THERMAL-HYDRAULIC TRANSIENT

ANALYSIS METHODOLOGY

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Safety Analysis Section
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Nuclear Generation Department
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1.0 INTRODUCTION

1.1 Objective

The objective of this report is to present the development and validation of thermal-hydraulic transient analysis methods at Duke Power Company in order to address the requirements of Generic Letter 83-11 "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions" (Reference 1-1). This letter requires that licensees performing their own safety analyses demonstrate their capability and technical competence. In particular, comparisons of computer code results to experimental data, plant operational data, or other benchmarked analyses, were identified as areas of interest. This report provides the details of extensive benchmarking efforts which utilize actual plant transient data from the Oconee, McGuire, and Catawba Nuclear Stations for comparisons to system code predictions. The capabilities of the RETRAN-02 system simulation code (Reference 1-2) and the VIPRE-01 core thermal-hydraulic simulation code (Reference 1-3) are demonstrated using plant and core simulation models developed by Duke Power Company.

1.2 RETRAN-02 Code Description

RETRAN-02 was developed by Energy Incorporated for the Electric Power Research Institute (EPRI) to provide utilities with a code capable of simulating most thermal-hydraulic transients of interest in both PWRs and BWRs. RETRAN-02 has the flexibility to model any general fluid system by partitioning the system into a one-dimensional network of fluid volumes and connecting flowpaths or junctions. The mass, momentum, and energy equations are then solved by employing a semi-implicit solution technique. The time step selection logic is based on algorithms that detect rapid changes in physical processes and limit time steps to ensure accuracy and stability. Although the equations describe homogeneous equilibrium fluid volumes, phase separation can be modeled by separated bubble-rise volumes and by a dynamic slip model. The pressurizer and other volumes can be modeled as non-equilibrium volumes when such phenomena are present. Reactor power generation can be represented by either a point kinetics model or a one-dimensional kinetics model. Heat transfer across steam generator tubes and to or from structural components can be modeled. Special component models for centrifugal~~centriugual~~ pumps, valves, trip logic, control systems, and other features useful for fluid system modeling are available. The RETRAN-02 MOD003 code version is used for the analyses presented in this report.

1.3 VIPRE-01 Code Description

VIPRE-01 was developed for EPRI by Battelle Pacific Northwest Laboratories for steady-state and transient core thermal-hydraulic analysis. The basic structure and computational philosophy of the VIPRE-01 code are derived from COBRA-IIIC (Reference 1-4). The subchannel analysis approach is applied in both codes. With this approach the nuclear fuel element is divided into a number of quasi-one-dimensional channels that communicate laterally by diversion crossflow and turbulent mixing. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flowrate, lateral flow per unit length, and momentum pressure drop. The flow field is assumed to be incompressible and homogeneous, although models are added to reflect subcooled boiling and co-current liquid/vapor slip. VIPRE uses an implicit boundary value solution scheme where the boundary conditions are inlet enthalpy, inlet mass flowrate, and core exit pressure. The VIPRE-01 Cycle-01 code version is used for the analyses presented in this report.

1.4 ~~Methodolgy~~Methodology Development

The development of inhouse plant transient simulation capability, which has evolved into the submittal of this report, began in April 1978. Initial efforts focused on following the development of the RETRAN-01 system simulation code (Reference 1-5) by EPRI. Following the first release of a production version of RETRAN-01 in December 1978, work began on assembling a simulation model of the Oconee Nuclear Station and was completed in July 1979. The Oconee Nuclear Station is a three unit site with similar 2568 MWt Babcock & Wilcox pressurized water reactors. The Oconee RETRAN model was then exercised during the next year by comparison to several plant transient events (References 1-6, 1-7), as well as some separate effects tests conducted at the plant. Based on the generally positive results of these initial transient simulation efforts, it was decided in mid-1980 to begin applications of the technology towards the resolution of technical and safety concerns. Additional Oconee RETRAN model comparisons to plant transients are described in References 1-8 and 1-9.

A separate and parallel effort was initiated in June 1979 to develop core thermal-hydraulic analysis technology. Although most of this effort was directed towards steady-state core reload design, models for predicting the departure from nucleate boiling phenomenon during transients were also developed. The early transient analysis applications utilized the

subchannel conditions and DNBR are presented in Section 2. Additional VIPRE validation by code comparison has been submitted by Duke Power Company in Reference 1-14 for Oconee and in Reference 1-15 for McGuire/Catawba.

1.6 Quality Assurance

The development, utilization, and documentation of transient analysis technology incorporated several stages of formal quality assurance (QA) activities. The major activities are controlled by formal QA procedural requirements as part of the Duke Power Company - Design Engineering Department Quality Assurance Manual. Other activities are administratively controlled by workplace procedures or by training that serves to maintain a high level of consistency in the application of the codes and models.

A major QA activity is the certification of computer codes to be used in safety-related analyses. The requirements for certifying a code are detailed in the Duke Power Company procedures PR-101, "Engineering Calculations," and MPR-102, "Nuclear Engineering Section Safety Related Analysis." Requirements include verification of technical accuracy, periodic testing, restricted access, measures to ensure backup capability via storage of duplicate source tapes in a secured vault, and documentation including a user manual, input description, job execution instructions, and sample input and output listings. Both the RETRAN-02 and VIPRE-01 code versions used for the analyses documented in this report have undergone this certification process.

A second major QA activity is the documentation of the simulation model. Due to the very large volume of information that is necessary to develop a model for a system simulation code such as RETRAN-02, a separate document is compiled to detail all calculations and references utilized in the model. These model documents describe a "base deck" which consists of all the code input necessary to initialize at 100% full power with all parameters at nominal conditions. A thorough review of the model document is performed along with a review of the derived input listing. The model document and the base deck are then controlled such that any changes must be documented, reviewed, and approved prior to implementation. This process is described in the workplace procedure, "Changes to Computer Code Base Models." Applications of the base deck with analysis-specific modifications are performed such that the base deck itself is not modified. Modifications are added at the end of the base deck so that the QA review can be limited to the analysis~~anlaysis~~ -specific additions. A model document is not developed for the VIPRE-01 code models since the volume of calculations necessary to develop the model and

used for a wide range of purposes. The most pertinent applications in the context of this report are those related to the resolution of licensing concerns. Licensing concerns include:

- Evaluation of the consequences of equipment failures and other items for documentation in LERs.
- Evaluation of the impact of proposed plant modifications, changes in Technological Specifications, and revisions to operating procedures on the design basis transients and accidents
- Reanalysis of design basis transients due to changes in plant parameters, such as those associated with a fuel reload
- Resolution of generic safety issues applicable to Duke nuclear stations
- Analytical basis for justification of continued operation under off-normal operating conditions

Other applications of the technology include:

- Analytical basis for Emergency Procedure Guidelines
- Data for validation of plant-specific control room simulators
- Developing responses to station concerns regarding plant transients
- Data for emergency drills
- Success criteria for PRA systems analysis

Based on a foundation of thorough analytical model development and substantial model benchmarking efforts, the capability to employ methodology applications towards the resolution of technical and safety concerns has been demonstrated.

1.8 References

1-1 Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (Generic Letter No. 83-11), USNRC, February 8, 1983

1-2 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM Revision 2, EPRI, November 1984

1-3 VIPRE-01: A Thermal-**Hydraulic**~~Hydraluic~~ Code for Reactor Cores, EPRI NP-2511-CCM Revision 2, EPRI, July 1985

1-4 COBRA-IIIC: A Digital Computer Program for Steady-State and Transient Core Thermal-Hydraulic Design Methodology, BNWL-1695, PNL, March 1973

1-5 RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI CCM-5, EPRI, December 1978

pump discharges. In addition there are many penetrations for RCS instrumentation such as temperature, pressure, and flow.

The high point of the primary system is located at the bend of the hot leg, before the pipe enters the SGs. This bend is commonly **referred**~~refered~~ to as the "U-bend" or "candy cane." This feature is different from the Westinghouse and Combustion Engineering PWR design, in which the high point in the primary system loop is located at the top of the SG tubes.

2.1.2.4 Reactor Coolant Pumps

Each unit has four RCPs. Unit One has Westinghouse Model 93A pumps, while Units 2 and 3 have Bingham Type RQV pumps. Both types are centrifugal pumps which operate at a constant speed, and both utilize 9000 hp Westinghouse motors. The hydraulic characteristics of the pumps are similar, but the Bingham pumps provide approximately 5% more flow. The Westinghouse pump seals are a hydraulic controlled-leakage design, while the Bingham pumps use mechanical seals.

The units are designed for operation with fewer than four pumps operating. With three pumps operating the maximum power level is 75%. With two pumps operating (one per loop) the maximum power level is 50%. Power operation with two inactive pumps in one loop is prohibited.

2.1.2.5 Steam Generators

The two once-through SGs provide for energy removal from the primary system. The primary side of a SG consists of the upper head, the upper tubesheet, the tubes, the lower tubesheet, and the lower head. Primary coolant enters the SG upper head through a nozzle connected to the hot leg piping. The coolant flows down through the 52 foot long SG tubes into the SG lower head. Two nozzles connect the lower head to the cold legs. The SG upper and lower heads are made of carbon steel clad with stainless steel. The tubesheets are also carbon steel.

The Inconel-600 tubes are fixed at the upper and lower ends by the two foot thick tubesheets, which separate the primary and secondary sides. There are

is no attempt to completely model the bends, flowpaths, etc., which make up the main steam lines. The loss coefficient of the SG outlet (Junction 138) is adjusted to give the nominal 25 psi pressure drop between the SG and the high pressure turbine. [ ] represents the flow to the high pressure turbine. [ ] models the steam relief function of the TBVs. The two TBVs on each steam line are [ ] simulate the eight MSRVs; the two sets of valves which [ ]

2.2.3 Heat Conductor Nodalization

2.2.3.1 Reactor Core

[ ] conductors are used to model the fuel pins in the reactor core. The conductors are separated into [three regions: the UO_2 fuel pellets, the gap, and the Zircaloy cladding.] Material properties (thermal conductivity and heat capacity) are [ ] The fuel gap [ ] to give the [ ] fuel temperature, which varies with core average burnup. This approach is used to properly account for the stored energy in the fuel. The RETRAN core conductor model is used for these conductors in order to allow power generation in the fuel material. 2.7% of the power generated in the core is assigned to direct heating of the moderator rather than deposition of energy in the fuel pellets.

