7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 DESCRIPTION

The general descriptions of control and instrumentation systems that are required for the reactor control, though not essential for the safety of the plant, are given below.

The reactivity feedback properties of the NSSS will inherently cause reactor power to match the total NSSS load. The resulting reactor coolant temperature condition at which this occurs is a controlled parameter and is adjusted by changes in total reactivity as implemented through CEA position changes or addition, or removal of boric acid dissolved in reactor coolant.

The ability of the NSSS to follow turbine load changes, while maintaining acceptable limits, is dependent on the ability of the control systems or operator to adjust reactivity, feedwater flow, bypass steam flow, reactor coolant inventory, and net energy input to the pressurizer.

Except as limited by xenon conditions, the major control systems described below provide the capability to automatically follow ramp load changes between 15 percent and 100 percent at a rate of five percent per minute and at greater rates over smaller load change increments up to a step change of 10 percent. Additionally, these automatic systems provide the capability to accommodate load rejections of any magnitude or the loss of a feedwater pump.

7.7.1.1 Reactivity Control

7.7.1.1.1 Reactor Regulating System

The Reactor Regulating System (RRS) automatically adjusts reactor power and reactor coolant temperature to follow turbine load transients within established limits. The RRS receives a turbine load index signal (first stage turbine pressure) as a linear indication of load. This power-reference is fed to a temperature programmer, which establishes the desired reactor coolant average temperature. Power range neutron flux and pressurizer pressure are compensating inputs to the system. The regulating system generates outputs of CEA drive direction and speed based on compensated error signals derived from these inputs. A programmed level signal, which is a function of primary coolant temperature, is provided by the RRS as a reference for the pressurizer level control system.

The Control Element Drive Mechanism Control System (CEDMCS) accepts automatic CEA motion demand signals from the RRS or manual motion signals from the CEDMCS control panel and converts these signals to direct current pulses that are transmitted to the CEDM coils to cause CEA motion.

A reactor trip initiated by the Reactor Protective System (RPS) causes the input motive power to be removed from the CEDMCS by the trip switchgear, which in turn causes all CEAs to be inserted by gravity. The CEDMCS is thus not required for safety (see Figures 7.7-1 and 7.7-2).

There are four modes of control; automatic sequential group movement, manual sequential group movement, manual group movement and manual individual movement. The sequential modes

of operation apply only to the regulating CEAs. The CEDMCS accepts automatic CEA motion demand signals from the RRS or manual signals from the CEDMCS control panel. These signals are converted to dc pulses which are transmitted to the CEDM coils. Sequential group movement functions such that, when the moving group reaches a programmed low (high) position, the next group begins inserting (withdrawing), thus providing for overlapping motion of the regulating groups. The initial group stops upon reaching its lower (upper) limit. Applied successively to all regulating groups, the procedure allows a smooth, continuous rate of change of reactivity. In automatic mode the CEDMCS accepts signals from the plant monitoring computer and the RRS to effect this sequencing of regulating CEA group motion. The CEDMCS utilizes sequencing signals from the plant monitoring computer that are derived from the CEDMCS up-down pulse counters. There is no tie into the CEA reed switch assembly sensors used in the RPS. In manual sequential group movement the CEDMCS accepts signals from the reactor operator and plant monitoring computer to effect this sequencing of regulating CEA group motion. The shutdown CEAs are moved in the manual control mode only, with either individual or group movement. A selector switch permits withdrawal of no more than one shutdown group at any time.
→(DRN 02-386, R12)

←(DRN 02-386, R12)
→(EC-18522, R304)
During plant startup, shutdown, axial shape index control, and when reactor power is less than 15 percent, manual control is used. The preferred mode of CEA control during normal operation is manual. Automatic control of the regulating CEAs by the Reactor Regulating System may be selected by the operator only when greater than 15 percent reactor power. Manual CEA control may be used to override automatic CEA control at any time.
←(EC-18522, R304)

7.7.1.1.2 Boron Control

Information is provided to the operator to allow regulation and monitoring of the boron concentration in the reactor coolant. The means by which reactor coolant system boron control is accomplished are dilution and boration. The volume control tank contents may be maintained at a prescribed boron concentration either manually or automatically. Two additional recorders plus two flow totalizers are used to record reactor makeup water flow and boric acid makeup flow which can be used to determine that boration or dilution is occurring.

The boron concentration information described above is used in addition to, and as a backup to, regular sampling of the reactor coolant to determine boron concentration.

The boron concentration, in conjunction with CEAs ensure that adequate shutdown margin is maintained for the reactor. This margin is maintained while critical by preventing the boron concentration from decreasing to a Reactor power level versus regulating CEA position is used to provide adequate shutdown margin. Thus, if for a certain power level, decreasing boron concentration is causing the regulating CEAs to be driven excessively into the core an alarm will indicate this condition to the operator. This alarm is provided from the pulse counting CEA position system.

7.7.1.2 Pressurizer Controls

7.7.1.2.1 Pressurizer Pressure Control System

The Pressurizer Pressure Control System maintains system pressure within specified limits by the use of pressurizer heaters and spray valves.

During normal operation, a small group of heaters is proportionally controlled to maintain operating pressure. If the pressure falls below the full on setpoint for the proportional band by approximately 25 psi all of the heaters are energized. Above the normal operating pressure range the heaters are deenergized and the spray valves are proportionally opened to increase the spray flow rate as pressure rises. A small, continuous spray flow is maintained through the spray lines at all times to keep the lines warm and thereby reduce thermal shock when the control valves open, and to ensure that the boric acid concentration in the coolant loops and pressurizer is in equilibrium.

