading conditions. The basic element is triangular whose membrane displacement fields are linear polynomial functions, and whose bending displacement field is a cubic polynomial function (Reference 4.1-6). Five degrees of freedom (three displacements and two bending rotations) are obtained at each nodal point. Output displacements and stresses are in a local (tangent) surface coordinate system.

Due to the approximation of element membrane displacements by linear functions, the in-plane rotation about the surface normal is neglected. Therefore, the only rotations considered are due

to bending of the shell cross section and application of the method is not recommended for shell intersection (or discontinuous surface) problems where in-plane rotation can be significant.

4.1.4.1.7.2 Program Version and Computer

A copy of the source deck of SHELL 5 is maintained in GE/NED by Y. R. Rashid, one of the originators of the program. SHELL 5 operates on the UNIVAC 1108 computer.

4.1.4.1.7.3 History of Use

SHELL 5 is a program developed by Gulf General Atomic Incorporated (Reference 4.1-7) in 1969. The program has been in production status at Gulf General Atomic, General Electric, and at other major computer operating systems since 1970.

4.1.4.1.7.4 Extent of Application

SHELL 5 has been used at General Electric to analyze reactor shroud support and torus. Satisfactory results were obtained.

4.1.4.1.8 HEATER

4.1.4.1.8.1 Program Description

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program utilizes test data obtained by GE using full scale mockups of feedwater spargers combined with a series of models which represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the nuclear steam supply system are modeled in detail (Reference 4.1-8).

4.1.4.1.8.2 Program Version and Computer

This program was developed at GE/NED in FORTRAN IV for the Honeywell 6000 computer.

4.1.4.1.8.3 History of Use

The program was developed by various individuals in GE/NED beginning in 1970. The present version of the program has been in operation since January 1972.

4.1.4.1.8.4 Extent of Application

The program is used in the hydraulic design of the feedwater spargers for each BWR plant, in the evaluation of design modifications, and the evaluation of unusual operational conditions.

4.1.4.1.9 FAP-71 (Fatigue Analysis Program)

4.1.4.1.9.1 Program Description

The FAP-71 computer code, or Fatigue Analysis Program, is a stress analysis tool used to aid in performing ASME-III Nuclear Vessel Code structural design calculations. Specifically, FAP-71 is used in determining the primary plus secondary stress range and number of allowable fatigue

cycles at points of interest. For structural locations at which the 3Sm (P+Q) ASME Code limit is exceeded, the program can perform either (or both) of two elastic-plastic fatigue life evaluations: 1) the method reported in ASME Paper 68-PVP-3, 2) the present method documented in Paragraph NB-3228.3 of the 1971 Edition of the ASME Section III Nuclear Vessel Code. The program can accommodate up to 25 transient stress states of as many as 20 structural locations.

4.1.4.1.9.2 Program Version and Computer

The present version of FAP-71 was completed by L. Young of GE/NED in 1971 (Reference 4.1-9). The program currently is on the NED Honeywell 6000 computer.

4.1.4.1.9.3 History of Use

Since its completion in 1971, the program has been applied to several design analyses of GE BWR vessels.

4.1.4.1.9.4 Extent of Use

The program is used in conjunction with several shell analysis programs in determining the fatigue life of BWR mechanical components subject to thermal transients.

4.1.4.1.10 CREEP/PLAST

4.1.4.1.10.1 Program Description

A finite element program is used for the analysis of two-dimensional (plane and axisymmetric) problems under conditions of creep and plasticity. The creep formulation is based on the memory theory of creep in which the constitutive relations are cast in the form of hereditary integrals. The material creep properties are built into the program and they represent annealed 304 stainless steel. Any other creep properties can be included if required.

The plasticity treatment is based on kinematic hardening and von Mises yield criterion. The hardening modulus can be constant or a function of strain.

4.1.4.1.10.2 Program Version and Computer

The program can be used for elastic-plastic analysis with or without the presence of creep. It can also be used for creep analysis without the presence of instantaneous plasticity. A detailed description of theory is given in Reference 4.1-11. The program is operative on Univac-1108.

4.1.4.1.10.3 History of Use

This program was developed by Y. R. Rashid (Ref. 4.1-11) in 1971. It underwent extensive program testing before it was put on production status.

4.1.4.1.10.4 Extent of Application

The program is used at GE/NED in the channel cross section mechanical analysis.