2.2.3.2 Steam Generators

[ ] heat conductors are used to represent the tubes in each of the SGs. The [ ] conductors to represent the remaining length of the tubes. The nominal Inconel-600 heat capacity is used, but the [ ]

2.2.3.3 Structural Conductors

These conductors represent the plant components which do not generate power or conduct heat from the primary to the secondary, but which can affect the plant transient response by transferring energy to or from the working fluid. The stored energy and heat capacitance of these conductors tend to dampen changes in RCS conditions. During an overcooling event the structural conductors transfer heat to the primary coolant and thus retard the cooldown. Conversely, during an overheating transient the structural conductors act as a heat sink and reduce the magnitude of the increase in the primary coolant temperature. The effect of the structural conductors is most apparent during long-term transients. During short transients which do not exhibit severe undercooling or overcooling, the heat transferred from the structural conductors is insignificant relative to the large amount of decay heat in the core. However, the structural conductors represent a significant heat load for long-term cooldown, once decay heat has decreased.

The key parameters for the structural conductors are the mass and the heat transfer area. These determine the initial stored energy and the effectiveness as a heat sink. In order to [ ]

structural conductors are modeled as [ ] the coolant. Those structures which are [ ] conductor.

Certain structural components are not included in the model because they are considered to have no potential impact on a plant transient. These components include [ ]

Passive heat conductors representing the pressurizer walls[ ]in the Oconee model. [ ]**The pressurizer vessel metal is** [ ]

The conductors which are used in the Oconee base model are listed and described in Table 2.2-1.

2.2.4 Control System Models

2.2.4.1 Process Variable Indications

RETRAN control systems are used to take the calculated plant thermodynamic conditions and put them into the form in which they are output by the plant instrumentation. This provides indications which are useful for comparison to plant data and which are familiar to the plant operators and engineering personnel.

RCS Pressure

The fluid pressure at the elevation of the hot leg pressure tap is converted to gauge pressure by subtracting 14.7 psi. This pressure is an input to RPS and ES functions in the model.

Pressurizer Level

~~The cross-sectional area of the RETRAN pressurizer volume is different than that of the plant pressurizer. This is due to the fact that the plant pressurizer is a right circular cylinder with a hemispherical top and bottom section, while RETRAN volumes are simple right circular cylinders. Therefore, a control system is used to relate the pressurizer liquid level in the model to the level that would be indicated at the plant.~~**The level indication is derived from the pressure difference between two taps in the pressurizer. The pressure difference is converted to an equivalent water level, which is then**

converted to a 0-100% reading. A RETRAN control system is used to simulate this process using the [ ]. This level is input to the interlock which turns off the pressurizer heaters on low pressurizer level.

Hot Leg Temperature

The hot leg temperature is indicated by RTDs located in thermowells in the hot leg piping. A change in fluid temperature is not indicated immediately at the plant due to the time required to transfer heat through the thermowell to the RTD and change the temperature of the measuring device. Experimental data indicates that the time delay can be approximated by a [ ] hot leg fluid temperature. A RETRAN control system is used to apply this [ ] to the hot leg water temperature to obtain the transient temperature response that would be seen at the plant. The output is used in the high temperature and variable low pressure reactor trips.

SG Pressure

In a manner similar to the RCS pressure indication, the fluid pressure at the outlet of the steam generator is adjusted by a RETRAN control system to output pressure in psig. This pressure is used as an input to the TBV controller.

SG Level

The operate range SG level instrument displays level from 0-100%, as shown on Figure 2.1-10. The level indication is derived from the pressure difference between two taps in the steam generator. The pressure difference is converted to an equivalent water level, which is then converted to a 0-100% reading. A RETRAN control system is used to simulate this process using the [ ] The output operate range level indication performs no control or trip function in the RETRAN base model.

]

2.2.6.2 Centrifugal Pumps

The RETRAN centrifugal pump model is used to simulate the performance of the RCPs. For benchmark analyses the pump input data is [ ] for other applications, the flow is typically specified to be ~~106.5%~~112.5% of 88,000 gpm per pump, and the head is taken to be that of the [

]

2.2.6.3 Valves

The basic RETRAN valve model is used for most of the valves in the Oconee base model. With this model the valves open and reseat according to the action of their associated trips. Modeled in this manner are the pressurizer spray, PORV, and safety valves; the core flood tank discharge check valves; the turbine stop valves; and the MSRVs. The turbine bypass valves use the RETRAN valve model option in which the junction area is controlled by a control system, which opens or shuts the valves based on SG pressure. The reactor vessel vent valves are also controlled by a control system, with the [ ]

2.2.6.4 Phase Separation

RETRAN has two methods of modeling phase separation within a fluid volume: the bubble rise model and slip. The bubble rise model is [ ] in the Oconee model, while slip [ ] used.

The bubble rise model is a correlation which allows the enthalpy in a volume to vary with height. It is a semi-empirical fit to data from a number of high pressure blowdown experiments. The void fraction in the volume is assumed to vary linearly with height from the bottom of the volume to the mixture level.

Above the mixture level, the fluid is 100% steam. The model is [ ] which have a definite separation between vapor and liquid, i.e. the [ ] In addition, the bubble rise model is [ ]

Phase separation is not normally expected in the [ ] because it usually remains in a subcooled state. In some cases, however, voids may develop in the [ ] The bubble rise model is [ ]

Slip models provide for unequal velocities between the liquid and vapor phases. Since Oconee is a PWR with subcooled water in the primary coolant loops, unequal phase velocities normally exist only in the steam generator secondary side. RETRAN has two slip models: algebraic slip and dynamic slip. The algebraic slip model uses a drift flux approach to calculating the relative velocity between the vapor and liquid phases, while the dynamic slip model uses a differential equation to determine the interphase velocity difference. Current RETRAN development efforts are geared toward improving the dynamic slip model and providing a true two-phase representation of transient fluid behavior. Extensive testing has shown that the current dynamic slip model [ ] in the base model.

For applications in which there is significant voiding and phase separation in the primary system (notably small break LOCA or extended loss of feedwater), the dynamic slip model can provide a reasonable simulation of two-phase phenomena in the RCS. In these instances the dynamic slip option [ ]

Use of the non-equilibrium option enhances the ability of a one volume pressurizer model to simulate the transient response during an insurge. Since the liquid region is considered separately from the steam bubble, an insurge does not result in rapid condensation of the pressurizer vapor. The non-equilibrium option allows the steam bubble to superheat during an insurge. The non-equilibrium option also includes the ability to specify a heat transfer coefficient between the pressurizer vapor and liquid regions, so interphase heat transfer can be modeled. However, this model is somewhat non-mechanistic, since the heat transfer coefficient is user-input rather than being calculated based on fluid conditions. ~~Furthermore, the ability to model~~ [ ] ~~in the heat transfer model which are mentioned in Section 2.2.3.3.~~

Another facet of the non-equilibrium representation is the pressurizer spray junction model. Using the spray junction option causes the spray water to condense steam while moving through the pressurizer steam bubble, thus removing both energy and mass from the region, rather than simply mixing with the fluid in the vapor region and desuperheating the steam. The spray junction model is used since it is considered to be more mechanistic than a normal junction for this application.

The non-equilibrium pressurizer model is used for best estimate safety analysis for Oconee. This model does not fully account for condensation effects in the pressurizer steam space and thus overpredicts RCS pressure during an insurge. However, it is superior to an equilibrium modeling approach. For some applications it is appropriate to use [ ]

Unit ~~3~~1, Cycle ~~8~~11, which is considered to be typical of current cores. None of the differences between the units has a significant effect on the plant transient response. The variation of reactor kinetics parameters with cycle and time-in-cycle can have a significant impact if the transient does not begin with reactor trip.

2.2.9 Summary of Experience

The major positive conclusions concerning the application of the code and its models are listed below.

- The basic constitutive equations accurately describe the fluid behavior in the RCS and the SGs during operational transients.

- The nodalization scheme is extremely flexible, allowing the user to construct a detailed plant model or to conduct separate effects analyses on components such as the pressurizer or the core flood tanks. This flexibility has also enabled the modeling of other plant systems, including HPI, EFW, and the condensers.

- The heat transfer package provides a good representation of heat transfer, both single phase and two-phase.

- The water properties are accurate in the range of application.

- Steady-state initialization greatly simplifies the process of obtaining a desired set of initial conditions when compared to other thermal-hydraulic systems analysis codes.

- Iterative numerics generally provides reasonable time step control and reduces the necessity of restarting jobs to circumvent time step-related errors.

- The generalized restart and reedit capabilities of RETRAN are very useful, and they significantly increase the efficiency with which the code is used.

- The lack of a general non-equilibrium modeling capability detracts from the ability of the code to simulate some small break LOCA behavior. This limitation must be recognized whenever such applications are undertaken.

- ~~The current pressurizer model does not~~ [

  ] ~~This limitation can be compensated for by careful modeling in order to ensure a conservative simulation.~~

In general, the overall experience with modeling the Oconee transient response using RETRAN has been good. Despite the shortcomings in the above areas, the code has been proven capable of accurately simulating the transient thermal-hydraulic behavior of a B&W 177 fuel assembly lowered-loop plant.

Table 2.2-1

Oconee Base Model Heat Conductors

Conductor Number	Adjacent Volume Number	Description	Material

Table 2.2-1 (continued)

Conductor Number	Adjacent Volume Number	Description	Material

The range of applicability of the Bowring (WSC-2) correlation is:

Pressure, psia	500 to 2300
Mass flux, 10^6 lbm/hr-ft^2	0.02 to 3.0
Quality (equilibrium)	-0.2 to 0.86

Justification will be provided as necessary, when applying these correlations in future analyses.

Other CHF correlations that have been reviewed and approved by the NRC may also be used to perform DNBR analyses.

2.3.3.2 Conservative Factors

Conservative factors are introduced in all the transient analyses as explained below.

Reduction of the Hot Channel Flow Area

The hot subchannel flow area is reduced by 2% to account for variations in as-built subchannel flow areas (Reference 2-7).