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A high pressurizer level error signal concurrent with normal or low pressure energizes backup heaters to full capacity to minimize the subcooling during the transient. A high pressurizer pressure regardless of level deenergizes all heaters. Also, low pressurizer water level signal deenergizes all heaters, thereby providing heater protection.

Two measurement channels are provided and the controlling channel is selected by a switch on the control board. Automatic control is normally used during operation, but manual control of the heaters and the spray may be selected at any time. Pressurizer pressure control is shown on Figure 7.7-3.

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7.7.1.2.2 Pressurizer Level Control System

The Pressurizer Level Control System regulates the reactor coolant system water inventory by use of the charging pumps and letdown control valve(s) in the Chemical and Volume Control System.

During normal operation, the pressurizer level is programmed by the RRS as a function of the average reactor coolant temperature in order to maintain the proper coolant inventory for anticipated transients. The level controller compares the measured and programmed level signals and generates a signal to modulate one of the letdown control valve(s) in such a way that the level is restored to its programmed value. Separate on-off controllers start or stop the standby charging pumps when level deviation setpoints are exceeded.

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Two measurement channels (differential pressure transmitters) are provided and the controlling channel is selected by a switch on the control board. Automatic control is normally used during operation, but manual control may be used at any time.

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One charging pump is operated continuously in order to provide makeup flow for reactor coolant pump seal leakage and to limit letdown temperatures.

Backup control action is provided by several on-off controllers. A high level error signal stops both backup charging pumps and activates an alarm. A low level error signal provides a backup signal to start all charging pumps and also activates an alarm. This low level signal is also used in the pressurizer pressure control system to shut off the pressurizer heaters. Pressurizer level control is shown on Figure 7.7-4.

7.7.1.3 Feedwater Control System

One Feedwater Control System (FWCS) is provided for each steam generator- The two steam generators are operated in parallel. Each FWCS maintains steam generator downcomer water level within acceptable limits by positioning feedwater control valves and adjusting the speed of the feedwater pumps to regulate the feedwater flowrate to the steam generators.

The FWCS is a four element control system using feedwater flow, feedwater temperature, steam flow, and steam generator downcomer water level as inputs. Each FWCS provides output signals to position the respective feedwater control valves. A hysteresis is built into each valve program to reduce wear at low flow. In addition, each system simultaneously provides a pump speed setpoint to the turbine-driven feed-pump speed control systems. This configuration allows automatic control above 15 percent power.

When a deviation occurs between the two level channel signals associated with a steam generator, or when there is a loss of ac power to the FWCS, the FWCS manual/automatic control stations are transferred to manual. These stations must be returned to automatic manually when the deviation is cleared or when ac power is restored.

When an abnormally high steam generator downcomer level is sensed in either steam generator, a signal is sent from the associated FWCS to close the associated feedwater control valves. This signal is automatically removed when the abnormal condition clears.

When a reactor trip occurs, each FWCS automatically reduces the feedwater flowrate to its respective steam generator by closing the associated main feedwater control valves, partially opening the bypass feedwater control valves, and limiting the feedwater pump speed.

The manual control mode of each FWCS may be selected by the operator at any power level. When in manual control, the operator in the main control room can use a master control station to simultaneously adjust the valve positions and pump speed setpoints to maintain steam generator downcomer water level, or can choose to control the valves and pump speeds separately from individual manual/automatic stations. Control at the master control station is the preferred manual operating mode since this minimizes operator control actions. The FWCS is shown in Figures 7.7-5 and 7.7-6.

7.7.1.4 Steam Flow Controls

7.7.1.4.1 Steam Bypass Control System

The main objective of Steam Bypass Control System (SBCS) is to maximize plant availability by making full utilization of the turbine bypass valve capacity to remove NSSS thermal energy. This objective is achieved by the selective use of turbine bypass valves and/or dropping of selected CEA groups to avoid unnecessary reactor trips and prevent the opening of secondary side safety valves whenever these occurrences can be averted by the controlled release of steam or rapid reduction of power.

In addition to providing a means for controlling NSSS thermal energy during plant start-up, cooldown, and hot standby, the SBCS is designed to accommodate load rejections, unit trips, and other conditions that result in the generation of excessive energy by the NSSS.

The system design has the following characteristics:

→(DRN 03-2155, R13)

a) Permits the accommodation of load rejections of any magnitude without opening either the pressurizer or the steam generator safety valves or tripping the reactor with the exception that there is a probability that the reactor may trip at power levels between 50% and 70% at certain times in core life (a trip is more probable at 70% power at Beginning of Cycle).

b) Allows the avoidance of reactor trips on turbine trips with the exception that there is a probability that the reactor may trip at power levels between 50% and 70% at certain times in core life (a trip is more probable at 70% power at Beginning of Cycle).

←(DRN 03-2155, R13)

c) Provides a load to the NSSS to keep reactor power at a selected point when a load rejection to a level below a chosen threshold occurs. This reactor power level is selectable with a low limit of 15 percent and a high limit automatically determined by the capacity of the bypass valves available for automatic control and the turbine load at the time.

d) Prevents the opening of safety valves following a unit trip, and effects a smooth transition to hot standby conditions.

e) Maintains main steam pressure at the zero power value during hot standby.

f) Provides a means of manually controlling reactor coolant temperature during plant start-up.

g) Provides a means for removing NSSS stored energy, decay heat, and pump energy during cooldown.

h) Provides modulation of the valves in a programmed sequential manner, thus minimizing their operation time at small valve openings, and improving controllability.

i) Facilitates operation during turbine start-up, synchronization, and limited loading by automatically providing a sink for the excess reactor power.

j) Includes in its design the criterion that no single equipment failure nor operator error will cause the unwanted opening of more than one valve. The system is not required nor designed to meet the criteria of IEEE-279-1971.

k) Use of master control station, in addition to the individual valve control stations, gives this system the capability of bumpless, balanceless transfer from manual to automatic and from automatic to manual operation.

l) Simplifies testing and troubleshooting because complete test capability is built into the system.