4.1.4.2 Fuel Rod Thermal Analysis

Fuel Rod Thermal Design Analyses are performed utilizing the classical relationships for heat transfer in cylindrical coordinate geometry with internal heat generation. Steady state fuel rod thermal-mechanical analyses are performed to assure that fuel rod thermal-mechanical limits (e.g., steady state cladding strain and stress, hydrogen absorption, and, corrosion, etc.) are not exceeded. Abnormal operational transients are also evaluated to assure that the damage limit of 1.0% cladding plastic strain is not violated.

4.1.4.2.1 ATRIUM 11 Fuel Type

Fuel rod analyses for the ATRIUM 11 fuel type were performed with the approved version of the RODEX4 code (Reference 4.1-13). The fuel rod performance characteristics modeled by the RODEX4 code are:

- Gas release
- Radial thermal conduction and gap conductance
- Free rod volume and gas pressure calculations
- Pellet clad interaction (PCI)
- Fuel swelling, densification, cracking, and crack healing
- Cladding creep deformation and irradiation induced growth

RODEX4 performs the power ramping analysis, determines the steady state strain, internal pressure, fuel cladding temperature, corrosion, hydrogen absorption, fuel temperature, and performs the fuel rod creep collapse analysis. This computer code is also used to determine gap conductance for transient analysis. Transient strain and power history are also evaluated using RODEX4.

4.1.4.2.2 ATRIUM-10 Fuel Type

Fuel rod analyses were performed with RAMPEX and approved versions of RODEX2, RODEX2A, and COLAPX codes. The fuel rod performance characteristics modeled by the RODEX2 and RODEX2A codes are:

- Gas release
- Radial thermal conduction and gap conductance
- Free rod volume and gas pressure calculations
- Pellet clad interaction (PCI)
- Fuel swelling, densification, cracking, and crack healing
- Cladding creep deformation and irradiation induced growth

RODEX2 determines the initial conditions for fuel rod power ramping analysis, performed using RAMPEX.

RODEX2A (Reference 4.1-28) determines the steady state strain, internal pressure, fuel cladding temperature, corrosion, hydrogen absorption, fuel temperature, and the fuel rod internal pressure for creep collapse analysis. This computer code is used to determine gap conductance for transient analysis.

Creep collapse analysis is performed using the COLAPX code.

Section 4.2 presents the fuel rod mechanical design and associated methodology.

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in section 4 of Reference 4.1-10. Subsection 4.4.4.6 also provides a complete stability analysis for the reactor coolant system.

Channel and core stability analyses were performed by Framatome, Inc. on a fuel design and cycle specific basis using the STAIF code. A description of the methods employed by the STAIF code is provided in Reference 4.1-20. Using the RAMONA5-FA code (References 4.1-26 and 27), Framatome also performs transient stability analyses in support of the generation of OPRM setpoints.

4.1.4.4 Nuclear Engineering Analysis

A brief summary of principal computer codes used in reactor core design and analysis is provided below.

4.1.4.4.1 CASMO-4

The CASMO-4 computer code (Reference 4.1-25) was developed by STUDSVIK of America to perform steady state modeling of fuel bundles. CASMO-4 uses deterministic transport methods. At the pin cell level it exclusively uses a collision probability method to collapse the energy nuclear data into multi-group data. At the lattice level, it uses a method of characteristics for the neutron equation solution. CASMO-4, as opposed to CASMO-3G, does not need to do pin cell homogenization to perform a 2-D lattice wide transport calculation. The code is used to model each unique fuel lattice in the reactor to calculate few-group cross sections, bundle reactivities and relative fuel rod powers within a fuel bundle. The effects of conditions such as void, control rod presence, moderator temperature, fuel temperature, soluble boron, etc., are included in the model.

4.1.4.4.2 MICROBURN-B2

The MICROBURN-B2 code (Reference 4.1-25) solves a two-group neutron diffusion equation based on an interface current method. It calculates the burnup chain equation for heavy nuclides and burnable poison nuclides, determines the three-dimensional core nodal power distribution, bundle flow, and void distributions. It also determines pin power distributions and thermal margins to technical specification limits. MICROBURN-B2 is used with CASMO-4.

4.1.4.5 Neutron Fluence Calculations

Vessel neutron fluence calculations were performed to determine the azimuthal and axial variation of fluence at the vessel inside surface and at 1/4 T depth. The azimuthal and axial results were synthesized to obtain the fluence profile at the vessel inside surface and 1/4 T depth. The calculations also evaluate vessel fluence at power uprate conditions.