Reduction of the Hot Assembly Flow

Based on the vessel model flow tests and Oconee core pressure drop measurement, the reductions in the hot assembly flow due to flow maldistribution are shown on the next page (Reference 2-9).

Operation	Flow reduction factor
4-pump	0.950
3-pump	0.882
2-pump	0.866

addition there are many penetrations for Reactor Coolant System instrumentation such as temperature, pressure, flow, and level. The high point of the primary loops is the top of the steam generator tubes.

3.1.2.4 Reactor Coolant Pumps

Each unit has four Westinghouse Model 93A reactor coolant pumps (RCPs). These are centrifugal pumps which operate at a constant speed and utilize 7000 hp Westinghouse motors. The pump seals are of a hydraulic controlled-leakage design. Within the discharge nozzle of each pump is a weir plate completely blocking [ ] inches of the circular flow channel into the cold leg piping. This prevents safety injection water which has accumulated in the bottom of the cold leg from flowing back through the pump and blocking the loop seal in the pump discharge piping during a LOCA.

3.1.2.5 Steam Generators

Four recirculating steam generators (SGs) provide for energy removal from the primary system. The primary side of a SG consists of the inlet plenum, the tubesheet, the tubes, and the outlet plenum. Primary coolant enters the SG inlet plenum through a nozzle connected to the hot leg piping. The coolant flows up and down the U-shaped SG tubes into the SG outlet plenum. A nozzle connects the outlet plenum to the pump suction piping. The SG inlet and outlet plena are made of carbon steel clad with stainless steel. The tubesheet is also carbon steel.

The preheat SGs consist of Inconel-600 tubes that are fixed at the ends by the 21 inch thick tubesheet, which separates the primary and secondary sides. There are approximately 4600 tubes per SG. Each tube has a nominal OD of 3/4 inches and a thickness of 0.043 inches.

The preheat SGs used at McGuire and Catawba are of two basic types. Catawba Unit 2 has the counter flow D5 design shown in Figure 3.1-7. The other three units have the split flow D2/D3 design shown in Figure 3.1-8. Differences between the two designs are discussed in Section 3.1.6.

The preheat steam generators at McGuire Units 1 and 2 and Catawba Unit 1 will be replaced with new feedring steam generators manufactured by Babcock & Wilcox International. The design of the feedring steam generators (FSGs) is shown in Figure 3.1-12. There are 6633 tubes per FSG that are made of Inconel-690. The tubes are fixed at the ends by the 27 inch thick tubesheet which is made of carbon steel and clad with stainless steel. Each tube has a nominal OD of 0.6875 inches and a thickness of 0.04 inches. Differences between the preheat and feedring designs are discussed in Section 3.1.6.

3.1.2.6 Pressurizer

The pressurizer is a vertical cylindrical vessel with hemispherical upper and lower heads. A surge line penetrates the bottom of the pressurizer and connects it to one of the hot legs. The pressurizer maintains and controls the RCS pressure and provides a steam surge chamber and liquid water reserve to compensate for changes in reactor coolant density during operation. A diagram of the pressurizer is shown on Figure 3.1-9.

There are four banks of electric heaters in the lower region of the pressurizer, with a total capacity of 1800 kW. These heaters make up for ambient heat losses during normal operation and restore pressure during operational transients. There is a low level interlock which prevents the heaters from being damaged due to uncovery during operation.

The pressurizer spray lines connect two of the cold legs to the pressurizer spray nozzle, which is located at the top of the steam space. Spray valve position is modulated by a proportional plus integral controller providing a maximum of approximately 900 gpm of colder water to the top of the pressurizer where it condenses steam, thus reducing pressure.

The three pressurizer PORVs are CCI drag valves located on lines connected to the top of the pressurizer. Each valve has a 210,000 lbm/hr steam relief capability and opens when RCS pressure exceeds approximately 2335 psig.

The three McGuire pressurizer code safety valves are 2.15 inch Crosby valves which also relieve fluid from the top of the pressurizer. The three Catawba code safety valves are 2.25 inch Dresser valves. The total relief capacity of

the valves at each station is greater than 1,200,000 lbm/hr steam. These spring-loaded valves are set to relieve at 2485 psig.

3.1.2.7 Charging and Letdown

Normal charging at McGuire and Catawba is provided by a centrifugal charging pump (CCP) drawing water from the volume control tank. A control valve in the charging line modulates to control pressurizer level at the programmed setpoint, which is a function of reactor coolant average temperature. Makeup capacity through this flowpath is approximately 140 gpm at nominal system pressure. The makeup capacity can be augmented by starting a parallel CCP, opening the Engineered Safeguards injection flowpath, which is parallel to the charging flowpath, or both. Normal charging injects into the A cold leg piping. An alternate charging line injects into the D cold leg piping.

A small amount of makeup is also provided by RCP seal injection. Approximately 8 gpm is pumped into the seals of each pump, most of which enters the primary system, and the remainder of which returns via the seal leakoff pathway to the volume control tank. Seal injection is provided by the same CCP which furnishes normal charging.

Letdown is taken from the C loop pump suction piping through heat exchangers and demineralizers to the volume control tank. Normal letdown flow is approximately 75 gpm.

3.1.2.8 Instrumentation

A large number of instruments monitor the primary system in order to provide information to the operators, inputs to the plant control systems, and signals for the actuation of the Reactor Protection System (RPS) and the Engineered Safety Features Actuation System (ESFAS). Core instrumentation includes neutron power indication (ionization chambers), movable incore neutron detectors, and core-exit thermocouples. RCS temperatures are measured by resistance temperature detectors (RTDs) in the hot leg and pump suction piping ~~and in bypass loops around the steam generators and reactor coolant pumps~~. Loop flow is measured by elbow taps in each pump suction leg. Pressure is measured by pressure taps in two of the four hot legs (B and C at Catawba, C

and D at McGuire). The pressurizer contains water level, pressure, and water temperature instruments. In addition, inadequate core cooling instrumentation includes a static level measurement for the reactor vessel from top to bottom and a dynamic pressure drop measurement for bulk void fraction indication.

3.1.3 Secondary System

McGuire and Catawba use a regenerative-reheat Rankine cycle to convert the thermal energy produced in the reactor core to electric power. Energy is removed from the primary system by feedwater boiled in the SGs. The steam is exhausted through a high pressure turbine, moisture separator-reheaters, and three low pressure turbines to the condensers. Hotwell pumps take suction from the condenser hotwells and discharge to the condensate booster pumps. The condensate passes through G and F feedwater heaters upstream of the booster pumps and then through E, D, and C feedwater heaters to the suction of the steam-driven main feedwater (MFW) pumps. The MFW pumps discharge through the B and A feedwater heaters to the SGs.

3.1.3.1.1 Preheat Steam Generators

The SGs remove energy from the primary system during normal operation, at hot standby, and if necessary at hot shutdown. A typical generator is shown in Figures 3.1-7 and 3.1-8. At full power most of the approximately 3.8 million lbm/hr feedwater enters each SG preheater through the 16 inch lower nozzle. The downcomer consists of the annular section in the lower part of the SG which is separated from the SG shell region by the cylindrical wrapper. Recirculated water flows under the wrapper and into the bundle region surrounding the U-tubes containing the primary coolant. Water emerging from the preheater region mixes with the recirculation flow in the bundle region. Heat transferred from the U-tubes boils some of the secondary fluid in the bundle region, and the resulting two phase mixture enters the primary and secondary separators. In the separators the steam is dryed to a minimum quality of 0.9975 before passing through the outlet nozzle into the steam line. The separated liquid collects in the downcomer. The nominal SG outlet pressure at full power operation is 1000 psia.

Tube support plates provide structural support for the SG U-tubes. The plates are distributed axially along the tube bundle and are more closely spaced near the bottom. They have clearance holes through which the U-tubes pass. In addition there are circulation holes in the plates to allow fluid to pass up the tube bundle at higher flow rates. Each tube bundle has a lane under the bend apex at the top of the tube bundle. This lane allows some of the interior tubes to be inspected. In addition there are untubed regions through which vertical stayrods pass. These stayrods connect the tube support plates for additional support. The height of the tallest U-tube is approximately 28 feet above the top of the tubesheet.

The elevations of the top and bottom of the reactor core are 155" and 299", respectively, below the top of the lower tubesheet of the SG. In order to promote stable natural circulation flow the thermal center for heat removal must be above the thermal center for heat addition to the primary system. This condition is therefore automatically satisfied because of loop geometry.

The SGs have an upper nozzle to allow auxiliary feedwater (AFW) to be injected into the downcomer above the tubes.

3.1.3.1.2 Feedring Steam Generators

The feedring steam generator (FSG) is shown in Figure 3.1-12. At full power, most of the approximately 3.8 million lbm/hr feedwater is delivered to the feedring through the main feedwater nozzle and gooseneck. The 32 J-tubes connected to the feedring distribute the feedwater axisymmetrically around the downcomer, where the feedwater mixes with the recirculation flow. The downcomer consists of the annular section in the lower part of the FSG which is separated from the shell region by the cylindrical wrapper. Recirculated water flows under the wrapper and into the bundle region surrounding the U-tubes containing the primary coolant. Heat transferred from the U-tubes boils some of the secondary fluid in the bundle region, and the resulting two-phase mixture enters the primary and secondary separators. In the separators, the steam is dried to a minimum quality of 0.9975 before passing through the steam outlet nozzle into the steam line. The separated liquid collects in the downcomer. The nominal FSG outlet pressure at full power is 1020 psia.

A lattice bar grid arrangement provides structural support for the U-tubes while minimizing resistance to fluid flow. The lattice grids are distributed axially along the tube bundle, with one high resistance lattice grid at the bottom of the bundle and eight low or medium resistance lattice grids above the high resistance lattice grid. The height of the tallest U-tube is approximately 35 feet above the top of the tubesheet.

The elevations of the top and bottom of the reactor core are 170" and 314", respectively, below the top of the tubesheet of the FSG. Thus, the difference in thermal centers promotes stable natural circulation flow.

The FSGs have an auxiliary feedwater nozzle approximately 3 feet above the main feedwater J-tubes to allow auxiliary feedwater to be injected into the downcomer above the tubes.