→(EC-1037, R301)

m) Provides a means of manually releasing steam directly to the condenser during Turbine Inlet Valve Testing.

←(EC-1037, R301)

The SCBS receives its power from an uninteruptable source which is backed up by a battery system. This arrangement ensures that SCBS is available during system voltage transient.

Figure 7.7-7 shows a simplified functional block diagram of the SBCS which is described here.

The SBCS compares the measured steam header pressure with a pressure setpoint generated as a function of NSSS load and biased by pressurizer pressure to provide an error signal to a proportional plus integral plus derivative controller. ’Me output signal from the controller is fed through a master control station to individual valve group opening demand programmers. These programmers generate signals for modulation of the valve groups in a sequential manner. From the programmers, a separate signal is sent to each valve control station. The valve control signals are sent to the valve actuators through individual gates enabled by a redundant bypass demand signal. Thus the opening of no more than one valve is assured upon the occurrence of any single equipment failure.

A valve quick opening demand signal is generated whenever the size of the load injection is such that it cannot be accommodated with the normal valve modulation speed. This is decided on the basis of the magnitude and rate of change of a combination of the decrease in steam flow out of the steam generators and the increase in pressurizer pressure. This signal is produced in a redundant manner and is sent to the valves through independent channels, thus eliminating the possibility of opening more than one valve with any single equipment failure. The quick opening signal is fed to a valve actuator only when that valve control station is in the automatic mode. Because the SBCS is designed to provide the best possible transient response, the number of valves to which the quick opening signal is applied is made to be a function of the magnitude of the load rejection. The signal is completely blocked if a reactor trip occurs when the NSSS is operating at low temperature. In addition, the valves will not be opened if the condenser is unavailable due to a high back pressure condition.

A reactor power cutback is initiated in the event of a load injection which exceeds the capacity of the turbine bypass valves. Two redundant reactor power cutback signals are generated by the SBCS in the same manner as the bypass valve quick-opening signals. The reactor power cutback system initiates the dropping of preselected CEA groups.

If the reactor power cutback system is unavailable, the system is designed to provide a load rejection or loss of load reactor trip. A key switch is used to provide selection between reactor power cutback on load rejection or a loss of load reactor trip.

An automatic CEA withdrawal prohibit (AWP) signal is generated whenever an automatic bypass valve-opening demand signal exists since this implies the existence of excess energy in the NSSS.

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An automatic CEA motion inhibit (AMI) feature is also supplied as a part of the SBCS. This feature provides the capability of maintaining a high reactor power output after load rejections, such as those to low power levels (i.e., 10-15 percent) or turbine trips (below 63 percent power) which are likely to result from faults that may be promptly corrected. The purpose of this feature is to allow a quick reloading of the unit once the fault is corrected. If a load rejection (including turbine trip) to a level below a preset point (typically 15 percent) occurs, the RRS will be prevented from withdrawing or inserting CEAs when the reactor power output falls below the threshold. This threshold is manually selectable over the range between 15 percent and 63 percent. The system compares the available bypass valve capacity to the manually selected AMI threshold setpoint and

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automatically limits the effective AMI threshold to the lower of the two values. The AMI action is terminated when the turbine load increases to the existing reactor power level and automatic CEA control by the reactor regulating system resumes.

7.7.1.4.2 Main Turbine Controls

The Turbine Control System (TCS) has automatic control and trip devices necessary for operation and protection of the main turbine.

Turbine trip signals are provided to trip the unit upon occurrence of conditions which are potentially hazardous to the turbine or other plant equipment.

The TCS is designed to:

a) Automatically control the turbine generator power during all phases of normal operation.

b) Trip the turbine upon occurrence of conditions which could, if operation were to continue, cause equipment damage.

c) Provide a turbine runback signal upon occurrence of loss of one FW pump or loss of load.

The TCS is a digital electro hydraulic (DEH) system which controls the turbine automatically using a process control computer, servo-mechanism and hydraulic valve actuators. The computer represents the digital portion of the system, the servo hardware represents the electronic portion of the system and the valve actuators represent the hydraulic part of the system.

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During automatic operation the DEH control system sends output signals to the servo system which in turn positions the hydraulic valve actuators and controls turbine speed or load. A manual system is provided to permit operation of the turbine in the event of the non-availability of the computer.

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7.7.1.4.3 Turbine Trip Signals

The following conditions will cause a turbine trip:

a) Manual emergency trip

b) Turbine bearing oil pressure low

c) Condenser vacuum loss (see Subsection 15.2.1.3)

d) Rotor Vibration Excessive

e) Stator rectifier cooling water flow low

f) Seal oil differential pressure low

g) Differential Expansion Excessive

h) Exhaust Hood temperature high

i) Feedwater Heaters 5 & 6 level high

j) Reactor trip

k) Excessive thrust bearing wear

l) Turbine overspeed

m) Moisture separator-reheater drain tank level high

n) Turbine oil tank level low

o) DEH DC Power Bus under voltage

p) Generator Lockout relays tripped

Turbine trip will not necessarily cause the reactor to trip. On turbine trip the SBCS and Reactor Power Cutback System operate as described in Subsections 7.7.1.4.1 and 7.7.1.9. The logic diagram for turbine trip is provided in Section 1.7.