Sections 4.3 and 5.3 provide additional detail regarding reactor pressure vessel irradiation.

4.1.4.6 Thermal Hydraulic Calculations

XCOBRA (References 4.1-21, 4.1-22, and 4.1-23) calculates the steady state thermal hydraulic performance of a BWR. The code determines the flow and local fluid conditions at various axial positions in the core and represents the core as a collection of discrete parallel channels. The only interaction allowed between channels is the equalization of pressure in the inlet and outlet plenums. This is achieved by allowing the core flow to distribute among the various flow channels until the pressure drop in each channel is equalized.

Pressure drop in each channel is determined through the application of two-phase pressure drop correlations and various data which hydraulically characterize the fluid channel. At a given axial position in the core, XCOBRA calculates a core-wide distribution of flow, enthalpy, density, quality, void fraction, and mass velocity.

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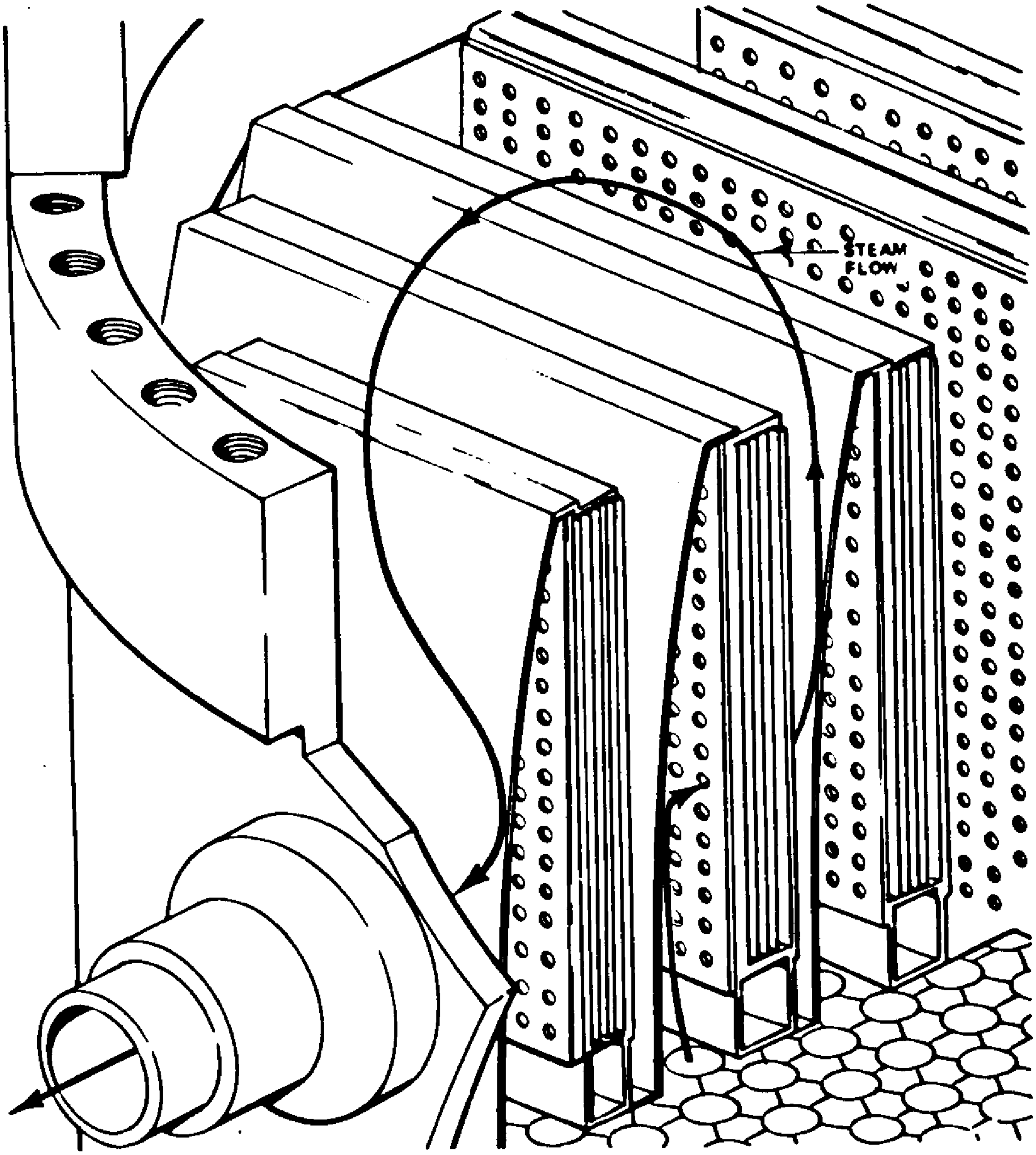