3.1.3.2 Main Feedwater

The MFW System consists of the MFW pumps, the A and B feedwater heaters, and the piping and valves between the pumps and the SGs. The MFW pumps have common suction and discharge lines, so neither of the two pumps is aligned to particular SGs. The variable-speed pumps are turbine-driven by either main steam or low pressure steam. The nominal feedwater temperature at the outlet of the A feedwater heaters is 440°F, at a pressure of approximately 1100 psia. MFW flow to each SG is controlled by the MFW control valves.

The MFW flow is normally aligned predominantly to the lower nozzle during power operation. At low power levels MFW is swapped to inject into the upper nozzle. AFW is aligned only to the upper nozzle.

For the FSGs, main feedwater flow is normally aligned to the main feedwater nozzle during power operation. It is not expected that main feedwater will be swapped to inject into the auxiliary feedwater nozzle at lower power levels for FSG operation at McGuire. For FSG operation at Catawba, MFW is swapped to inject into the upper nozzle at lower power levels. Auxiliary feedwater is aligned only to the auxiliary feedwater nozzle.

3.1.3.3 Main Steam

The main steam lines carry the high pressure, high temperature steam from the SGs to the high pressure turbine. One 32" line exits each SG and expands to a 34" line. The 34" line leaves the Reactor Building and enters the Doghouse. Inside the Doghouse there is a main steam isolation valve (MSIV) on each line. Downstream of the MSIV each line leaves the Doghouse, goes across the yard and enters the Turbine Building. From then on the configuration is station specific and is discussed in Section 3.1.6.4.

Process steam is taken off of the steam headers to power station auxiliaries. These include the auxiliary steam header, the MFW pumps, the turbine-driven AFW pump, the condensate steam air ejectors, and the steam seals. In addition, main steam is used to reheat the steam between the high and low pressure turbines. Various steam drains and traps are also provided on each steam line. Main steam relief is provided by five steam line safety valves and one Power Operated Relief Valve (PORV) per steam line. Downstream of the MSIVs, further steam relief is provided by condenser dump valves and atmospheric dump valves.

The steam line safety valves provide overpressure protection to the steam lines and SGs. The valve opening setpoints range between 1170 and ~~1225~~**1220** psig~~ at McGuire and are 5 psi higher at Catawba~~. The total relief capacity through the valves is greater than the nominal full power steam flow rate. The condenser dump valves control steam pressure prior to putting the turbine on-line and after turbine trip. The nine valves have a total capacity of 40% of nominal full power steam flow. The atmospheric dump valves provide additional steam relief for load rejection transients. These valves have a total capacity of 45% of nominal steam flow. The two sets of valves, together with the steam line PORVs, are designed to allow a full load rejection without tripping the reactor or opening the steam line safety valves.

3.1.3.4 Turbine-Generator

The turbine-generator converts the thermal energy of steam produced in the SGs into mechanical shaft power and then into electrical energy. The turbine-generator of each unit consists of a tandem (single shaft) arrangement of a

of a double-flow high pressure turbine and three identical double-flow low pressure turbines driving a direct-coupled generator at 1800 rpm. Turbine-generator functions under normal and abnormal conditions are monitored and controlled automatically by the Turbine Control System, which includes redundant mechanical and electrical trip devices to prevent excessive overspeed of the turbine generator. Once the turbine is brought online (at approximately 10% rated power) the turbine control valves maintain the first stage (impulse chamber) pressure at a programmed value that is proportional to power level. The turbine stop valves close rapidly to preclude turbine damage after the receipt of a turbine trip signal.

3.1.3.5 Instrumentation

A wide variety of secondary system instrumentation is available to the operators. Pressure is available at the MFW pump discharge and on the steam lines upstream and downstream of the MSIVs. Fluid temperature is indicated for each part of the Main Feedwater System and for the steam lines. Feedwater flow is available for each SG. Two SG level indications, wide range and narrow range, are provided, with the ranges indicated on Figures 3.1-7 and 3.1-8. **Two FSG level indications, wide range and narrow range, are provided with the ranges indicated on Figure 3.1-12.** The SG level instruments are ΔP devices, with the taps located at various elevations in the downcomer and shell. ΔP devices measure collapsed liquid levels, not the actual mixture or froth level of a fluid. The two level ranges are used for distinct purposes. The narrow range covers the middle portion of the SG and is used during normal operation, when the SGs have a significant water inventory. The wide range covers the middle and lower portions and is primarily used for evolutions which take place while at shutdown, such as wet layup.

3.1.4 Control Systems

Nuclear plants include a large number of control systems which monitor and adjust the performance of individual components and systems. In this section the control systems which have a major effect on the overall transient response of the plant are discussed.

Load Rejection Controller

Following a large, sudden load rejection or turbine trip without a reactor trip, all condenser dump valves and atmospheric dump valves may be enabled, depending on the magnitude of the load rejection. The load rejection controller operates in a manner similar to the plant trip controller and is also driven by an error signal derived from a temperature difference. The components of the temperature difference are the average temperature, as used in the plant trip controller, and the reference temperature, which is based on turbine impulse chamber pressure and is therefore indicative of turbine power. There is a [ ] deadband on the temperature difference before the first bank begins to open in load rejection control mode. The trip setpoints are

McGuire		Catawba	
Bank 1 (five valves)	[ ]	Bank 1 (three valves)	[ ]
Bank 2 (four valves)		Bank 2 (three valves)	
Bank 3 (four valves)		Bank 3 (three valves)	
Bank 4 (four valves)		Bank 4 (four valves)	
		Bank 5 (five valves)	

Steam Header Pressure Controller

Residual heat removal is maintained by the steam header pressure controller (manually selected) which controls the amount of steam flow to the condensers. This controller operates three of the condenser dump valves.

3.1.4.4 Pressurizer Level Control

The pressurizer water level is programmed as a function of coolant average temperature, with the highest average temperature (auctioneered) being used. The pressurizer water level decreases as the load is reduced from full load. This is a result of coolant contraction following programmed coolant temperature reduction from full power to low power. The programmed level is designed to match as nearly as possible the level changes resulting from the coolant temperature changes.

3.1.4.5 Steam Generator Level Control

Each McGuire steam generator is equipped with a three element feedwater flow controller which maintains a programmed water level as a function of neutron flux. The three element feedwater controller regulates the feedwater valve by continuously comparing the feedwater flow signal, the water level signal, the programmed level and the pressure compensated steam flow signal.

The Catawba Digital Feedwater Control System (DFCS) automatically controls feedwater flow to each steam generator to maintain programmed steam generator water levels. The level setpoint is a function of nuclear power. At power levels above approximately 25 percent, the feedwater flow to individual steam generators is controlled by a three element DFCS which uses temperature compensated feedwater flow, main steam flow, and steam generator water level as control parameters for the feedwater control valves. At power levels below approximately 25 percent, the DFCS automatically positions the feedwater bypass control valve and feedwater control valve to each steam generator based on the level setpoint.

3.1.4.6 Feedwater Pump Speed Control

The feedwater pump speed is varied to maintain a programmed pressure differential between the steam header and the feed pump discharge header. The speed controller continuously compares the actual ΔP with a programmed ΔP_{ref} which is a linear function of steam flow.

3.1.5 Safety Systems

Various systems are required to ensure that the plant does not exceed applicable limits during design basis transients. The major safety-related systems which affect the plant transient response are discussed in this section.

3.1.5.1 Reactor Protection System

The Reactor Protection System (RPS) monitors parameters related to safe operation of the core and trips the reactor to protect against fuel and cladding damage. In addition, by tripping the reactor and limiting the energy input to the coolant, the RPS protects against Reactor Coolant System structural damage caused by high pressure. A coincidence logic scheme is used to sense a trip condition. When the minimum number of channels trip, power is removed from the control rod drives of the shutdown banks, S_a - S_e, and the control banks, A-D. The rods fall into the reactor core and shut down the nuclear chain reaction.

The RPS will initiate a reactor trip on the following conditions:

1) Power range high neutron flux, high setting
2) Power range high neutron flux, low setting
3) Intermediate range high neutron flux
4) Source range high neutron flux
5) Loop temperature difference higher than the DNB limit (Overtemperature ΔT)
6) Loop temperature difference higher than the centerline fuel melt limit (Overpower ΔT)
7) Reactor coolant pump undervoltage
8) Reactor coolant pump underfrequency
9) High pressurizer pressure
10) Low pressurizer pressure
11) High pressurizer level
12) Low reactor coolant loop flow
13) Low-low steam generator level
14) Power range neutron flux high positive rate
~~15) Power range neutron flux high negative rate~~
1~~6~~5) Safety injection
1~~7~~6) Turbine trip while above a certain power level (48% at McGuire and 69% at Catawba)

Trips 2, 3, and 4 are enabled only at various low power levels. Trips 7, 8, 10, 11, and 12 are modified or ~~di~~sabled at various low power levels. Trips 1,

5, 6, 9, 13, 14, and 15~~, and 16~~ are always enabled while the reactor is critical.

3.1.5.2 Engineered Safeguards System

The Engineered Safeguards System consists of the Engineered Safety Features Actuation System (ESFAS) and various safeguards components. These components may also have dual functions, being used during normal operation as well serving as Engineered Safety Features. The ESFAS is divided into the following functions:

1) Safety Injection
2) Containment Heat Removal
3) Containment Isolation
4) Steam Line Isolation
5) Turbine Trip and Feedwater Isolation
6) Auxiliary Feedwater
7) Automatic Switchover to Recirculation
8) Loss of Essential Auxiliary Power System

These functions and the components actuated by them are discussed below.

Safety Injection

The Safety Injection System can be divided into four subsystems:

1) Two high head safety injection (HHSI) pumps
2) Two intermediate head safety injection (IHSI) pumps
3) Two low head safety injection (LHSI) pumps
4) Four passive cold leg accumulator tanks (CLAs)

All six pumps start on a safety injection signal. This signal is automatically generated on any of the following conditions:

1) Pressurizer pressure decreases below 1845 psig
2) ~~Lead-lag compensated s~~Steam line pressure decreases below ~~585~~775 psig~~ (McGuire) or 725 psig (Catawba)~~

3) Containment pressure increases above 1.1 psig (McGuire) or 1.2 psig (Catawba)

The first two actuations can be blocked when the reactor is being cooled down. The third actuation is always enabled.