→(DRN 05-1393, R14-A)

Turbine setback, runback and inhibit signals interfaced with the NSSS are described in Subsection 7.7.1.9.

←(DRN 05-1393, R14-A)

Additional turbine control description is in Section 10.2.

7.7.1.5 <u>Core Operating Limit Supervisory System</u>

7.7.1.5.1 General

The Core Operating Limit Supervisory System (COLSS) consists of process instrumentation and algorithms implemented by the plant monitoring computer to continually monitor the limiting conditions for operation on:

a) Peak linear heat rate

b) Margin to DNB

c) Total core power

d) Azimuthal tilt

e) Axial shape index.

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The COLSS continually calculates DNB margin, peak linear heat rate, total core power, and azimuthal tilt magnitude and compares the calculated values to the limiting condition for operation on these parameters. If a limiting condition for operation is exceeded for any of these parameters, COLSS alarms are initiated by the plant monitoring computer and operator action is taken as required by Technical Specifications.
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The selection of limiting safety system settings, core power operating limits, and the azimuthal tilt operating limit are specified such that the following criteria are met:

a) No safety limit will be exceeded as a result of an anticipated operational occurrence

b) The consequences of postulated accidents will be acceptable

The RPS functions to initiate a reactor trip at the specified limiting safety system settings. The COLSS is not required for plant safety since it does not initiate any direct safety-related function during anticipated operational occurrences or postulated accidents. The Technical Specifications define the limiting conditions for operation (LCO) required to ensure that reactor core conditions during operation are no more severe than the initial conditions assumed in the safety analyses and in the design of the low DNBR and high local power density trips. The COLSS serves to monitor reactor core conditions in an efficient manner and provides indication and alarm functions to aid the operator in maintenance of core conditions within the LCOs given in the Technical Specifications.

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The COLSS algorithms are executed in the plant monitoring computer. The calculational speed and capacity of COLSS enables numerous separate plant operating parameters to be integrated into two more easily monitored parameters: (1) margin to a limiting core power (based upon margins to DNBR, peak linear heat rate and licensed power limits), and (2) azimuthal tilt. If COLSS were not provided, maintenance of reactor core parameters within the LCOs, as defined by the Technical Specifications, would be accomplished by monitoring and alarms on the separate non-safety related process parameters used in the COLSS calculations. Therefore, the essential difference in using COLSS in lieu of previous monitoring concepts is the integration of many separate process parameters into a few easily monitored parameters. The conciseness of the COLSS displays has distinct operational advantages, since the number of parameters that must be monitored by the operator is reduced.
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Sensor validity checks are performed by COLSS on those measured input parameters used in the COLSS calculations. The validity checks consist of checking sensor inputs against the following criteria:

a) Sensor out of range

b) Deviations between like sensors monitoring the same variable

The following options are available for dealing with non-valid sensors:

a) Automatic replacement of the failed sensor by a redundant sensor (when available).

b) Automatic algorithm alteration for certain functional inputs by the generation of software flags when no redundant sensors are available.

c) Substitution of like sensors plus penalty factors (performed under administrative control).

d) Substitution of constants for selected COLSS inputs (performed under administrative control).

In the event that a non-valid sensor is detected, an alarm to the operator is actuated and corrective action is automatically initiated (where applicable).

A more detailed discussion of sensor validity checks is included in Reference 1.

Detailed process testing of COLSS is performed to ensure proper system performance. While COLSS is in the test mode, sets of stored constants are substituted for live sensor inputs and COLSS is executed. Comparison of the results with known outputs is then made to ensure continuity of all COLSS algorithms. While COLSS is in the operational mode, hard copy printout of the COLSS inputs, intermediate calculated values and results, is available on operator demand. Testing is performed under administrative control on a periodic basis to assure proper performance of COLSS. Since COLSS is not required for plant safety, COLSS testing requirements are not included in the Technical Specifications.

7.7.1.5.2 System Description

The core power distribution is continually monitored by COLSS, and a core power operating limit based on peak linear heat rate is computed. Operation of the reactor at or below this operating limit power assures that the peak linear heat rate is never more adverse than that postulated in the loss of coolant analyses.

Core parameters affecting the margin to DNB are continually monitored by COLSS, and a core power operating limit based on margin to DNB is computed. Operation of the reactor at or below this operating limit power level ensures that the most rapid DNB transient that can result from an anticipated operational occurrence does not result in a DNB reduction to a value less than the DNBR limit.

A core power operating limit based on licensed power level is also monitored by COLSS. Operation of the reactor at or below this operating limit ensures that the total core power is never greater than that assumed as an initial condition in the accident analyses.

The core power and the core power operating limits based on peak linear heat rate, margin to DNB, and licensed power level are continually provided by the plant computer through indication on the control board. The margin between the core power and the nearest core power operating limit is also displayed on the control board indicator. An alarm is initiated in the event that the COLSS calculated core power level exceeds a COLSS calculated core power operating limit.

In addition to the above calculations, the azimuthal flux tilt is calculated in COLSS. The azimuthal flux tilt is not directly monitored by the Plant Protection System (PPS)- An azimuthal flux tilt allowance, based on the maximum tilt anticipated to exist during normal operation is an input to the PPS and used in the low DNBR and high local power density trip calculations. The azimuthal flux tilt is continually monitored by COLSS and an alarm initiated in the event that the azimuthal flux tilt exceeds the azimuthal flux tilt allowance setting in the PPS.