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STEAM DRYER PANEL

FIGURE 4.1-2, Rev. 47

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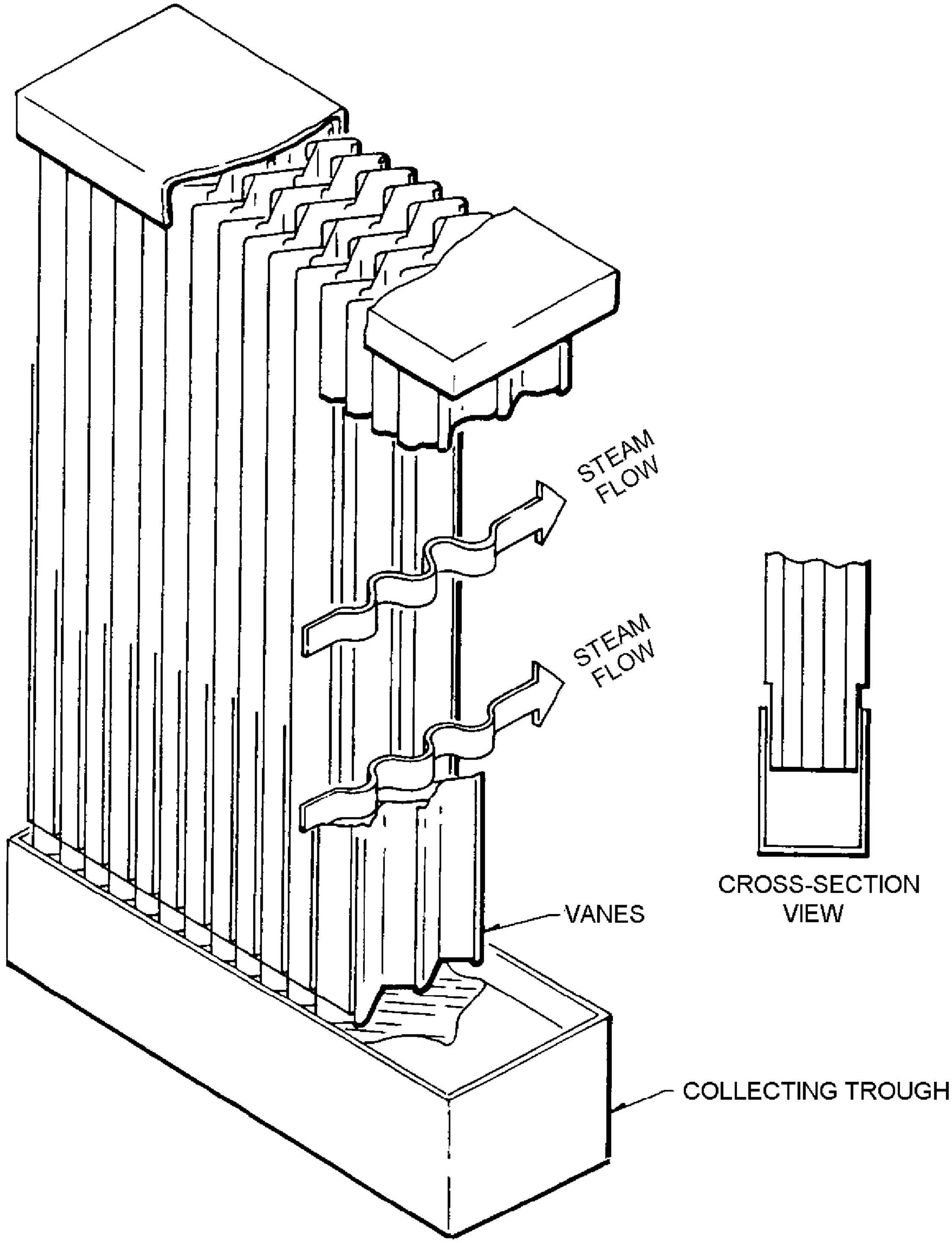

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FIGURE 4.1-3, Rev. 54

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