The HHSI pumps have a shutoff pressure of approximately [ ] psig and runout flows of approximately [ ] These flow rates are for operation through the boron injection flowpath which terminates in 1½" lines which inject into each cold leg. The HHSI pumps also provide normal charging and reactor coolant pump seal injection. On a safety injection signal the suction source for the HHSI pumps is automatically switched from the volume control tank to the Refueling Water Storage Tank (RWST).

The IHSI pumps have a shutoff pressure of approximately [ ] psig and runout flows of approximately [ ] The IHSI pumps initially inject through four 2" lines which empty into the 6" lines from the LHSI pumps. If injection flow is to be maintained after ~~157~~ hours, the IHSI pumps are realigned to inject into four 6" lines which connect directly to each hot leg. The IHSI pumps are normally aligned to the RWST.

The LHSI pumps have a shutoff pressure of approximately [ ] and runout flows of [ ] The LHSI pumps initially inject through four 6" lines which empty into the 10" lines from the cold leg accumulator tanks. If injection flow is maintained long enough to empty the RWST, the suction of the LHSI pumps is automatically swapped to the containment sump. The operator then aligns the HHSI and IHSI pumps to take suction from the LHSI pumps. If injection flow is to be maintained after ~~157~~ hours, the LHSI pumps are realigned to inject into the B and C hot leg piping instead of the cold legs into which they previously injected. This realignment prevents unacceptable concentration of boron following a LOCA.

The four CLAs constitute a passive part of the Emergency Core Cooling System that performs no function during normal operation. Each of the four tanks is connected to its corresponding cold leg by a 10" injection line. The tanks

are pressurized to **approximately** 600 psig by nitrogen. Each 1393 ft^3 tank contains 918 ft^3 of borated water **at McGuire and 1020 ft^3 of borated water at Catawba** which, following a large break LOCA, is discharged into its cold leg. Each injection line contains two check valves which isolate the tank from RCS pressure during normal operation, but open to allow flow during a design basis accident. In addition to large break LOCAs, the CLAs will inject water into the RCS during major depressurization events, e.g., some small break LOCAs.

In addition to actuating the pumps discussed above, a safety injection signal will do the following:

1) start the motor driven AFW pumps
2) initiate a Phase A containment isolation
3) initiate a containment purge and exhaust isolation

Containment Heat Removal

The containment heat removal portion of the ESFAS and the components it controls, such as spray pumps and air return fans, do not play a major role in NSSS transient analysis and are not described here.

Containment Isolation

The containment isolation portion of the ESFAS and the isolation valves it controls are divided into two groups, Phase A and Phase B, depending on the signal which generated the isolation. Both signals can result in the closure of valves in lines which affect the NSSS. Although no general explanation is given here, such effects are modeled appropriately in the RETRAN analyses of applicable transients.

Steam Line Isolation

Steam line isolation occurs automatically from pressurization of the containment or uncontrolled depressurization of the steam lines. The containment pressure setpoint is 2.9 psig (McGuire) or 3.0 psig (Catawba). Steam line isolation on uncontrolled steam line depressurization depends on plant status. For normal pressurized operation ~~the lead-lag compensated~~ steam

line pressure is compared with a setpoint of 775 psig.~~, 585 psig (McGuire) or 725 psig (Catawba).~~ For depressurized operation the operator blocks this actuation to allow cooldown with the SGs. The blocking enables an automatic isolation on any steam line pressure rate more negative than -100 psi/second. A steam line isolation signal closes the MSIVs, the MSIV bypass valves, and the steam line PORVs.

Turbine Trip and Feedwater Isolation

If narrow range SG level exceeds the high-high setpoint, 82% (McGuire), ~~or~~ 82.4% (Catawba), **or 83.9% (FSG)**, the ESFAS will initiate closure of the turbine stop valves and of all valves supplying MFW flow to the SGs. These actions protect the turbine from damage due to moisture entrainment and stop MFW flow to help prevent SG overfill. In addition MFW isolation can occur on high water level in one of the Doghouses. This protects against continued MFW addition for a feedwater line break in the Doghouse. Feedwater isolation signals will also be generated by safety injection or by low RCS average temperature coincident with reactor trip, although not technically a part of the McGuire ESFAS, as defined by Technical Specifications.

Auxiliary Feedwater

The Auxiliary Feedwater (AFW) System has two 50% capacity motor-driven pumps and one 100% capacity turbine-driven pump. One motor-driven pump is aligned to SGs A and B, the other to SGs C and D. The turbine-driven pump is aligned to all four SGs~~ at McGuire, but only to SGs B and C at Catawba~~. The motor-driven pumps are automatically started on any of the following:

1) low-low narrow range level in any SG
2) safety injection
3) loss of offsite power
4) trip of both MFW pumps

The turbine-driven pump is automatically started on either of the following:

1) low-low narrow range level in two or more SGs
2) loss of offsite power

AFW flow is manually controlled by the operator following reactor trip to achieve and maintain the programmed narrow range SG level for zero power.

Automatic Switchover to Recirculation

On low RWST level the LHSI pump suction is automatically swapped from the RWST to the containment sump.

Loss of Essential Auxiliary Power System

Upon low voltage on the 4160 volt essential electrical busses, the diesel generators automatically start. The diesel generator load sequencers open the breakers for loads on the busses, close the diesel generator breakers to energize the busses, and then reclose the breakers for the various load according to prescribed timed sequences. The presence of a safety injection signal starts the diesel generator safeguards loading sequence, while a loss of offsite power with no safety injection signal starts the diesel generator blackout loading sequence.

3.1.6 Dissimilarities Between Units and Stations

~~3.1.6.1 Barrel-Baffle Flow Direction~~

~~The McGuire units have downflow in the region between the core barrel and baffle, making this region a part of the downcomer. There are holes in the core barrel below the top former plate which allow reactor vessel inlet flow to enter the region and travel through holes in the former plates to rejoin the main downcomer flow before entering the fuel assemblies. The Catawba units have upflow in the barrel-baffle region, making it a part of the core bypass. There are holes in the top former plate which allow the flow coming through the holes in the lower former plates to exit into the upper plenum.~~

3.1.6.~~2~~1 Steam Generator Type

The McGuire units and Catawba Unit 1 have split flow preheater regions. In such a preheater the MFW flow enters the middle of the region on the side and

divides into two flow streams. The upper stream flows across a series of baffle plates and upward, counter to the direction of RCS flow in the U-tubes. This stream exits into the upper tube bundle on the cold leg side. The lower flow stream flows across a different series of baffle plates and downward, along the direction of RCS flow in the U-tubes. This stream exits into a mixing region below the preheater where it joins with recirculated flow from the downcomer and flows over the lower tube bundle on the hot leg side.

Catawba Unit 2 has a counterflow preheater region. In this preheater design the MFW flow enters the middle of the region, is diverted to the bottom, and divides into two streams. One stream flows across the tube bundle to the hot leg side and joins recirculated flow from the downcomer. The other flows across a series of baffle plates and upward, counter to the direction of RCS flow in the U-tubes. This stream exits into the upper tube bundle on the cold leg side.

In addition to the preheater, the Catawba Unit 2 SGs differ from those of the other units in several other respects. There are sixteen primary separators (risers) on the Catawba Unit 2 SGs but only twelve on SGs at the other units. Fitting the four extra risers through the plate at the top of the tube bundle necessitated raising it to a higher and thus wider area in the transition cone. This results in a larger tube bundle region relative to the other units. The 4578 Catawba Unit 2 SG U-tubes are taller than the corresponding 4674 U-tubes on the other three units. The longer U-tubes at Catawba Unit 2 increase the resistance of the primary loop. This necessitated an increase in the rated head of the reactor coolant pumps for that unit to a value greater than the rated value for the reactor coolant pumps on the other three units. Finally, the split flow preheater configuration flow patterns necessitated a wide variation in programmed water level with power. ~~To accommodate this variation, the narrow range level span in a split flow preheater SG extends from the bottom of the secondary separators to just above the tops of the U-tubes. In contrast, the narrow range level span at Catawba Unit 2 extends from the bottom of the secondary separators to just above the top of the transition cone. Also, t~~The Catawba Unit 2 SG level program ~~is constant~~**has a narrow variation in programmed water level** as a function of power.

In order to correct U-tube wear problems associated with high MFW flow into the counterflow preheater region, the MFW flow delivery characteristics of the Catawba Unit 2 generators were modified. A flow restricting orifice was installed in the MFW line to the lower nozzle, limiting flow to this nozzle at full power to [ ] of total flow. The remaining [ ] of full power MFW flow is diverted to the upper nozzle. In contrast, the other units have upper nozzle MFW flows at full power of approximately [ ] of total flow, enough to prevent heatup of the discharge lines and upper nozzle.

The preheat SGs at McGuire Units 1 and 2 and Catawba Unit 1 will be replaced with feedring SGs. The main difference between the preheat and feedring designs is the manner in which main feedwater is delivered to the steam generators. In the feedring SG, the main feedwater flow is delivered to the feedring through the main feedwater nozzle and gooseneck. The J-tubes connected to the feedring distribute the feedwater axisymmetrically around the downcomer, where the feedwater mixes with the recirculation flow.

In addition, the feedring SGs differ from the preheat SGs in several other respects. Each FSG has a greater number of primary separators that are smaller than the preheat SG separators. The FSG tube bundle is taller than that of the preheat SGs and has a greater number of tubes. Thus, the FSGs have a much larger heat transfer area. The FSG level program is constant as a function of power, as is the Catawba Unit 2 SG level program.