The COLSS sensors provided for generating the input data required to compute the core power, the core power operating limits, and the azimuthal tilt are given in Table 7.7-1. A functional block diagram of COLSS is presented in Figure 7.7-8.

The algorithms below are executed in COLSS.

a) Reactor coolant volumetric and mass flowrate

b) Core power as determined by:

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1) Reactor coolant ΔT

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2) Secondary system calorimetric

3) Turbine first stage pressure

c) Core power distribution

d) Core power operating limit based on peak linear heat rate

e) Core power operating limit based on margin to DNB

The algorithms are executed in the plant monitoring computer. Technical Specifications on the reactor core provide alternate means of monitoring the limiting conditions for operation in the event that the plant monitoring computer is out of service.

Control board indication of the following COLSS parameters is continually available to the operator.

a) Core power operating limit based on peak linear heat rate

b) Core power operating limit based on margin to DNB

c) Total core power

d) Margin between core power and nearest core power operating limit

e) Core average axial shape index

COLSS alarms are initiated for the following conditions:

a) Core power exceeds a core power operating limit

b) Azimuthal flux tilt exceeds azimuthal flux tilt allowance

A description of COLSS algorithms, a COLSS uncertainty analysis, and a discussion of the treatment of COLSS input information is included in Reference 1. Table 7.7-1 provides a listing of the types, quantities, and ranges of sensors that provide input information for the COLSS algorithms.

7.7.1.5.3 Description of COLSS Algorithms

7.7.1.5.3.1 Reactor Coolant Volumetric Flowrate

The margin to DNB is a function of the reactor coolant volumetric flowrate. The four reactor coolant pump (RCP) rotational speed signals and four RCP differential pressure instruments are monitored by COLSS and used to calculate the volumetric flowrate. COLSS sets each RCP rotational speed value to a nominal value after positively determining the operation of the respective RCP. The pump characteristics are determined from testing conducted at the pump vendors' test facility and correlations between the pump rotational speed, pump differential pressure, and the volumetric flowrate are developed. Measurement uncertainties in the pump testing and COLSS measurement channel uncertainties are factored into the calculation of the pump volumetric flowrate. The four pump volumetric flowrates are summed to obtain the reactor volumetric flowrate. Necessary allowances for core bypass flow, flow factors, reactor coolant temperature, etc., are factored into the value of flow used in the DNBR calculation.

7.7.1.5.3.2 Core Power Calculation

The reactor coolant ΔT power, the secondary calorimetric power, and turbine power (based on a correlation of core power with turbine first stage pressure) are computed in COLSS. The reactor coolant ΔT power is a less complex algorithm than the secondary calorimetric power and is performed at a more frequent interval. The secondary calorimetric power is used as a standard to periodically adjust the pin coefficient on the calculation of the reactor coolant ΔT power. This arrangement provides the benefits of the secondary calorimetric accuracy and the reactor coolant ΔT power speed of computation.

The reactor coolant ΔT power is calculated based on the reactor coolant volumetric flowrate, the reactor coolant cold leg temperature, and the reactor coolant hot leg temperature. The reactor coolant ΔT power contains a dynamic term which provides a rapid indication of power changes during transients.

→(DRN 04-934, R13-B)

The secondary calorimetric power is based on measurements of feedwater flowrate, feedwater temperature, steam flow, steam pressure, steam generator pressure and blowdown flow. A detailed energy balance is performed for each steam generator. The energy output of the two steam generators is summed and allowances made for reactor coolant pump heat, pressurizer heaters, and primary and secondary system energy losses. The secondary calorimetric power is very accurate at steady state, but due to the system response characteristics is less accurate during transients.

←(DRN 04-934, R13-B)

The turbine power is calculated based on turbine first stage pressure. Turbine power provides a leading indicator of core power changes in response to load changes.

The best features of the ΔT power and turbine power measures are obtained by calibrating them to secondary calorimetric power in a manner such that, at steady state, the calibrated powers equal the more accurate secondary calorimetric power, and during transients, calibrated powers closely track their respective uncalibrated powers, affording the dynamic tracking ability of the latter. The auctioneered high of the two calibrated powers is selected as core power.

7.7.1.5.3.3 COLSS Determination of Power Distribution

The determination of F^n_q. F^n_r (see Subsection 4.3.2.2.2 for definition of terms), the power shape in the hottest channel, and the azimuthal tilt magnitude is performed based on in-core measurements of the flux distribution, processed by preprogrammed algorithms and stored constants. A brief description is given here of the data processing approach employed by COLSS to yield the desired power distribution information. This analysis is repeated at least once per minute, and thus represents continual on-line monitoring.

→(DRN 02-386)
The core is regarded as being divided into several radial regions in the X-Y plane. The regions are selected taking into account the locations of the regulating CEA groups, and the locations of the various generations of reload fuel.
←(DRN 02-386)

The dynamic response characteristic of the self-powered rhodium in-core detectors is a function of both prompt and delayed components of electrical current generated in the detector and cabling. The delayed portion of the current signal is governed by the decay of isotopes of Rh having halflives of 0.7 minutes and 4.4 minutes. To provide the capability to compensate for the delayed portion of the signal, the COLSS power distribution determination includes a compensation algorithm for the in-core signals used as input to COLSS. The algorithm approximately represents the inverse of the in-core detector dynamic response, such that the combination of detector response and dynamic compensation produces a signal closely representative of the actual neutron flux response. The basis for the dynamic compensation is described in CENPD-169.[(1)]

The capability for signal filtering is provided through selection of algorithm constants. With the capability for dynamic compensation and filtering on the in-core signals, changes in local flux level during operational load follow transients are adequately represented by the COLSS power distribution determination.

