1.42 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.

1.43 CORE OPERATING LIMITS REPORT

The Oyster Creek CORE OPERATING LIMITS REPORT (COLR) is the document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.f. Plant operation within these operating limits is addressed in individual specifications.

1.44 LOCAL LINEAR HEAT GENERATION RATE

The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) shall be applicable to a specific planar height and is equal to the AVERAGE PLANAR LINEAR GENERATION RATE (APLHGR) at the specified height multiplied by the local peaking factor at that height.

1.45 SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN is the amount of reactivity by which the reactor would be subcritical when the control rod with the highest reactivity worth is fully withdrawn, all other operable control rods are fully inserted, all inoperable control rods are at their current position, reactor water temperature is 68°F, and the reactor fuel is xenon free. Determination of the control rod with the highest reactivity worth includes consideration of any inoperable control rods which are not fully inserted.

1.46 IDLE RECIRCULATION LOOP

A recirculation loop is idle when its discharge valve is in the closed position and its discharge bypass valve and suction valve are in the open position.

1.47 ISOLATED RECIRCULATION LOOP

A recirculation loop is fully isolated when the suction valve, discharge valve and discharge bypass valve are in the closed position.

E. Reactor Coolant Quality

1. The reactor coolant quality during power operation with steaming rates to the turbine-condenser of less than 100,000 pounds per hour shall be limited to:

conductivity	2 us/cm [s=mhos at 25°C (77°F]
chloride ion	0.1 ppm

2. When the conductivity and chloride concentration limits given in 3.3.E.1 are exceeded, an orderly shutdown shall be initiated immediately, and the reactor coolant temperature shall be reduced to less than 212°F within 24 hours.

3. The reactor coolant quality during power operation with steaming rates to the turbine-condenser of greater than or equal to 100,000 pounds per hour shall be limited to:

conductivity	10 uS/cm	[S=mhos at 25°C (77°F)]
chloride ion	0.5 ppm	

4. When the maximum conductivity or chloride concentration limits given in 3.3.E.3 are exceeded, an orderly shutdown shall be initiated immediately, and the reactor coolant temperature shall be reduced to less than 212°F within 24 hours.

5. During power operation with steaming rates on the turbine-condenser of greater than or equal to 100,000 pounds per hour, the time limit above 1.0 uS/cm at 25°C (77°F) and 0.2 ppm chloride shall not exceed 72 hours for any single incident.

6. When the time limits for 3.3.E.5 are exceeded, an orderly shutdown shall be initiated within 4 hours.

F. Recirculation Loop Operability

1. During POWER OPERATION, all five recirculation loops shall be OPERATING except as specified in Specification 3.3.F.2.

2 POWER OPERATION with a maximum of two IDLE RECIRCULATION LOOPS or one IDLE RECIRCULATION LOOP and one ISOLATED RECIRCULATION LOOP is permitted. The reactor shall not operate with two ISOLATED RECIRCULATION LOOPS.

a. With one ISOLATED LOOP the following conditions shall be met:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) as a function of average planar exposure, at any axial location shall not exceed 98% of the limits specified in 3.10.A. The action to bring the core to 98% of the APHLGR limits shall be completed prior to isolating the recirculation loop.

2. The circuit breaker of the recirculation pump motor generator set associated with an ISOLATED RECIRCULATION LOOP shall be open and defeated from operation.

3. An ISOLATED RECIRCULATION LOOP shall not be returned to service unless the reactor is in the COLD SHUTDOWN condition.

b. When there are two inoperable recirculation loops (either two IDLE RECIRCULATION LOOPS or one IDLE RECIRCULATION LOOP and one ISOLATED RECIRCULATION LOOP) the reactor core thermal power shall not exceed 90% of rated power.

3. If Specifications 3.3.F.1 and 3.3.F.2 are not met, an orderly shutdown shall be initiated immediately until all operable control rods are fully inserted and the reactor is in either the REFUEL MODE or SHUTDOWN CONDITION within 12 hours.

4. With reactor coolant temperature greater than 212°F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the full open position.

5. If Specification 3.3.F.4 is not met, immediately open one recirculation loop discharge valve and its associated suction valve.

