SUPPLEMENTAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

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Question No. 04.03-4

REQUIREMENTS AND GUIDANCE

10 CFR Part 50 Appendix A, General Design Criterion (GDC) 10 requires the reactor core design to include appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or anticipated operational occurrences (AOOs). GDC 11, “Reactor Inherent Protection,” requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity. GDC 20, “Protection System Functions,” requires automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and that automatic operation of systems and components important to safety occurs under accident conditions. In addition, GDC 28, “Reactivity Limits,” requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor cause sufficient damage to impair significantly the system’s capability to cool the core.

To assess compliance with these requirements, Section 4.3 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) guides the staff to review the applicant’s analysis of reactivity coefficients and power distributions for “steady-state operations and allowed load-follow transients.” In addition, SRP Section 15.0 guides reviewers to ensure that the application “specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.”

ISSUE

Noting that the applicant refers to load-follow operations and transients in DCD Sections 4.3 and 4.4, the staff is concerned that the application lacks much of the information that would

necessary for approving the APR1400 design for load-follow operations. For example, the DCD does not specify the ranges of allowed load-follow power maneuvers (e.g., power swings, power ramp rates), does not detail how load-follow power maneuvers and resulting xenon transients would be controlled with rods versus soluble boron, and does not include analyses of core and system transients associated with load-follow operations. Moreover, the analyses presented in DCD Chapter 15 do not explicitly consider transient load-follow operational conditions in determining the most limiting initial operating conditions for analyzed transients and accidents.

INFORMATION NEEDED

In its response, the applicant should either provide all information necessary for the consideration of load-follow operations or else state that it is seeking approval of the APR1400 design only for defined baseload operations. The applicant should then revise the affected parts of the DCD and its incorporated references accordingly.

Response

APR1400 is basically requested for approval for baseload operation. However, it does not exclude a possible power change necessary for plant operation. For some instances, reactor power needs to be decreased to lower level due to plant conditions. Such plant operations can be divided into two categories, scheduled load maneuver and unexpected load transients such as load rejection. Power maneuvers with schedule load maneuver shall be performed within the limits of Technical Specifications (TS) and the safety analyses are not affected by such operations. Plant conditions during unexpected load transient can be outside of the limits of TS but they are restored automatically or manually after some periods of time. If the limits of TS are not met, power shall be decreased as required in TS.

Subsection 4.4.3.4 in DCD Tier 2 will be revised to clearly describe that APR1400 is requested for approval for baseload operation. Also, Subsection 4.3.2.2.3 in DCD Tier 2 will be revised for clarification.

Supplemental Response

Load-following can be used in various manners. When it is used in comparison with base load, it is the load change operation depending on the electrical grid. This type of operation is characterized by continuous change in reactor power. Load-following is sometimes used in a wider meaning including the capability of reactor power change. Reactor power of APR1400 can be controlled automatically or manually. This type of operation by the reactor operator can be defined as normal power maneuvering operation. The APR1400 also has the loss of load capability up to full load rejection. Such operation is an unexpected load transient. Section C.I.4.4.3.5 in RG 1.206 requires that load-following characteristics are to be described. We think that it requires the control characteristics of the plant for load change. The APR1400 is base load plant but it has the load change capability for normal power maneuvers and unexpected load transients as described above.

Subsection 3.9.1.1 in DCD Tier 2 will be revised to clearly describe that APR1400 is requested for approval for baseload operation. Subsection 5.3.2, 5.4.10.1, 7.7.1.1, 11.1.1.1 and 11.2.2.2 in DCD Tier 2 will also be revised for clarification.

Impact on DCD

DCD Tier 2, Subsection 4.3.2.2.3 will be revised as shown in Attachment 1.

DCD Tier 2, Subsection 4.4.3.4 will be revised as shown in Attachment 2.

DCD Tier 2, Subsection 3.9.1.1, 5.3.2, 5.4.10.1, 7.7.1.1, 11.1.1.1 and 11.2.2.2 will be revised as shown in Attachment 3.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

It is shown in the following subsections that reactor operation within these design limits is achievable.