Following correction of the fixed detector signals for background and burnup five axially distinct region average power integrals, corresponding to the five Rh detector segments, are constructed for each X-Y region, taking into account signal-to-power conversion factors which are a function of burnup in the surrounding fuel and the location of the CEAS. Multi-node axial power shapes are constructed from the appropriate set of axial power integrals using fits based on combinations of simple nodal shapes.

Employing tables of factors relating power in the hot pin to the region average $F_q^{n,}$ F^n_r the axial power profile in the hot pin is computed for each region for use in the margin computation.

Malpositioning of a CEA or CEA group or the uncontrolled insertion or withdrawal of a CEA or CEA group will be detected by COLSS with inputs received from the pulse-counting CEA position indicating system described in Subsection 7.5.1.6.1. Should these deviations or out-of-sequence group movements occur, adjustments to planar radial peaking factor are performed to ensure that the COLSS, DNBR and peak linear heat rate calculations remain valid. It is noted that COLSS only provides a monitoring function. The protective action for the CEA related anticipated operational occurrences is provided by the CEAC/CPC system described in Section 7.2.

→(DRN 04-1097, R14)
Flux tilts are detected by comparison of signals from symmetrically located sets of fixed in-core detectors, at various levels in the core, and the flux tilts are included in the computation of margin to the power operating limit. In this way, postulated nonseparable asymmetric xenon shifts are identified and reflected in the power distribution assessment. Radial xenon redistributions are accounted for through the use of a regionwise combination of local-to-average and axial data discussed above. Alarms are provided by COLSS when the xenon tilt exceeds the allowances for these effects carried in the core protection calculators as penalties.
←(DRN 04-1097, R14)

If an inoperable fixed in-core detector is identified during internal consistency checks of the data, the inoperable detector is deleted from the COLSS calculations.

The COLSS power distribution algorithms, region selection, and initial set of stored constants are validated during the preoperational phase by analytical means, including processing in-core instrument signals produced by a 3-D reactor simulation code, and comparing the COLSS power distribution assessment to that of the 3-D simulator. This is done for steadystate and power maneuvering conditions, including simulations of conditions resulting from operator error.

→(DRN 01-1153, R12)
After the inception of operation, periodic confirmation of the COLSS assessment of the power distribution, including the suitability of any updated stored constants, is obtained by comparison with a more detailed, off-line process of an extensive in-core flux map produced by the fixed in-core instrument system. One means of analyzing the detailed flux map is to compare it with detailed calculations of the power distribution which include computations of the flux at the instrument location. Folding this together with other analyses of the ability of the detailed calculation to estimate the local pin-by-pin power distribution, enables an overall assessment of the COLSS power distribution error. This is factored into the margin assessment as noted in Subsection 7.7.1.5.4.
←(DRN 01-1153, R12)

7.7.1.5.3.4 Core Power Operating Limit Based on Peak Linear Heat Rate

The core power operating limit based on peak linear heat rate is calculated as a function of the core power distribution (F_q). The power level that results from this calculation corresponds to the limiting condition for operation of peak linear heat rate.

7.7.1.5.3.5 Core Power Operating Limit Based on Margin to DNB

The core power operating limit based on margin to DNB is calculated as a function of the reactor coolant volumetric flowrate, the core power distribution, the maximum value of the four reactor coolant cold leg temperatures, and the reactor coolant system pressure. The CE-1 correlation is used in conjunction with an iterative scheme to compute the operating

→(EC-13881, R304)
limit power level. (See Section 4.4. for a detailed discussion of the CE-1 correlation used in the CPC algorithm and a discussion on the current critical heat flux correlation reflected in the Technical Specifications and SAFDLs.) The power level that results from this calculation corresponds to the limiting conditions for operation on DNB margin.
←(EC-13881, R304)

7.7.1.5.4 Calculation and Measurement Uncertainties

The uncertainties in COLSS algorithms can be categorized as:

a) Uncertainties associated with the computation methods used to correlate the monitored variables to the calculated parameters

b) The measurement uncertainties associated with the COLSS process instrumentation

All algorithms used in COLSS will be biased such that, for a fixed set of input values, the calculated parameter in COLSS is no more adverse than the value of that same parameter calculated using standard design methods. Standard design methods are discussed in Chapters 4 and 15. Periodic checks are made throughout core life to ensure that this objective is achieved.

Each input sensor to COLSS is automatically checked in the program for an out-of-range condition before it is used in the COLSS calculation. In addition, allowances for measurement uncertainties are applied to the COLSS algorithm for each monitored variable to ensure COLSS is conservative.

7.7.1.6 Plant Computer System

The plant monitoring computer system is discussed in FSAR Section 7.5A.

7.7.1.7 In-Core Instrumentation System

The in-core instrumentation system is used to confirm core power distribution, perform periodic calibrations of the excore flux measurement system, and provide inputs to COLSS.

→(DRN 03-1873, R13; EC-18688, R304)
There are 280 rhodium in-core detectors spaced radially and axially to permit representative flux mapping of the core. The detectors are grouped into 56 instrument assemblies. Each assembly contains five self-powered detectors placed at selected locations and one background detector. Of the 56 assemblies, only 53 are active. Three assemblies have been installed with simulated (dummy) ICI assemblies.
←(DRN 03-1873, R13; EC-18688, R304)

The fixed in-core instrumentation system is designed to measure neutron flux at discrete locations within the core. The system includes both analog and digital processing to compensate for background and changes in detector sensitivity. The in-core measurements are processed by the plant monitoring computer utilizing specific application programs to obtain power distribution data.