6. With reactor coolant temperature less than 212°F and irradiated fuel in the reactor vessel, at least one recirculation loop discharge valve and its associated suction valve shall be in the full open position unless the reactor vessel is flooded to a level above 185 inches TAF or unless the steam separator and dryer are removed.

G. Primary Coolant System Pressure Isolation Valves

Applicability:

Operational conditions - Startup and Run Modes; applies to the operational status of the primary coolant system pressure isolation valves.

Objective:

To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an inter-system loss of coolant accident.

Specification:

1. During reactor power operating conditions, the integrity of all pressure isolation valves listed in Table 3.3.1 shall be demonstrated. Valve leakage shall not exceed the amounts indicated.

2. If Specification 1 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

H. Required Minimum Recirculation Flow Rate for Operation in IRM Range 10

1. During STARTUP mode operation, a minimum recirculation flow rate is required before operating in IRM range 10 to ensure that technical specification transient MCPR limits for operation are not exceeded. This minimum flow rate is no longer required once the reactor is in the RUN mode.

2. 39.65×10^6 lb/hr is the minimum recirculation flow rate necessary for operation in IRM range 10 at this time. This flow rate leaves sufficient margin between the minimum flow required by the RWE analysis performed and the minimum flow used while operating in IRM range 10.

NRC Order Dated April 20, 1981

Section 3.3

Bases:

The reactor coolant system(1) is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The Oyster Creek reactor vessel was designed and manufactured in accordance with General Electric Specification 21A1105 and ASME Section I as discussed in Reference 13. The original operating limitations were based upon the requirement that the minimum temperature for pressurization be at least 60°F greater than the nil ductility transformation temperature. The minimum temperature for pressurization at any time in life has to account for the toughness properties in the most limiting regions of the reactor vessel, as well as the effects of fast neutron embrittlement.

Curves A, B and C on Figures 3.3.1, 3.3.2 and 3.3.3 are derived from an evaluation of the fracture toughness properties performed on the specimens contained in Reactor Vessel Materials Surveillance Program Capsule No. 2 (Reference 14). The results of dosimeter wire analyses (Reference 14) indicated that the neutron fluence (E>1.0 MeV) at the end of 32 effective full power years of operation is 2.36 x 10^{18} n/cm^2 at the 1/4T (T=vessel wall thickness) location. This value was used in the calculation of the adjusted reference nil-ductility temperature which, in turn, was used to generate the pressure-temperature curves A, B and C on Figures 3.3.1, 3.3.2 and 3.3.3 (Reference 15). The 250°F maximum pressure test temperature provides ample margin against violation of the minimum required temperature. Secondary containment is not jeopardized by a steam leak during pressure testing, and the Standby Gas Treatment system is adequate to prevent unfiltered release to the stack.

Stud tensioning is considered significant from the standpoint of brittle fracture only when the preload exceed approximately 1/3 of the final design value. No vessel or closure stud minimum temperature requirements are considered necessary for preload values below 1/3 of the design preload with the vessel depressurized since preloads below 1/3 of the design preload result in vessel closure and average bolt stresses which are less than 20% of the yield strengths of the vessel and bolting materials. Extensive service experience with these materials has confirmed that the probability of brittle fracture is extremely remote at these low stress levels, irrespective of the metal temperature.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surface adjacent to the "O" rings of the head and vessel flange. The original Code requirement was that boltup be done at qualification temperatures (T3OL) plus 60°F. Current Code requirements state (Ref. 16) that for application of full bolt preload and reactor pressure up to 20% of hydrostatic test pressure, the RPV metal temperature must be at RT_{NDT} or greater. The boltup temperature of 85°F was derived by determining the highest value of (T3OL + 60) and the highest value of RT_{NDT}, and by choosing the more conservative value of the two. Calculated values of (T3OL + 60) and RT_{NDT} of the RPV metal temperature were 85°F and 36°F, respectively (Ref. 15). Therefore, selecting the boltup temperature to be 85°F provides 49°F margin over the current Code requirement based on RT_{NDT}.