4.3.2.2.3 Expected Power Distributions

Figures 4.3-4 through 4.3-18 show planar average radial power distributions of a typical first cycle and Figures 4.3-19 through 4.3-24 show unrodded core average axial power distributions, respectively. They illustrate conditions expected at full power for various times in the fuel cycle. It is expected that for normal, base load operation of the plant, reactor operation will be with limited CEA insertion so that the unrodded power distributions in Figures 4.3-4 through 4.3-24 represent the expected power distribution during most of the cycle. Normal operation of the reactor may include full insertion of the part-strength CEA group during ~~load transients~~ turbine load changes.. Therefore, Figures 4.3-4 through 4.3-18 show radial power distributions both for unrodded steady-state operation with the part-strength CEA group fully inserted. The expected three-dimensional peaking factor, F_q^n, during steady-state operation, is the product of the planar radial peaking factor and the axial peaking factor.

Figures 4.3-25 through 4.3-27 show a loading pattern and planar radial power distributions of typical equilibrium cycle based on a refueling interval of approximately 18 months. The expected power distributions for this cycle are similar to those of the first cycle except for the reduced power in fuel assemblies located on the periphery of the core and consequently higher radial peaking factors in the interior region of the core. The expected power distributions are well within the nuclear design limits described in Subsection 4.3.2.2.2. The uncertainty associated with these calculated power distributions is described in Subsection 4.3.3.1.2.2.2.

The capability of the core ~~to follow load transients~~ to follow turbine load changes without exceeding power distribution limitations depends on the margin to operating limits ~~during a load-follow operation~~ compared to the margin required for unrodded base-loaded operation. The radial and axial power distributions and estimates of F_q^n and F_r^n are obtained by using ROCS Code System (Reference 5).

The radial power distribution in an assembly is a function of the location of the assembly in the core, the time in the fuel cycle, CEA insertion, and other considerations. A normalized assembly power distribution used for a sample DNB calculation is described in Subsection

power maneuvers.

power conditions. Soluble boron can also be used to compensate for changes in reactivity due to power level changes and minor changes in reactivity that may occur during normal reactor operation. Twelve part-strength CEAs are provided in the design. A major function of the part-strength CEAs is to assist in the control of core power distribution, including suppression of xenon-induced axial power oscillations during power operations, and control of axial power shape during load-following transients. The part-strength CEAs can also provide reactivity control to compensate for minor variations in moderator temperature and boron concentration during power operations and to assist in compensating for changes in reactivity from power level and xenon during load-following transients. The reactivity worth of the part-strength CEA groups is shown in Table 4.3-6.

power maneuvers.

4.3.2.4.3 Shutdown Reactivity Control

The reactivity worth requirements of the full complement of CEAs is determined primarily by the power defect, the excess CEA worth with the stuck-rod criteria described in Subsection 4.3.1.9, and the total CEA reactivity allowance for the cycle. Table 4.3-8 shows the reactivity component allowances. These data are based on EOC conditions, when the fuel and moderator temperature coefficients are most negative and the shutdown reactivity requirement is at maximum. Each allowance component is described in subsequent subsections. No CEA allowance is provided for xenon reactivity effects because these effects are controlled with soluble boron rather than with CEAs.

As shown in Table 4.3-9 for the end of the first cycle, the worth of all CEAs except for the most reactive, which is assumed to be stuck in the fully withdrawn position, provides shutdown capability required by the total reactivity allowance shown in Table 4.3-8. The margin is sufficient to compensate for calculated uncertainties in the nominal design allowances and in the CEA reactivity worth. Therefore, the shutdown reactivity control provided in this design is sufficient at all times during the cycle.

4.3.2.4.3.1 Fuel Temperature Variation

The increase in reactivity that occurs when fuel temperature decreases from the full-power value to the zero-power value is due primarily to the Doppler effect from U-238. The CEA reactivity allowance for fuel temperature variation, shown in Table 4.3-8, is a conservative allowance for the EOC conditions.