Some planned uses of in-core data are:

a) To determine the gross power distribution in the core at different operating conditions over the range from 10 percent to 125 percent average reactor power

b) To estimate the fuel burnup in each fuel assembly

c) To evaluate thermal margins in the core

d) To check that care power distribution is consistent with calculated values

e) To calibrate and enhance correct operation of the reactor protection and control systems

f) To support COLSS.

→(DRN 03-1609, R13)

g) To detect some fuel misloadings

←(DRN 03-1609, R13)

When the incore detector instrumentation is used to perform the functions listed above, it must consist of:

1. At least 75% of all incore detectors operable (at least 210 detectors) with at least one incore detector in each quadrant at each level, and

2. At least 75% of all incore detector locations operable (at least 42 locations), and

3. A minimum of two quadrant symmetric incore detector locations per core quadrant.

→(DRN 03-1609, R13)

4. At least one incore detector location in each 4x4 array of adjacent fuel assemblies with at least three functional rhodium detectors (one each at any three of the five axial levels).

←(DRN 03-1609, R13)

If the minimum conditions are not met, the incore instrumentation is not used for the listed applicable monitoring or calibration functions.

An operable detector location consists of a fuel assembly containing a fixed detector string with a minimum of 4 operable rhodium detectors.

A quadrant symmetric incore detector location consists of any four fold symmetric group with all four detector locations operable.

Operability of the incore instrumentation system is demonstrated by the performance of a channel check within 24 hours prior to use and at least once per seven days thereafter when required for monitoring the azimuthal power tilt, radial peaking factors, local power density or DNB margin. In addition, operability is demonstrated at least once per 18 months by performance of a channel calibration operations which exempts the neutron detectors but includes all electronic components. The neutron detectors are calibrated prior to installation in the reactor core.

Redundancy is provided by the number of fixed in-core detector assemblies.

The fixed in-core detectors are used in the calibration of the excore neutron detectors by providing axial power distribution information.

The fixed in-core background detectors allow direct field measurements of the background effects on the signal from the cable in the active core.

The confirmation of power and calibration of excore detectors is done on a periodic basis as specified in Technical Specifications.

7.7.1.8 ExCore Neutron Flux Monitoring System

The excore neutron flux monitoring system includes neutron detectors located around the reactor core and signal conditioning equipment located in the control room area. Neutron flux is monitored from source levels through full power operation, and signal outputs are provided for reactor control, reactor protection, and for information display. Eight channels of instrumentation are furnished.

Two startup channels provide source level neutron flux information to the reactor operator for use during extended shutdown periods, initial reactor startup, and startups after extended periods of reactor shutdown, such as core refueling operations. Each channel consists of a dual section proportional counter assembly, with each section having multiple BF, proportional counters, a preamplifier located outside the reactor shield, and a signal processing drawer. This drawer contains power supplies, a logarithmic amplifier, and test circuitry. High voltage power to the proportional counters is automatically terminated several decades of neutron flux above the source level to extend detector life. These channels provide readout and audio count rate information but have no direct control or protective functions.

Four safety channels provide neutron flux information from routine startup neutron flux-levels to 200 percent of rated power covering a range of approximately $2x1O^{-8}$ to 200 percent (10 decades). Each channel consists of three fission chambers, a preamplifier, and a signal processing drawer containing power supplies, a logarithmic amplifier (including combination counting and mean square variation circuits), linear amplifiers, test circuitry, and a rate of change of power circuit. The Excore Channel Required for 1OCFR50 Appendix R requirements is mounted in the remote shutdown room in a cabinet beside LCP 43. In the event of a control room/cable vault fire, the Appendix R excore drawer is connected to the safety channel D preamplifier/filter assembly for logarithmic neutron flux indication.

These channels feed the RPS and provide information for rate of change of power display, DNBR, local power density, and overpower protection (see Subsections 7.2.1.1.1, 7.2.1.1.2, 7.2.1.1.3, and 7.2.1.1.4).

→

The fission chambers in each RPS channel are stacked vertically to permit an axial power shape measurement. Each chamber is operated in the current mode with each chamber feeding a linear amplifier to provide neutron flux information in the power operating range of 1 to 200 percent power. The center chamber, operating also in the pulse counting mode, feeds the wide range logarithmic amplifier (from a preamplifier) to provide single range neutron flux information from routine startup neutron flux levels to 200 percent of rated power.

←

Two control channels provide neutron flux information, in the power operating range of I percent to 125 percent, to the Reactor Regulating System for use during Automatic turbine load - following operation (see Subsection 7.7.1.1.1). Each control channel consists of two uncompensated ion chambers and a signal conditioning drawer containing power supplies, a linear amplifier, and test circuitry. These channels are completely independent of the safety channels.

The excore neutron detectors are mounted in holder assemblies and located in instrument wells (thimbles) external to the reactor vessel. The physical relationship of these detectors to the reactor vessel and cavity are shown in Section 1.7.

7.7.1.9 Reactor Power Cutback System

7.7.1.9.1 General

The NSSS normally operates with relatively minor perturbations in turbine/reactor power and feedwater/steam flow. These minor perturbations are normally accommodated by operator action and/or with the combined actions of the Reactor Regulating System (RRS), Steam Bypass Control System (SBCS), the Feedwater Control System (FWCS), and the Control Element Drive Mechanism Control System (CEDMCS). Certain large plant imbalances can occur however, such as a loss of one of two main feedwater pumps for which the normal operator or control system actions must be supplemented to assure that NSSS parameters are maintained within prescribed ranges. Under these conditions maintaining the NSSS within the control band ranges can be accomplished by rapid reduction of NSSS power at a rate which is greater than that provided by the normal high speed CEA insertion.