3.1.6.~~3~~**2** Auxiliary Feedwater Runout Protection

The means by which excessive AFW flow to a faulted SG is prevented for a secondary line break accident is station specific. At McGuire, there are travel stops on the discharge valves in the lines from each AFW pump to each SG. These stops are set to allow no more than a certain amount of flow to any SG assuming it is fully depressurized while the other SGs are at the setpoint of the steam line safety valves. Catawba uses an active runout protection system. ~~:~~

~~1) The turbine driven AFW pump is not normally aligned to SGs A and D~~

~~2) The isolation valve on the motor driven AEW pump discharge line to either SG B or C will close if~~

~~i)~~ ~~the turbine driven AFW pump is running, and~~

~~ii)~~ ~~the motor driven AFW pump not aligned through that valve fails to start~~

~~3)~~ The isolation valve on the motor-driven AFW pump discharge line to either SG B or C will close if

~~i)~~ ~~the turbine driven AFW pump is running, and~~

~~ii)~~ discharge flow from the motor-driven AFW pump aligned through that valve exceeds a certain setpoint

3.1.6.~~4~~3 Steam Line Layout

The McGuire main steam lines exit the four SGs and go to the MSIVs in the Doghouse. Downstream of the MSIVs the 34" steam lines enter the side of a 48" diameter header. At one end of this header a 24" line goes to the eight atmospheric dump valves. From the other end another 24" line goes to the nine condenser dump valves. From the side of the header four 34" lines carry main steam to the turbine inlet via the stop and control valves. The McGuire arrangement is shown in Figure 3.1-10. At Catawba the arrangement is similar through the MSIVs. Downstream of the MSIVs each 34" line maintains its identity separately from the other lines, reducing to 28" each before reaching the turbine stop and control valves. At the stop valves is a 35" equalization header connecting each steam line. Further upstream, a 28" line separates from each steam line. These four lines join to form a 28" header. At one end of this header a 24" line goes to the nine atmospheric dump valves. From the other end another 24" line goes to the nine condenser dump valves. The Catawba arrangement is shown in Figure 3.1-11.

3.1.6.~~5~~4 Miscellaneous Differences

There are several miscellaneous differences between stations and units which affect transient analysis modeling:

1) The outlet nozzle on the McGuire Unit 1 reactor vessel is [ ] as the nozzles on the other three units, giving it a [ ] as the nozzles on the other units.

2) The number and types of the various upper internals structures is different for McGuire Unit 1 than for the other three units as shown below:

McGuire Unit 1	Other Units
[ ] 17 x 17 guide tubes	[ ] 17 x 17A guide tubes
15 x 15 guide tubes	15 x 15 guide tubes
support columns	support columns
flow columns	flow columns
thermocouple columns	thermocouple columns

3) McGuire Unit 1 has thermocouple instrumentation in the reactor vessel upper head while the other three units do not.

4) The RCS average temperature program is 2.6°F higher at full power for Catawba than for McGuire. **The RCS average temperature program with the FSGs (McGuire Units 1 and 2 and Catawba Unit 1) is 5.7°F lower at full power than for Catawba Unit 2.**

5) Due to noise problems encountered with the pressurizer pressure transmitters on initial startup, the McGuire pressure signals have a ~~2~~1 second lag imposed before being used for control and protection purposes. The Catawba pressure signals have no lag.

6) There are several minor setpoint differences between McGuire and Catawba, e.g., the ΔT reactor trip gains and time constants and the pressurizer level program.

7) Because of the variation in operating time among the four units, differences exist in the number of tubes plugged on the various SGs. These differences are modeled, where appropriate, in RETRAN transient analyses.

~~8) The Catawba Emergency Core Cooling System includes an additional pressurized accumulator, part of the Upper Head Injection (UHI) System, which discharges water into the reactor vessel upper head.~~

~~This accumulator discharges at a RCS pressure of approximately 1200 psig and is automatically isolated on low water level. The UHI system has been removed from service at McGuire and removal at Catawba is also planned. This system does not actuate in any of the analyses documented in the report. Due to its imminent removal and non-applicability to this report, the UHI System is not modeled.~~

Figure 3.1-7
Counterflow Preheater Steam Generator

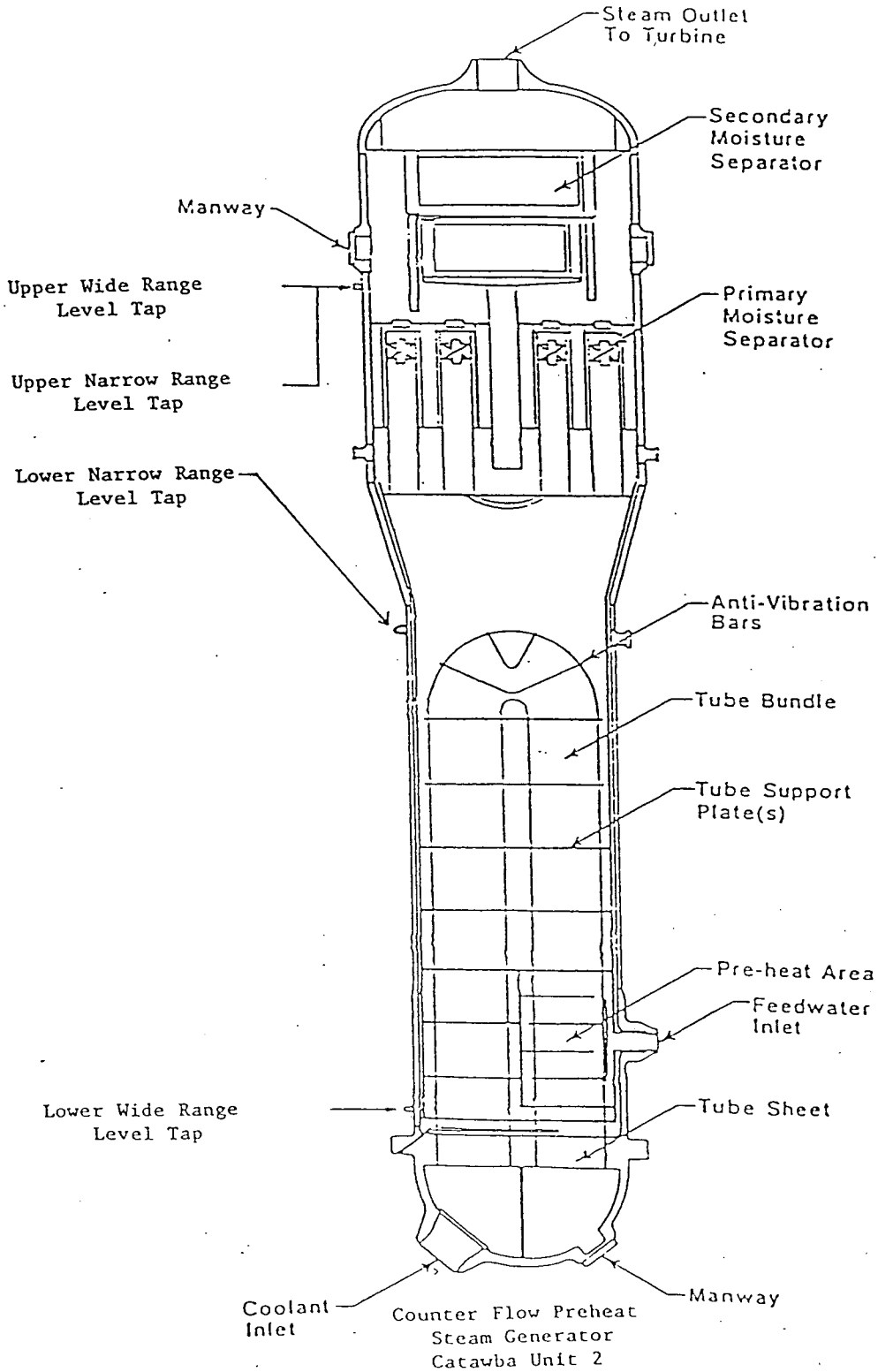
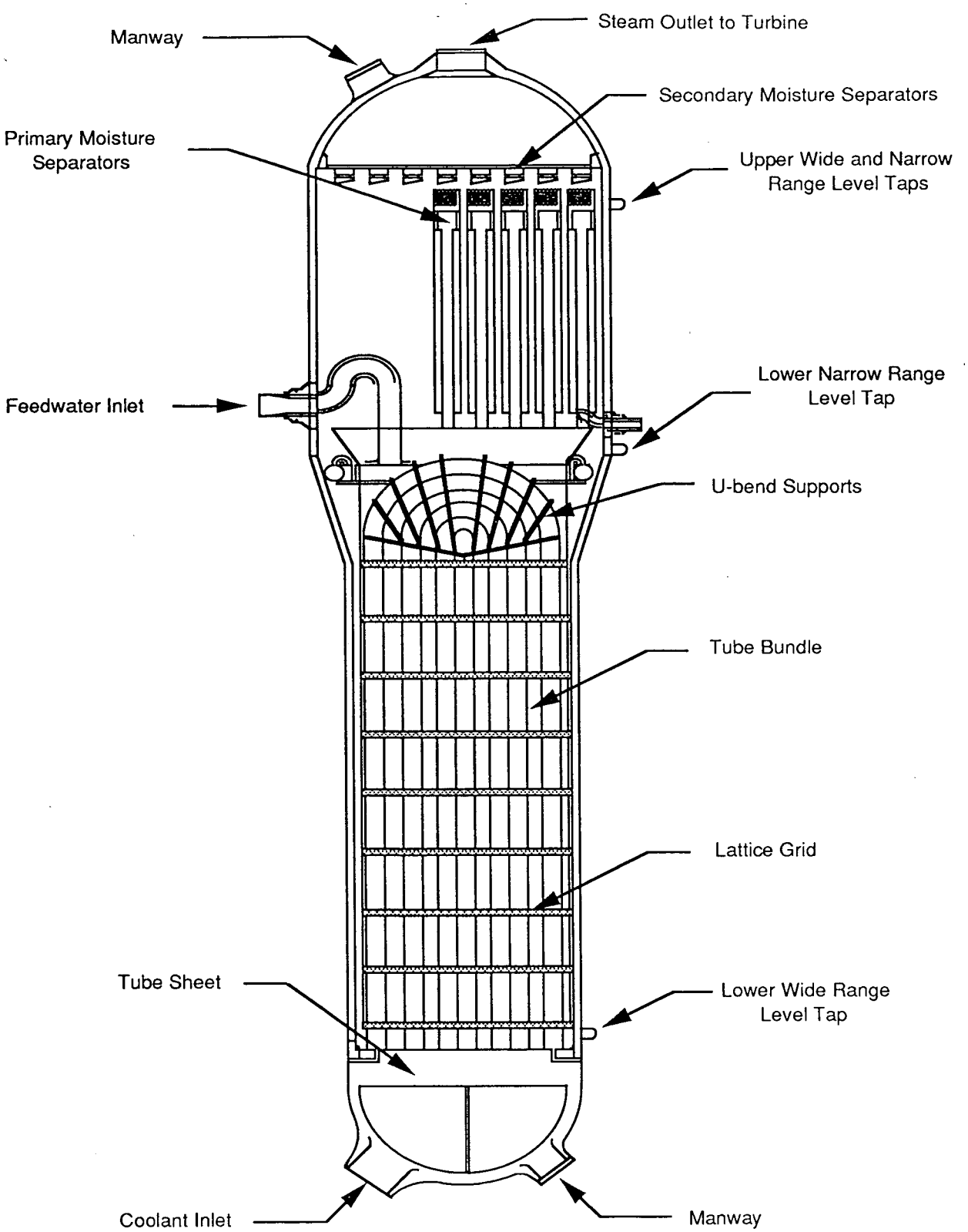