Detailed stress analyses(4) were made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these analyses are presented and compared to allowable stress limits in Reference (4). The specific conditions analyzed currently include 240 cycles (17) of normal startup and shutdown with a heating and cooling rate of 100°F per hour applied continuously over a temperature range of 100°F to 546°F and for 10 cycles of emergency cooldown at a rate of 300°F per hour applied over the same range. A review of the original analysis shows that the components with the highest fatigue usage factor are the reactor vessel studs and reactor vessel basin seal skirt. These components have the potential to exceed the allowable fatigue usage factor if the number of thermal cycles (i.e., heatup/cooldown) exceed design assumptions. The number of heatup and cooldown cycles was reanalyzed, as documented by Reference (17), for a higher number of cycles (240) than expected in the original analysis (120). The reanalysis confirmed that the original fatigue usage factor limit of 0.8 is maintained. All other components have relatively low usage factors and are not expected to exceed fatigue usage factor limit of 0.8 for the design life of 40 years. Thermal stresses from this analysis combined with the primary load stresses fall within ASME Code Section III allowable stress intensities. Although the Oyster Creek Unit 1 reactor vessel was built in accordance with Section I of the ASME Code, the design criteria included in the reactor vessel specifications were in essential agreement with the criteria subsequently incorporated into Section III of the Code.(6)

The expected number of normal heatup and cooldown cycles to which the vessel will be subjected is 80(7). Although no heatup or cooldown rates of 300°F per hour are expected over the life the vessel and the vessel design did not consider such events(6), stress analyses have been made which showed the allowable number of such events is 22,000 on the basis of ASME Section III alternating stress limits.

During reactor operation, the temperature of the coolant in an idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the sump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel as discussed above.

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite AC power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work (8) utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in the 3.3-D on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.3-D, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage of the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the plant should be shut down to allow further investigation and corrective action.

The drywell floor drain sump and equipment drain tank provide the primary means of leak detection(9,10). Identified leakage is that from valves and pumps in the reactor system and from the reactor vessel head flange gasket. Leakage through the seals of this equipment is piped to the drywell equipment drain tank. Leakage from other sources is classified as unidentified leakage and is collected in the drywell floor drain sump. Leakage which does not flash in a vapor will drain in the sump. The vapor will be condensed in the drywell ventilation system and routed to the sump.

Condensate cannot leave the sump or the drywell equipment drain tank unless the respective pumps are running. The sump and the drain tank are provided with two pumps each. Alarms are provided for the sump that will actuate on a predetermined pumpout rate(10) and will be set to actuate at a leakage that is less than the unidentified leakage limit of 5 gpm.

Additional qualitative information(10) is available to the operator via the monitored drywell atmospheric condition. However, this information is not quantitative since fluctuation in atmospheric conditions are normally expected, and quantitative measurements are not possible. The temperature of the closed cooling water which serves as coolant for the drywell ventilation system is monitored and also provides information which can be related to reactor coolant system leakage(9). Additional protection is provided by the drywell high pressure scram which would be expected to be reached within 30 minutes of a steam leak of about 12 gpm(10).

During a loss of offsite AC power, the control rod drive hydraulic pumps, which are powered by the diesels, each can supply 110 gpm water makeup to the reactor vessel. A 25 gpm limit for total leakage, identified and unidentified, was established to be less than the 110 gpm makeup of a single rod drive hydraulic pump to avoid the use of the emergency core cooling system in the event of a loss of normal AC power.

Materials in the primary system are primarily 304 stainless steel and zircaloy fuel cladding. The reactor water chemistry limits are placed upon conductivity and chloride concentration since conductivity is measured continuously and gives an indication of abnormal conditions or the presence of unusual materials in the coolant, while chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Chlorides are known to (1) promote intergranular stress corrosion cracking of sensitized steels, (2) induce transgranular cracking of non-sensitized stainless steels, (3) promote pitting and (4) promote crevice attack in most RCS materials (BWR Water Chemistry Guidelines, EPRI, April 1, 1984). The higher the concentration, the faster the attack. Therefore, the level of chloride in the reactor water should be kept as low as is practically achievable. The limits are therefore set to be consistent with Regulatory Guide 1.56 (Rev. 1).