→(DRN 02-1478)

The Reactor Power Cutback System (RPCS) is a control system designed to accommodate certain types of imbalances by providing a "step" reduction in reactor power. The step reduction in reactor power is accomplished by the simultaneous dropping of one or more preselected groups of regulating CEAs into the core. The CEA groups are dropped in their normal sequence of insertion. The RPCS also provides control signals to rebalance turbine and reactor power following the initial reduction in reactor power. The system is designed to accommodate the loss of one feedwater pump.

←(DRN 02-1478)

7.7.1.9.2 System Description

The Reactor Power Cutback System (Figure 7.7-9) consists of two-out-of-two sensory logic, actuation logic to initiate the drop of Control Element Assemblies (CEAS) in a predetermined pattern, and turbine control logic.

The predetermined pattern of appropriate CEA groups for use in the reactor power cutback is accomplished via manual CEA selection.

The RPCS is actuated upon receiving coincident two-out-of-two sensory logic signals indicating either large turbine load rejection or loss of one or two main feedwater pumps. The actuation logic initiates the dropping of the preselected pattern of CEAS. Subsequent insertion of other groups either automatically by the Reactor Regulating System (RRS) or manually by the operator occurs as necessary. On loss of a feedwater pump a rapid turbine power reduction to 50 percent power is initiated, followed by a further reduction if necessary to balance turbine power with reactor power.

To enhance operational flexibility in the reactor power cutback system, a key switch design is used to provide the capability to select a loss of load reactor trip or a reactor power cutback on loss of load. Therefore, if the reactor power cutback system is unavailable, the operator can select a loss of load reactor trip. (Additional descriptions are provided in sections 7.2.1.1.1 and 7.7.1.4.1.)

7.7.2 DESIGN COMPARISON

The NSSS control systems are identical to the control systems provided for Arkansas Nuclear One-Unit 2 (ANO-2) (NRC Docket No. 50-368), except that the Waterford 3 Steam Bypass Control System includes six bypass valves with a load rejection capacity of 63 percent, all of which dump to the main condenser. In ANO-2, the system includes seven valves having a capacity of 85 percent. Three of the seven valves in ANO-2 dump to the main condenser, and the remaining four dump directly to atmosphere.

The BOP Control Systems are basically the same as St. Lucie Unit 1, with the exception of an expanded use of the plant monitoring computer.

7.7.3 ANALYSIS

The plant control systems and equipment are designed to provide high reliability during steady-state operation and anticipated transient conditions.

Operation of the control systems described in this section is not required in order to mitigate the consequences of the transients analyzed in Chapter 15. For each accident analysis in Chapter 15 an evaluation is made to determine the operational mode for each control system (i.e., manual or automatic) which will produce the most adverse results for the event under consideration. Each analysis in Chapter 15 then includes the assumption that these control systems are in that most adverse operational mode and that the control systems respond correctly, i.e., normally, for the input that is received in that mode. In addition, when the control system is in the automatic, malfunction of the system is evaluated in the process of establishing that single failure which produces the most adverse results for each accident so analyzed. The consequences produced by any credible malfunction of these control systems would be less severe than any which would be produced by the mechanisms considered as causes of the transients analyzed in Chapter 15.

SECTION 7.7: REFERENCES

→(DRN 03-2061, R14; EC-8460, R307)

1. Westinghouse Calculation, CN-OA-10-11, “Waterford-3 RSG COLSS Secondary Calorimetric Measurement Uncertainty”

←(DRN 03-2061, R14; EC-8460, R307)

2. C-E Topical Report, CENPD-145, "INCA Method of Analyzing In-Core Detector Data in Power Reactors," April 1975.

Table 7.7-1 Revision 304 (06/10)

COLSS MONITORED PLANT VARIABLES

Monitored Parameters	COLSS Sensors	Number of Sensors	Sensor Range
Core volumetric flow	RCP rotational speed	2 per pump	0 - 1,320 rpm[(b)]
	RCP differential pressure	2 per pump	0 - 150 psid
→(DRN 01-1153, R12; 04-744, R14)			
Core power			
Primary calorimetric	Cold leg temperature	1 per cold leg	525 - 625°F
	Hot leg temperature	1 per hot leg	525 - 625°F
Secondary calorimetric	Feedwater flow	1 per generator	0 - 9 x 10^6 lbm/hr
	Steam flow	1 per generator	0 - 9 x 10^6 lbm/hr
	Feedwater temperature	1 per generator	0 - 500F
	Steam pressure	1 per generator	0 - 1,200 psig
	Steam generator pressure	4 per generator	0 - 1,300 psia
	Blowdown Flow	1 per generator	0 - 450 gpm
←(DRN 01-1153, R12; 04-744, R14)			
→(DRN 03-1873, R13; EC-18688, R304)			
Core power distribution ←(DRN 03-1873, R13; EC-18688, R304)	In-core monitoring system	53 in-core assemblies each containing 5 axially stacked detectors, plus one background detector	NA[(a)]
	CEA position	1 per CEA	0 - 150 in.
Reactor coolant pressure	Pressurizer pressure	2 (on pressurizer)	1,500 - 2,500 psia
Turbine power	Turbine first stage pressure	2 (on turbine)	0 - 1,000 psia

(a) Core power distribution is provided in a graphic format.

(b) RPM set to nominal value of 1188 after pump to be positively determined as running.