Power operation with inoperative pumps is not allowed in the APR1400. The adequacy of natural circulation for decay heat removal after reactor shutdown is described in Subsection 5.4.7.3.1.

Power Maneuvering

4.4.3.4 ~~Load Following~~ Characteristics

APR1400 will be operated as a base load plant. When the change in reactor power is required, APR1400 provides automatic and manual control capabilities.

Reactivity during ~~load following~~ power maneuvers is controlled by maneuvering of the CEAs and boron concentration in the RCS. When load changes are initiated, the reactor regulating system (RRS) senses a change in the turbine power and positions CEAs to attain the programmed average coolant temperature. RCS boron concentration can also be adjusted to attain the appropriate coolant temperature. The feedwater system uses a controller that senses changes in steam flow, feedwater flow, and water level and acts to maintain steam generator level at the desired point. The pressurizer pressure and level control systems respond to deviations from preselected setpoints caused by the expansion or contraction of the reactor coolant and actuate the spray or heaters and the charging or letdown systems as necessary to maintain pressurizer pressure and level. The steam pressure increase of the steam generator due to load rejection or rapid load reduction can be controlled by opening the turbine bypass valves as necessary to maintain steam pressure within a certain setpoint.

4.4.3.5 Thermal and Hydraulic Characteristics Summary Table

Principal thermal and hydraulic characteristics of the RCS components are listed in Table 4.4-9.

4.4.4 Evaluation

4.4.4.1 Critical Heat Flux

The margin to CHF or DNB is expressed in terms of the DNBR. The KCE-1 correlation (Reference 5) was used with the TORC and CETOP computer codes (References 9 and 12) to determine DNBR values for normal operation and AOOs. The KCE-1 correlation was developed in conjunction with the TORC code for DNB margin predictions for the 16 × 16 PLUS7 fuel assemblies with R-type split mixing vane spacer grids. The correlation is based on data from tests conducted for PLUS7 fuel development at the Columbia University chemical engineering research laboratories. The tests used electrically heated

characteristics of the event. The following are the event descriptions for specific events considered in Service Level A conditions.

a. Steady-state operation with normal NSSS parameter variations in the increasing and decreasing directions.

The plant could undergo primary and secondary process parameter variations because of secondary steam conditions. Power step changes of 10 percent are used to envelop this event. This is conservative since the 10 percent power step change produces more severe plant process parameter variations than those that occurred during any normal plant variations as the result of changing steam conditions.

Each event is assumed to occur 1,500,000 times during the 60-year plant design life.

b. Daily load follow operation

For daily load follow operation, the power is maintained at 100 percent for 10 through 16 hours, ramped down from 100 percent to 50 percent over a 2-hour period, operated for 4 through 10 hours at 50 percent, and then ramped up from 50 percent to 100 percent power over a 2-hour period.

Each event is assumed to occur 22,000 times during the 60-year plant design life.

c. Turbine power step changes of 10 percent power (15 to 100 percent power)

This event is a turbine power change from 100 to 90 percent power and from 90 to 100 percent power. The transients of step change from 25 to 15 percent power and step change from 15 to 25 percent power are also considered. These power step changes are representative of other power levels and serve to envelop smaller power step changes.

Each event is assumed to occur 3,200 times during the 60-year plant design life.

Although APR1400 will be operated as a base load plant, the effects of daily load follow operation are accounted for in the structural design and analysis of ASME Code Class 1 components, reactor internals, and component supports.

o. Manual operation of the auxiliary spray system

The manual operation of the auxiliary spray system may be required during power operation to reduce primary pressure excursions when the main spray system is out-of-service. This transient may be especially severe for the pressurizer spray line since isolated fluid in the auxiliary spray line may cool down to low temperatures before being sprayed into the pressurizer.