In the case of BWR's where no additives are used in the primary coolant, and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. When the conductivity is within its proper normal range, pH, chloride, and other impurities affecting conductivity and water quality must also be within their normal ranges. Significant changes in conductivity provide the operator with a warning mechanism so that he can investigate and remedy the conditions causing the change.

Measurements of pH, chloride, and other chemical parameters are made to determine the cause of the unusual conductivity and instigate proper corrective action. These can be done before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Several techniques are available to correct off-standard reactor water quality conditions including removal of impurities from reactor water by the cleanup system, reducing input of impurities causing off-standard conditions by reducing power and reducing the reactor coolant temperature to less than 212°F. The major benefit of reducing the reactor coolant temperature to less than 212°F is to reduce the temperature dependent corrosion rates and thereby provide time for the cleanup system to re-establish proper water quality.

Specifications 3.3.F.1 and 3.3.F.2 provide the OPERABILITY requirements for recirculation loops including acceptable valve alignments for OPERATION with less than five OPERABLE loops.

The IDLE loop configuration allows back flow through the loop discharge bypass valve and the loop temperature can be maintained within 50 °F of the reactor coolant inlet temperature. An idle loop can be restarted since the restart of the loop will not result in a cold water addition transient causing a concern from either reactivity addition or reactor nozzle thermal stresses.

The ISOLATED RECIRCULATION LOOP will experience a cooling of the loop temperatures greater than 50 °F and restart of an isolated loop could result in a cold water addition transient. Therefore, restart of an ISOLATED loop is not permitted and the circuit breakers for the motor generator set are open and defeated from operation to prevent an inadvertent startup of an ISOLATED RECIRCULATION LOOP. The ISOLATED LOOP can only be returned to service when the reactor is in COLD SHUTDOWN. When a recirculation loop is ISOLATED, the coolant between the suction and discharge and discharge bypass valves is no longer available during a loss of coolant accident (LOCA). This lose of inventory requires a reduction to 98% of the MAPLHGR limits in the Core Operating Limits Report.

During three-loop operation reactor power is limited to 90% of rated power. This is a physical restriction, since it is unlikely that the plant could operate at 90% of rated power with three operating recirculation pumps; and it is the maximum power analyzed for three-loop operation. No more than one recirculation loop can be ISOLATED. This restriction is required since the loss of inventory from a second ISOLATED REICRCULATION LOOP has not been analyzed. Operation with two IDLE or one IDLE and one ISOLATED RECIRCULTION LOOPS is permissible.

A non-operating recirculation loop may not be configured with both the suction valve and discharge valve in the open position since the back flow through the loop would result in non-conservative instrument readings for recirculation flow. Therefore, the reactor would be shutdown according to Specification 3.3.F.3 if a recirculation loop cannot be placed into an IDLE or ISOLATED configuration.

Specifications 3.3.F.4 and 3.3.F.6 assure that an adequate flow path exists from the annular space, between the pressure vessel wall and the core shroud, to the core region. This provides sufficient hydraulic communication between these areas, thus assuring that reactor water instrument readings are indicative of the level in the core region. For the bounding loss of feedwater transient(2), a single fully open recirculation loop transfers coolant from the annulus to the core region at approximately five times the boiloff rate with no forced circulation(3). With the reactor vessel flooded to a level above 185 inches TAF or when the steam separator and dryer are removed, the core region and all recirculation loops can therefore be isolated. When the steam separator and dryer are removed, safety limit 2.1.D ensures water level is maintained above the core shroud.

References:

1. FDSAR, Volume I, Section IV-2

2. Letter to NRC dated May 19, 1979, "Transient of May 2, 1979"

3. General Electric Co. Letter G-EN-9-55, "Revised Natural Circulation Flow Calculation", dated May 29, 1979

4. Licensing Application Amendment 16, Design Requirements Section

5. (Deleted)
6. FDSAR, Volume I, Section IV-2.3.3 and Volume II, Appendix H

7. FDSAR, Volume I, Table IV-2-1

8. Licensing Application Amendment 34, Question 14

9. Licensing Application Amendment 28, Item III-B-2

10. Licensing Application Amendment 32, Question 15

11. (Deleted)

12. (Deleted)

13. Licensing Application Amendment 16, Page 1

14. GPUN TDR 725 Rev. 3: Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens

15. GENE-B13-01769 (GE Nuclear Energy): Pressure-Temperature Curves Per Regulatory Guide 1.99, Revision 2 for Oyster Creek Nuclear Generating Station.