Figure 3.1-12
Replacement Steam Generator



Replacement Steam Generator

McGuire Unit 1
McGuire Unit 2
Catawba Unit 1

3.2 McGuire/Catawba RETRAN Model

The McGuire/Catawba RETRAN model nodalizations are shown in Figures 3.2-1 and 3.2-2 for the two-loop and one-loop models respectively. **For feedring SG transient analysis, the feedring SG nodalization shown in Figure 3.2-3 replaces the preheat steam generator nodalization shown in Figures 3.2-1 and 3.2-2.** The one loop model is used for transients which exhibit a sufficient amount of symmetry. For certain applications the amount of detail is excessive and can be reduced to save computer time, while on occasion additional detail e.g., a three loop model, is required.

The primary system model is symmetric relative to the two loops. The single loop components (volumes, junctions, and conductors) have numbers in the 100s. The triple loop component numbering scheme is the same, except that the numbers are in the 300s. Thus Volume 113 corresponds to Volume 313, the former being in the single loop and the later in the triple loop.

3.2.1 Primary System Nodalization

3.2.1.1 Reactor Vessel

The reactor vessel is modeled by [ ] fluid volumes. The boundaries between the volumes are chosen due to actual physical separations, or to provide an additional level of detail in the hydrodynamic calculation.

Downcomer

[

] Flow enters through the four cold legs and exits into the lower plenum.

Lower Plenum

The reactor vessel lower plenum is represented by [ ] Flow from the downcomer goes through the lower plenum into the core and the core bypass.

Core

[ ] represent the reactor core region from the [ ] There is no physical separation between [ ] Flow enters from the lower plenum and discharges into the upper plenum. The [ ] to provide a more accurate simulation of the temperature profile in the core at power.

Core Bypass

The core bypass region is modeled by [ ] The bypass flow channels include the control rod guide tubes and instrument tubes inside the fuel assemblies. In addition, ~~at Catawba~~ it includes the area between the core baffle plate and the core barrel which is exterior to the fuel assemblies. All of the bypass constituents are [ ] The control rods are assumed to be [ ] Flow enters the bypass from the lower plenum and exits into the upper plenum.

Upper Plenum

The upper plenum of the reactor vessel, which extends from the [ ] In the upper plenum the coolant flows upward from the core and then turns radially outward to leave the vessel through the outlet nozzles. Another flowpath involving the upper plenum is the one between the lower portion of the upper plenum, just above the active fuel, and the control rod guide tubes and UHI support columns. Some flow goes through these structures into the vessel head.

] between the liquid and vapor regions.

3.2.1.5 Cold Leg Accumulators

The four cold leg accumulators and their associated injection lines are [ ] The RETRAN air model is used to simulate the nitrogen overpressure on top of the tanks. [ ] allows the tanks to discharge when the RCS pressure drops below 600 psig.

3.2.2 Secondary System Nodalization

3.2.2.1 Main Feedwater Lines

[ ] represents the MFW lines between the [ ] represents the MFW lines between the [ ] and the SGs.

3.2.2.2.1 **Preheat** Steam Generators

The **preheat** steam generator secondary side is modeled by a total of [ ] volumes. [ ] Both McGuire units and Catawba Unit 1 have split flow preheat steam generators, which have [ ] full power feedwater flows out the top and side preheater outlets, [ ] respectively. Catawba Unit 2 has counterflow preheat steam generators, which have a [ ] during power operation. Flow through [

]

The basis for the SG secondary nodalization is twofold. The tube bundle has been [ ] encountered there. The downcomer has been [ ]

The RETRAN [ ]

3.2.2.2.2 Feedring Steam Generators

The feedring SG secondary side is modeled by a total of [ ] volumes. The downcomer is [

]

The basis for the FSG secondary nodalization is similar to that for the preheat SGs. The tube bundle has been [ ] encountered there. The downcomer has been [ ] The RETRAN [ ]

3.2.2.3 Main Steam Lines

[ ] Downstream of the MSIVs the nodalization used is station and transient dependent. Section 3.1.3.3 describes the actual plant steam line layouts. Since flows from individual SGs at McGuire are not separated all the way to the turbine, [

common header, the turbine inlet lines, and the atmospheric dump valve header.] At Catawba, [

] at Catawba since the steam lines are separate between the MSIVs and the turbine stop valves. [ ] For load rejection transients the atmospheric dump header is modeled as [ ]

The main steam lines are not physically different for the FSGs; however, the volume representing the main steam lines is [ ] for the FSG nodalization.

3.2.3 Heat Conductor Nodalization

3.2.3.1 Reactor Core

[ ] conductors are used to model the fuel rods in the reactor core. The conductors are separated into [ ] Material properties (thermal conductivity and heat capacity) are [ ] The fuel gap [ ] to give the [ ] fuel temperature, which varies with core average burnup. This approach is used to properly account for the stored energy in the fuel. The RETRAN core conductor model is used for these conductors in order to allow power generation in the fuel material. 2.6% of the power generated in the core is assigned to direct heating of the moderator rather than deposition of energy in the fuel pellets.

3.2.3.2 Steam Generator Tubes

[ ] heat conductors are used to represent the tubes in each of the SGs. There is [ ] volume. Material properties are [ ]

3.2.3.3 Structural Conductors

These conductors represent the plant components which do not generate power or conduct heat from the primary to the secondary, but which can affect the plant transient response by transferring energy to or from the working fluid. The stored energy and heat capacity of these conductors tend to dampen changes in RCS conditions. During an overcooling event the structural conductors transfer heat to the primary coolant and thus retard the cooldown. Conversely, during an overheating transient the structural conductors act as a heat sink and reduce the magnitude of the increase in the primary coolant temperature. The effect of the structural conductors is most apparent during long transients. During short transients which do not exhibit severe undercooling or overcooling, the heat transferred from the structural conductors is unimportant relative to the large amount of decay heat in the core. However, the structural conductors represent a significant heat load for long-term cooldown, once decay heat has decreased.

The key parameters for the structural conductors are the mass and the heat transfer area. These determine the initial stored energy and the effectiveness as a heat source or sink. In order to [ ]

Certain structural components are not included in the model because they are considered to have no potential impact on the plant transient. These components include the [ ]

Passive heat conductors representing the pressurizer walls [ ] in the McGuire/Catawba model. [ ] The pressurizer vessel metal is [ ]

The McGuire and Catawba tubesheet for both the preheat SGs and the feedring SGs is modeled by [

]

The heat conductors which are used in the McGuire/Catawba base model are listed and described in Table 3.2-1. **The heat conductors which are used in the FSG model are listed and described in Table 3.2-2.**

3.2.4 Control System Models

3.2.4.1 Process Variable Indications

RETRAN control systems are used to take the calculated plant thermodynamic conditions and put them into the form in which they are output by the plant instrumentation. This provides indications which are useful for comparison to plant data and which are familiar to the plant operators and engineering personnel.

Pressurizer Pressure

The fluid pressure at the elevation of the pressurizer upper pressure tap is converted to gauge pressure by subtracting 14.7 psi. This pressure is used as input to RPS and ESF functions in the model.

Steam Line Pressure

The volume pressures [ ] This pressure is used as an input to the ESF functions in the model.

SG Level

The narrow range and wide range SG level instruments display level from 0-100%, as shown on Figures 3.1-7, ~~and~~ 3.1-8, and 3.1-12. The level indication is derived from the pressure difference between taps in the steam generator. A RETRAN control system is used to simulate this process using the [ ] The output narrow range level indication is input to the RPS and ESF functions in the model. The wide range indication is used for information only. Control system simulated level indications [ ]

3.2.4.2 Reactor Protection System Functions

Five RPS functions are modeled with control systems:

1) Overtemperature ΔT
2) Overpower ΔT
3) Low steam line pressure safety injection (SI)
4) Low pressurizer pressure
5) Low-low SG narrow range level

The control systems for the ΔT trips compute the appropriate ΔT setpoints based on the Technical Specification equations and subtract the indicated ΔT values from these setpoints to determine whether trip occurs. The overtemperature ΔT trip equation reduces the ΔT setpoint for low coolant pressure and high coolant temperature to protect against departure from nucleate boiling. The overpower ΔT trip equation reduces the ΔT setpoint for high or increasing coolant temperature to protect against centerline fuel melt. Both ΔT trip equations also reduce the setpoints for excessive axial

flux imbalance, [ ] Lead-lag compensation is applied to the ~~low steam line pressure SI and~~ low pressurizer pressure reactor trip~~s~~ via a control system~~s~~ before the relevant value~~s are~~ **is** compared against **a** fixed setpoint~~s~~. The low-low SG narrow range level trip setpoint is a lagged programmed function of neutron flux **for the preheat SGs. The feedring SG low-low narrow range level trip setpoint does not vary with neutron flux.** A lagged value of indicated level is then compared with the setpoint to determine whether trip occurs.

3.2.4.3 Engineered Safeguards Functions

Four ESFAS functions are modeled with control systems:

1) Steam line isolation on low steam line pressure
2) ECCS pump start on safety injection
3) AFW pump start on low-low SG narrow range level
4) Turbine trip and MFW isolation on high-high SG narrow range level

The first three actions are coincident with reactor trip and use the same control systems. The fourth function is similar to the reactor trip on low-low SG narrow range level but uses a higher setpoint.