This event is assumed to occur 250 times during the 60-year plant design life.

p. High-capacity steam generator blowdown

The steam generator high capacity blowdown is performed with a flow rate of approximately 5 percent of the steam generator maximum steaming rate to maintain steam generator chemistry within control limit.

This event is assumed to occur 3,200 times during the 60-year plant design life.

q. Shift from normal to maximum CVCS flow rate

The chemical and volume control system (CVCS) letdown and charging flow rates may be increased to support ~~daily load follow operations~~ required changes in boron concentration or to more rapidly reduce impurities in the RCS.

This event is assumed to occur 3,200 times during the 60-year plant design life.

r. Low-low VCT level and charging pump diversion to the boric acid storage tank

Low volume control tank (VCT) level results in diverting the charging flow sources from VCT to the boric acid storage tank.

This event is assumed to occur 60 times during the 60-year plant design life.

Table 3.9-1 (1 of 7)

(2)
(2)

Transients Used in Stress Analysis

Event Conditions	Event Description	Specific Events	Occurrences for Design Purpose 60 Years[1]	40 Years[1]
Level A Service Conditions				
Normal Event-1A	Steady-state operation with normal parameter variations in the increasing direction (5 to 100 %)		1,500,000	1,000,000
Normal Event-1B	Steady-state operation with normal parameter variations in the decreasing direction (5 to 100 %)		1,500,000	1,000,000
Normal Event-2A	Daily load follow operation (from 100 to 50 % power)		22,000	15,000
Normal Event-2B	Daily load follow operation (from 50 to 100 % power)		22,000	15,000
Normal Event-3A	Turbine step load change in the increasing direction	Turbine power steps of +10 percent (15 to 100 % power)	3,200	2,000
		Turbine power steps of +1 percent (5 to 15 % power)	1,600	1,000
		(Total)	4,800	3,000
Normal Event-3B	Turbine step load change in the decreasing direction	Turbine power steps of -10 percent (15 to 100 % power)	3,200	2,000
		Turbine power steps of -1 percent (5 to 15 % power)	1,600	1,000
		(Total)	4,800	3,000
Normal Event-3C	Large turbine load step decrease	Turbine load rejection up to 50 % (50 to 100 % power)	60	40
		Turbine generator runback to house load	60	40
		Reactor trip	150	100
		Turbine trip	150	100
		(Total)	420	280

Table 3.9-1 (7 of 7)

Event Conditions	Event Description	Specific Events	Occurrences for Design Purpose	
			60 Years(1)	40 Years(1)
Test Conditions				
RCS hydrostatic test			15	10
Secondary hydrostatic test			15	10
RCS leak test			200	200
Secondary leak test			200	200
SIS/SCS preoperational and maintenance test			360	240
SIS/SCS check valve operability test			120	80

(1) The design life for RCS main components and Class 1 piping is 60 years, and the design life for Class 2 and 3 piping and other components except RCS main components is 40 years.

(2) Although APR1400 will be operated as a base load plant, the effects of daily load follow operation are accounted for in the structural design and analysis of ASME Code Class 1 components, reactor internals, and component supports.

surfaces at installation to further enhance anti-galling properties. Field experience to date has shown no evidence of deleterious breakdown of either phosphate coating or lubricant.

The stud holes in the vessel flange are protected from the refueling water using the seal plugs prior to refueling activity.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper Shelf Energy Data and Analyses

normal power maneuver, loss of load transient.

All components in the RCS are designed to withstand the effects of cyclic loads due to RCS temperature and pressure changes. These cyclic loads are introduced by ~~normal load transients~~, reactor trips, and startup and shutdown operation. The design number of cycles for heatup and cooldown is based on a rate of 55.6 °C/hr (100 °F/hr). During unit startup and shutdown, the rate of temperature change is limited to less than 55.6 °C/hr (100 °F/hr) by administrative procedure. The maximum allowable RCS pressure at the corresponding minimum allowable temperature is based upon the stress limitations for brittle fracture. These limitations are derived using linear elastic fracture mechanics principles, the procedures prescribed by ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," NRC RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," NRC SRP BTP 5-3, "Fracture Toughness Requirements," and the procedures recommended by Welding Research Council (WRC) Bulletin 175, "Pressure Vessel Research Committee (PVRC) Recommendations on Toughness Requirements for Ferritic Materials" (Reference 28). The reactor vessel is also designed, fabricated, erected, and tested to conform with the requirements of 10 CFR 50.55a, 10 CFR 50.60, and 10 CFR Part 50, Appendix A (GDC 1, 14, 31, and 32).