16. Paragraph G-2222(C), Appendix G, Section XI, ASME Boiler and Pressure Vessel Code, 1989 Edition with 1989 Addenda, "Fracture Toughness Criteria for Protection Against Failure."

17. GPUN Safety Evaluation, SE-000221-004, "Reactor Vessel Thermal Cycles"

C. Minimum CRITICAL POWER RATIO (MCPR)

During steady state POWER OPERATION the minimum CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit as specified in the COLR.

The MCPR limit for each cycle as identified in the COLR shall be greater than or equal to 1.49.

When APRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, provided that the control rod block is placed in operation during this interval.

For core flows other than rated, the nominal value for MCPR shall be increased by a factor of k_f, where k_f is as shown in the COLR.

If at any time during POWER OPERATION it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the COLD SHUTDOWN CONDITION within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limit at which time POWER OPERATION may be continued.

Bases:

The Specification for AVERAGE PLANAR LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46. The analytical methods and assumptions used in evaluating the fuel design limits are presented in FSAR Chapter 4.

LOCA analyses are performed for each fuel design at selected exposure points to determine APLHGR limits that meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using GE calculational models which are consistent with the requirements of 10 CFR 50, Appendix K.

The PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. Since expected location variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46.

The maximum AVERAGE PLANAR LHGR limits for the various fuel types currently being used are provided in the COLR. The COLR includes MAPLHGR limits for five loop operation. Additional limits on MAPLHGR for operations with less than five loops are given in Specification 3.3.F.2.

Fuel design evaluations are performed to demonstrate that the cladding 1% plastic strain and other fuel design limits are not exceeded during anticipated operational occurrences for operation with LHGRs up to the operating limit LHGR.

The analytical methods and assumptions used in evaluating the anticipated operational occurrences to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6 and 15 and in Technical Specification 6.9.1.f. To assure that the Safety Limit MCPR is not exceeded during any moderate frequency transient event, limiting transients have been analyzed to determine the largest reduction in CRITICAL POWER RATIO (CPR). The types of transients evaluated are pressurization, positive reactivity insertion and coolant temperature decrease. The operational MCPR limit is selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state, manufacturing, and in the critical power correlation itself. This limit is derived by addition of the CPR for the most limiting transient to the safety limit MCPR designated in Specification 2.1.

A lower bound of 1.49 has been established for the operating limit MCPR value to provide sufficient margin to the MCPR safety limit in the event of reactor thermal-hydraulic instability. The 1.49 limit will be considered against the minimum operating CPR limit based on reload transient and accident analysis. The higher of core stability or reactor transient and accident determined MCPR will be used to determine the cycle operating limit.

The APRM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM status as a consequence of bypassed APRM channel and/or failed/bypassed LPRM inputs. The steady state MCPR required to protect the minimum transient CPR for the worst case APRM status condition (APRM Status 1) is determined in the rod withdrawal error transient analysis. The steady state MCPR values for APRM status conditions 1, 2, and 3 will be evaluated each cycle. For those cycles where the rod withdrawal error transient is not the most severe transient the MCPR value for APRM status conditions 1, 2, and 3 will be the same and be equal to the limiting transient MCPR value.

The time interval of Eight (8) hours to adjust the steady state of MCPR to account for a degradation in the APRM status is justified on the basis of instituting a control rod block which precludes the possibility of experiencing a rod withdrawal error transient since rod withdrawal is physically prevented. This time interval is adequate to allow the operator to either increase the MCPR to the appropriate value or to upgrade the status of the APRM system while in a condition which prevents the possibility of this transient occurring.

Transients analyzed each fuel cycle will be evaluated with respect to the operational MCPR limit specified in the COLR.

The purpose of the k_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the k_f factor. Specifically, the k_f factor provides the required thermal margin to protect against a flow increase transient.

The k_f factor curves, as shown in the COLR, were developed generically using the flow control line corresponding to rated thermal power at rated core flow. For the manual flow control mode, the k_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the value of k_f.