3.2.4.4 Plant Control Systems

RETRAN control systems are used to model the performance of certain plant control systems during transient analyses. These control systems fall into two general types. Some control systems, examples of which are given below, are modeled directly as designed. Other control systems are modeled indirectly. Indirect modeling is used when the desired control system action is known beforehand. This method saves time over direct modeling and can also be used to simulate controller action with undocumented setpoints, e.g. a field adjusted gain setting, or with failed components, e.g. a valve which cycles erratically.

coolant loops, unequal phase velocities normally exist only in the steam generator secondary side. RETRAN has two slip models: algebraic slip and dynamic slip. The algebraic slip model uses a drift flux approach to calculating the relative velocity between the vapor and liquid phases. ~~Current RETRAN development efforts are geared toward improving t~~The dynamic slip model [

] ~~Efforts to initialize the current McGuire/Catawba SG secondary model with the dynamic slip option~~ [

]

For applications in which there is significant voiding and phase separation in the primary system (notably small break LOCA or extended loss of feedwater), the dynamic slip model can provide a reasonable simulation of two-phase phenomena in the RCS. ~~In these instances the dynamic slip option~~ [

]

3.2.6.5 Non-Equilibrium Pressurizer

RETRAN has a general non-equilibrium volume option which can be used with any bubble rise volume. This option allows the liquid and vapor regions of the volume to have different temperatures. The [

]

Accurate modeling of the pressurizer is necessary to correctly predict the transient RCS pressure response. During normal operation the pressurizer is at near equilibrium conditions - heat from the pressurizer heaters balances condensation from pressurizer spray bypass flow and ambient heat losses - so both the liquid and vapor regions of the pressurizer are essentially at saturation. During a pressurizer outsurge, such as that characteristically seen immediately following reactor trip, the pressure decreases as the steam bubble expands. Bulk flashing of the saturated liquid occurs as the pressure decreases, and the temperature in the liquid decreases with the pressure along the saturation line. In both cases the standard RETRAN homogeneous equilibrium model (HEM) technique will adequately simulate pressurizer phenomena.

During a pressurizer insurge, however, non-equilibrium effects can be significant. Subcooled water from the hot leg mixes with the saturated water in the pressurizer liquid region to produce a somewhat subcooled liquid region or, in some cases, a layered effect of saturated water over subcooled liquid. As the liquid level increases the steam bubble compresses and, since the steam behaves like an ideal gas, the temperature increases. The overall result is superheated steam above subcooled liquid, separated by a layer of saturated water. Since the temperature of the steam is higher than both the liquid and the pressurizer walls, the steam will tend to condense on the metal and the steam-water interface, reducing the pressure and temperature of the vapor. A one volume HEM representation of the pressurizer would instantaneously mix the subcooled fluid from the hot leg with all of the saturated fluid in the pressurizer, and it would not account for the different temperatures in the liquid and vapor regions. It is evident that a HEM representation of the pressurizer cannot account for the important phenomena during an insurge.

Use of the non-equilibrium option enhances the ability of a one volume pressurizer model to simulate the transient response during an insurge. Since the liquid region is considered separately from the steam bubble, an insurge does not result in rapid condensation of the pressurizer vapor. The non-equilibrium option allows the steam bubble to superheat during an insurge. The non-equilibrium option also includes the ability to specify a heat transfer coefficient between the pressurizer vapor and liquid regions, so interphase heat transfer can be modeled. However, this model is somewhat non-mechanistic, since the heat transfer coefficient is user-input rather than being calculated based on fluid conditions. ~~Furthermore, the ability to model~~

~~in the heat transfer model which are discussed in Section 3.2.3.3.~~

Another facet of the non-equilibrium representation is the pressurizer spray junction model. Using the spray junction option causes the spray water to condense steam while moving through the pressurizer steam bubble, thus removing both energy and mass from the region, rather than simply mixing with fluid in the vapor region and desuperheating the steam. The spray junction

]

3.2.6.7 Local Conditions Heat Transfer

The local conditions model allows the approximation of variable heat transfer in a volume in which void fraction varies substantially with elevation, particularly in the case of a separated volume with a variable mixture level. This model [

]

3.2.7 Code Options

3.2.7.1 Steady-State Initialization

The RETRAN steady-site initialization option is used to obtain stable initial conditions for each transient analysis. This option greatly simplifies the specification of the initial conditions of a RETRAN run. The steady-state initialization routine solves the mass, momentum, and energy equations without the time-dependent terms and thus obtains consistent initial values with a minimal amount of input data.

Primary system conditions for McGuire/Catawba models are set by specifying

[

The initial SG power removal fraction is set at 0.25 for each generator in order to provide a symmetric initialization.

3.2.7.2 Iterative Numerics

The iterative numerics option is used for time step control. Iterative numerics is a semi-implicit numerical solution method which allows the results of the time step advancement to be evaluated before the solution is accepted. Predictive algorithms are used to calculate an appropriate time step size which will give a stable, accurate solution to the fluid conservation equations. If a converged solution is not achieved in a given number of iterations, a reduced time step size is used. This is similar to restarting a job with smaller time steps, but it has the advantage of being automatic.

3.2.7.3 Enthalpy Transport

The enthalpy transport option is used to account for situations in which the fluid in a volume exchanges a significant amount of energy with an external source or sink. In those situations, the fluid enthalpy will vary between the volume inlet, center, and outlet, and the enthalpy transport option accounts for this variation. The option is used

3.2.7.9 Wall Friction

RETRAN calculates the pressure drop due to wall friction using the Fanning friction factor, which is a function of Reynolds number. Several options are available to model the change in wall friction due to two-phase effects. The Baroczy, homogeneous, or Beattie multipliers can be applied to the calculated single phase pressure drop. The [ ] model is used in the McGuire/Catawba RETRAN model.

3.2.7.10 General Transport Model

The general transport model is used to calculate the boron concentration in [ ]. The boron is assumed to be soluble in the transport medium and to have no direct effect on the fluid equations. The basic equation computes the time rate of change of boron mass in a control volume from the net inflow through connected junctions.

3.2.8 Dissimilarities Between Units

The differences in RCS loop geometry are significant enough to warrant separate base models for each unit. For a given unit model the coolant loops are lumped or separated depending on the asymmetry of the transient being analyzed. Differences between units, including the major differences discussed in Section 3.1.6, are included in unit specific models depending on the degree to which such differences affect the transient being analyzed.

- The point kinetics model adequately predicts the reactor power response to the types of reactivity changes which arise during typical operational transients.

- The reactor coolant pump model accurately reflects the interaction of the pumps and the primary fluid during normal pump operation and coastdown.

- The control system models and trip logic are extremely flexible and useful for modeling automatic control actions as well as operator action.

Similar to other one dimensional HEM codes, the current models in RETRAN have been found to have shortcomings in some areas and are incapable of adequately simulating particular phenomena. **One recognized shortcoming is that** ~~These areas are discussed below.~~

- ~~T~~the lack of a general non-equilibrium modeling capability detracts from the ability of the code to simulate some small break LOCA behavior. This limitation must be recognized whenever such applications are undertaken.

- ~~The current pressurizer model does not~~ [ ] ~~This limitation can be compensated for by careful modeling in order to ensure a conservative simulation.~~

In general, the overall experience with modeling the McGuire and Catawba transient response using RETRAN has been good. Despite shortcomings in some areas, the code has been proven capable of accurately simulating the transient thermal-hydraulic behavior of a Westinghouse PWR with preheater-type steam generators. Due to the relatively minor differences between the preheater-type steam generator and the feedring steam generator, the code should be capable of accurately simulating the transient thermal-hydraulic behavior with the feedring steam generators.

Table 3.2-1

McGuire/Catawba Preheat SG Base Model Heat Conductors

Conductor Number	Adjacent Volume Number(s)	Description	Material

Table 3.2-1 (continued)

McGuire/Catawba Preheat SG Base Model Heat Conductors

Adjacent Conductor Number	Volume Number(s)	Description	Material

Table 3.2-1 (continued)

McGuire/Catawba Preheat SG Base Model Heat Conductors

Adjacent Conductor Number	Volume Number(s)	Description	Material

Table 3.2-2

McGuire/Catawba Base Model

Feedring Steam Generator Heat Conductors

Adjacent Conductor Number	Volume Number(s)	Description	Material

Figure 3.2-1

Figure 3.2-2

Figure 3.2-3
McGuire/Catawba RETRAN Model
Feedring Steam Generator
Nodalization Diagram

Feedring Steam Generator
McGuire Unit 1
McGuire Unit 2
Catawba Unit 1

Pressure, psia	1000 to 23
Mass velocity, 10^6 lbm/hr-ft^2	1.0 to 5.0
Quality, %	-0.15 to 0

However, it has been shown recently (Reference 3-8) that the W-3S correlat is also applicable for pressure and mass flux as low as 700 psia and 0.5x1 lbm/hr-ft^2, respectively.

Other correlations that may be utilized for the low pressure and low flow analysis are the MacBeth (Reference 3-4) and Bowring (WSC-2) (Reference 3- CHF correlations. The range of applicability of the MacBeth correlation i

Pressure, psia	15 to 2700
Mass flux, 10^6 lbm/hr-ft^2	0.0073 to 13.7

The range of applicability of the Bowring (WSC-2) correlation is:

Pressure, psia	500 to 2300
Mass flux, 10^6 lbm/hr-ft^2	0.02 to 3.0
Quality (equilibrium)	-0.2 to 0.86

Justification will be provided as necessary, when applying these correlati in future analyses.

Other CHF correlations that have been reviewed and approved by the NRC may also be used to perform DNBR analyses.

3.3.3.2 Conservative Factors

When predicting DNBR with the BWCMV or DCHF-1 correlations, the SCD design limit will generally be used. The SCD design limit accounts for all of th uncertainties (with one exception), and therefore additional conservative factors are unnecessary. Only the conservative factor to account for a possible core inlet flow maldistribution, which is detailed below, is appl with the SCD design limit.

Attachment II