5.3.2.1 Pressure-Temperature Limitation Curves

5.3.2.1.1 Material Properties

Pressure-temperature limitations (P-T limits) are determined using material property test data for reactor coolant pressure boundary materials, as required by ASME Section XI, Appendix G. Based on considerations of existing material property test data, an initial

e. Provide sufficient steam volume to allow acceptance of the insurge resulting from any loss of load transient without liquid or two-phase flow reaching the pressurizer pilot-operated safety relief valve nozzles.

f. Minimize the total reactor coolant mass change and associated charging and letdown flow rates in order to reduce the quantity of wastes generated by ~~load follow operations~~. reactor power changes.

g. Provide sufficient pressurizer heater capacity to heat up the pressurizer, filled with water at the zero power level, at a rate that provides reasonable assurance of a pressurizer temperature (and thus pressure) that will maintain an adequate degree of subcooling of the water in the reactor coolant loop as it is heated by core decay heat and/or pump work from the reactor coolant pumps.

h. Contain a total water volume that does not adversely affect the total mass and energy released to the containment during the maximum hypothetical accident.

i. Provide reasonable assurance that, in addition to being specified as seismic Category I, the pressurizer vessel, including heaters, baffles, and supports is designed so that no damage to the equipment is caused by the frequency ranges of 19-24 cps and 118-143 cps. The lower frequency is produced by vibratory excitations associated with RCP rotating speed. The design basis for the higher frequency consists of a pressure pulse of 0.56 kg/cm^2 (8 psi), which diminishes internally within the vessel.

j. Maintain the pressurizer at normal operating pressure during hot standby conditions by taking into account the energy balance of maximum heat loss from the pressurizer and the pressurizer heater capacity. This capability is provided by redundant trains of heaters powered from off-site power and Class 1E emergency power. The Class 1E emergency power is used if off-site power is not available.

k. Maintain sufficient spray flows to keep the pressure below the reactor trip setpoint during ~~maneuvering and load follow operations~~ and loss of load transients.

power maneuvers

within limits for power maneuvers.

The boric acid concentration can be used ~~within limits to load follow~~. Since boric acid concentrations changes occur slowly, operator action is acceptable for boric acid concentration control. The CEAs can be controlled manually by the operator or automatically to maintain the programmed reactor coolant temperature and power level during boric acid concentration changes within the limits of CEA travel.

The PCS integrates the control systems that control the reactor power level, which include the RRS, DRCS, and RPCS.

Providing control limits and interlocks prevents abnormal power and temperature conditions that could result from excessive control rod withdrawal initiated by a control system malfunction or by an operator.

Table 7.7-2 summarizes the DRCS control limits and interlocks.

changes

The RRS automatically adjusts reactor power and reactor coolant temperature to follow the turbine load ~~transients~~ within the established limits. Figure 7.7-1 shows that the RRS receives a turbine load index signal (linear indication of load) and reactor coolant temperature signals.

The desired average temperature is determined by a reference temperature (T_{REF}) program that is inputted with the turbine load index. The hot leg and cold leg temperature signals are averaged (T_{AVG}) in the RRS. The T_{REF} signal is then subtracted from the T_{AVG} signal to provide a temperature error signal. Power range neutron flux is subtracted from the turbine load index to provide compensation to the $T_{AVG} – T_{REF}$ error signal generated.

This resulting error signal is fed to a CEA rate program to determine whether the CEAs are to be adjusted at a high or low rate, and to a CEA motion demand program that determines if the CEAs are to be withdrawn, inserted, or held. The outputs of the rate and motion demand programs are used by the DRCS.

If the temperature error signal is high (i.e., T_{AVG} is higher than T_{REF}, or the cold leg temperature (T_{COLD}) is higher than a limit), the RRS provides an automatic withdrawal prohibit (AWP) signal to the DRCS. The withdrawal of CEAs causes the T_{AVG} to increase. Prohibiting a withdrawal prevents an increase in the error signal.

The boronometer detects the boron concentration by passing reactor coolant around a neutron source. Around the source are BF_3 neutron detectors. As the boron concentration decreases, the neutron flux increases.

At power, the boron concentration and the CEA position affect reactor coolant temperature. Because of the long time required to change the boron concentration, the boron is used for long-term effects such as fuel burnup and fission product build up. Boron concentration control is also used for ~~load following~~ power maneuvers. By adjusting the boron concentration, the CEAs can be withdrawn to provide an adequate shutdown margin. Boron control is provided by use of the P-CCS.

g. In-core instrumentation system

The in-core instrumentation system consists of core exit temperature (CET) instrumentation and in-core nuclear instrumentation.

The CET instrumentation consists of 61 thermocouples at fixed core outlet positions that measure the fuel assembly coolant outlet temperatures in the core.

Likewise, the in-core nuclear instrumentation consists of fixed in-core neutron flux detectors that are spaced radially and axially in sufficient numbers to permit the representative flux mapping of the entire core.

The in-core nuclear instrumentation is used to monitor the core power distribution, and the detectors are fixed in place at all times during operation.

There are 61 fixed in-core neutron flux detector assemblies with 5 self-powered Rhodium detectors and 1 background detector in each location. The 61 assemblies are distributed in the reactor core to optimize a core power distribution monitoring capability.

The five Rhodium detectors are axially distributed along the length of the core at 10, 30, 50, 70, and 90 percent of core height. This permits representative three-dimensional flux mapping of the core.

The Rhodium detectors produce a delayed beta current proportional to the neutron activation of the detectors that is proportional to the neutron flux in the detector region.

and following reactor refueling operations. The drawers have no direct control or protection functions.

Two control signal processing drawers consist of linear amplifier and test circuitry. Each drawer provides the neutron flux information in the power operating range of 0 percent to 125 percent to the RRS ~~for use during automatic turbine load-following operation~~. to follow the turbine load changes.

Startup and control signal processing drawers are independent of the safety channels.

i. Boron dilution alarm system

Reactivity control in the reactor core is affected, in part, by soluble boron in the RCS. The boron dilution alarm system (BDAS) utilizes the ENFMS startup channel neutron flux signals to detect a possible inadvertent boron dilution event while in Modes 3-6. The BDAS has two separate channels to provide reasonable assurance of detection and alarming of the event, and alarm signals are provided to the QIAS-N and the IPS.

When neutron flux signals increase (during Modes 3-6) to equal or greater than the calculated alarm setpoint, alarm signals are generated. The alarm setpoint is periodically lowered automatically to be a fixed amount above the current neutron flux signal setpoint. The alarm setpoint only follows decreasing or steady flux levels, not an increasing signal. The current neutron flux indication and alarm setpoint are available on the operator console in the MCR. There is also a reset capability to allow the operator to acknowledge the alarm and reinitialize the system.

j. Turbine control system

The turbine control system is described in Subsection 10.2.2.

k. P-CCS

The P-CCS controls non-safety components such as pumps, valves, heaters, and fans. The P-CCS sends process variables and P-CCS status information to the IPS and QIAS-N. In case the P-CCS is unavailable, non-safety signals for monitoring

11.1.1 Design Basis Source Term

11.1.1.1 Fission Product Activities in the Reactor Coolant

The DAMSAM Code (Reference 4) is used to calculate the design basis source term of fission products in the reactor coolant for the design of the radioactive waste management system and to determine design lifetime integrated doses for plant equipment. The isotopes considered in the maximum case are those that are significant for design purposes by reason of a combination of energy, half-life, and/or abundance.

The mathematical model used to determine the concentration of nuclides in the reactor coolant system (RCS) involves a group of linear, first-order differential equations. These equations are obtained by applying a mass balance for production and removal in both the fuel pellet region and the reactor coolant region.

In the fuel pellet region, the mass balance includes fission product production by direct fission yield, by parent fission product decay, and by neutron activation, while removal includes decay, neutron activation, and escape to the reactor coolant.

In the reactor coolant region, the fission product is introduced when it escapes from the fuel pellet through defective fuel rod cladding, parent decay in the reactor coolant, and neutron activation of the fission products in the reactor coolant. The fission products are removed by decay; coolant purification; boron feed-and-bleed operations (to accommodate fuel burnup); leakage and other feed-and-bleed operations during startups, shutdowns, and ~~load following operation~~; and neutron activation.

power maneuvers

The expression to determine the fission product inventory in the fuel pellet region is:

$$\frac{dN_{c,i}}{dt} = (F)(Y_i)(P) + (f_{i-1}\lambda_{i-1})N_{p,i-1} + \sigma_j\phi N_{p,j} - (\lambda_i + Dv_i + \sigma_i\phi)N_{p,i}$$

(Eq. 11.1-1)

The expression to determine the fission product inventory in the reactor coolant region is:

$$\frac{dN_{c,i}}{dt} = (D)(v_i)(N_{p,i}) + (f_{i-1}\lambda_{i-1})N_{c,i-1} + (\sigma_j\phi CVR)\,N_{c,j}$$

$$-(\dot{Q}\lambda_i + \frac{\dot{Q}}{w}\eta_i + \frac{(1-\eta_i)\dot{C}}{C_0 - t\dot{C}} + \frac{L}{W} + \sigma_i\phi CVR)\,N_{c,i} \quad \text{(Eq. 11.1-2)}$$

The chemicals are then mixed with the tank contents using the recirculation/mixing mode, followed by sampling and further chemical addition, if necessary. This process is repeated until the tank contents meet the required fluid pH for discharge.

11.2.2.2 Monitoring and Discharge

LWMS monitor tanks collect liquid processed through the R/O package. Following the sample analysis as described in Subsection 11.2.2.1.4, the operator determines where the contents of the monitor tank are to be transferred. If the water quality and radionuclide concentrations of the contents in the monitor tank meet the water specifications for the holdup tank, and the plant ~~load following~~ operation is not affected by the recycle operation, the contents of the monitor tank are transferred to the holdup tanks for plant reuse. If the water is not recycled and it is determined to be acceptable for offsite release, the contents of the monitor tank are discharged to the offsite release point.

The LWMS is designed to control the release of the treated effluent. The effluent is stored in the monitor tank and is sampled and analyzed to confirm that the release nuclide concentrations are within the limits of 10 CFR 20, Appendix B, Table 2 (Reference 3), prior to release. The LWMS is designed with recycle capability for further treatment for any batch that exceeds the release specifications (Table 11.2-1 for normal operation). The release is also continuously monitored by dual in-line radiation monitors (RE-183 and RE-184) during release. Any portion of the flow that exceeds the predetermined setpoint will trigger alarms in the MCR and the radwaste control room for operator actions, simultaneously turn off the monitor tank pump, and close the effluent discharge valve that is under supervisory control. The design and setpoints of the radiation monitors are described in Section 11.5. The LWMS is designed with no release bypass.

11.2.2.3 Component Description

The LWMS components are determined for the radioactive safety classification in accordance with the guidance provided in NRC RG 1.143 (Reference 1). The component safety classification is summarized in Table 11.2-6. Accordingly, the LWMS is classified as RW-IIa, based on the highest safety classification for the components within the system boundary. The LWMS components are housed within the compound building, which has been determined to be RW-IIa.