3.5.2 CONTROL ROD INSERTION LIMITS

APPLICABILITY: MODES 1 and 2

OBJECTIVE: This specification defines the insertion limits for the control rods in order to ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

SPECIFICATION:

A. Except during low power physics tests or surveillance testing pursuant to Specification 4.1.1.G, the Shutdown Groups and Control Group 1 shall be fully withdrawn, and the position of Control Group 2 shall be at or above the 21-step uncertainty limit shown in Figure 3.5.2.1.

B. The energy weighted average of the positions of Control Group 2 shall be at least 90% (i.e. > Step 288) withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all Control Group 2 positions during the core cycle.

ACTION:

A. With the control groups inserted beyond the above insertion limits either:

1. Restore the control groups to within the limits within 2 hours, or
2. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or
3. Be in at least HOT STANDBY within 6 hours.

B. With a single dropped rod from a shutdown group or control group, the provisions of Action A are not applicable, and retrieval shall be performed without increasing THERMAL POWER beyond the THERMAL POWER level prior to dropping the rod. An evaluation of the effect of the dropped rod shall be made to establish permissible THERMAL POWER levels for continued operation. If retrieval is not successful within 3 hours from the time the rod was dropped, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% $\Delta k/k$.

BASIS: During STARTUP and POWER OPERATION, the shutdown groups and control group 1 are fully withdrawn and control of the reactor is maintained by control group 2. The control group insertion limits are set in consideration of maximum specific shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta}H$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design value of specific power, $F^N_{\Delta}H$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta}H$ and F_Q are 13.2 kW/ft, 1.57 and 2.78, respectively.(8) At partial power, the $F^N_{\Delta}H$ maximum values (limits) increase according to the following equation, $F^N_{\Delta}H\ (P) = 1.57\ [1 + 0.2\ (1-P)]$, where P is the fraction of RATED THERMAL POWER. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

2. The minimum shutdown capability required is 1.25% $\Delta\rho$ at BOL, 1.9% $\Delta\rho$ at EOL and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available SHUTDOWN MARGIN is greater than the above values.

3. The worst case ejected rod accident (9) covering HFP-BOL, HZP-BOL, HFP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 220 cal/gm for irradiated fuel.

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

   Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% RATED THERMAL POWER. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed.

   It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available SHUTDOWN MARGIN.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

The presence of a dropped rod leads to abnormal power distribution in the core. The location of the rod and its worth in reactivity determines its effect on the temperatures of nearby fuel. Under certain conditions, continued operation could result in fuel damage, and it is the intent of ACTION 8 to avoid such damage.

References:

(1) Final Engineering Report and Safety Analysis, revised July 28, 1970.

(2) Amendment No. 18 to Docket No. 50-206.

(3) Amendment No. 22 to Docket No. 50-206.

(4) Amendment No. 23 to Docket No. 90-206.

(5) Description and Safety Analysis, Proposed Change No. 7, dated October 22, 1971.

(6) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 4, WCAP 8131, May, 1973.

(7) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

(8) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989.

(9) An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January, 1975.

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

APPLICABILITY: MODE 1

OBJECTIVE: To provide corrective action in the event that the axial offset monitoring system limits are approached.

SPECIFICATION: The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $$IAO = \frac{2.78/P - 2.10}{0.033} - FCC$$

For negative offsets: $$IAO = \frac{2.78/P - 2.10}{-0.033} + FCC$$

where

IAO = incore axial offset

P = fraction of RATED THERMAL POWER

FCC = The larger of 3.0 or the value in percent of incore axial offset by which the current correlation check differs from the incore-excore correlation.

ACTION:

A. With IAO exceeding the limit defined by the specification, within 1 hour action shall be taken to reduce THERMAL POWER until IAO is within specified limits.

B. With one or both excore axial offset channel(s) inoperable, as an alternate, one OPERABLE NIS channel for each inoperable excore axial offset channel, shall be logged every two hours to determine IAO.

C. With no method for determining IAO available, be in MODE 2 within 6 hours.

BASIS: The percent full power axial offset limits are conservatively established considering the core design peaking factor, analytical determination of the relationship between core peaking factors and incore axial offset considering a wide range of maneuvers and core conditions, and actual measurements relating IAO to the axial offset monitoring systems(1). The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed IAO limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and IAO has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION.

Reducing power until IAO is within the specified limits in cases when limits are exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to place the plant in MODE 2 within 6 hours. However, if axial offset channel(s) are inoperable, hand calculational methods of determining IAO from OPERABLE NIS channels can be employed until OPERABILITY of the axial offset channel(s) is restored.

References:

(1) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989

(2) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973

(3) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974

(4) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1 Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

ATTACHMENT 2

3.5.2 CONTROL ROD INSERTION LIMITS

APPLICABILITY: MODES 1 and 2

OBJECTIVE: This specification defines the insertion limits for the control rods in order to ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

SPECIFICATION:

A. Except during low power physics tests or surveillance testing pursuant to Specification 4.1.1.G, the Shutdown Groups and Control Group 1 shall be fully withdrawn, and the position of Control Group 2 shall be at or above the 21-step uncertainty limit shown in Figure 3.5.2.1.

B. The energy weighted average of the positions of Control Group 2 shall be at least 90% (i.e. > Step 288) withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all Control Group 2 positions during the core cycle.

ACTION:

A. With the control groups inserted beyond the above insertion limits either:

1. Restore the control groups to within the limits within 2 hours, or

2. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

3. Be in at least HOT STANDBY within 6 hours.

B. With a single dropped rod from a shutdown group or control group, the provisions of Action A are not applicable, and retrieval shall be performed without increasing THERMAL POWER beyond the THERMAL POWER level prior to dropping the rod. An evaluation of the effect of the dropped rod shall be made to establish permissible THERMAL POWER levels for continued operation. If retrieval is not successful within 3 hours from the time the rod was dropped, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% Δ k/k.

BASIS: During STARTUP and POWER OPERATION, the shutdown groups and control group 1 are fully withdrawn and control of the reactor is maintained by control group 2. The control group insertion limits are set in consideration of maximum specific shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 11.3 kW/ft, 1.57 and 2.38, respectively. At partial power, the $F^N_{\Delta H}$ maximum values (limits) increase according to the following equation, $F^N_{\Delta H}(P) = 1.57\ [1 + 0.2\ (1-P)]$, where P is the fraction of RATED THERMAL POWER. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

2. The minimum shutdown capability required is 1.25% $\Delta\rho$ at BOL, 1.9% $\Delta\rho$ at EOL and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available SHUTDOWN MARGIN is greater than the above values.

3. The worst case ejected rod accident (8) covering HFP-BOL, HZP-BOL, HFP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 220 cal/gm for irradiated fuel.

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% RATED THERMAL POWER. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed.

It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available SHUTDOWN MARGIN.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

The presence of a dropped rod leads to abnormal power distribution in the core. The location of the rod and its worth in reactivity determines its effect on the temperatures of nearby fuel. Under certain conditions, continued operation could result in fuel damage, and it is the intent of ACTION B to avoid such damage.

<u>References</u>:

(1) Final Engineering Report and Safety Analysis, revised July 28, 1970.

(2) Amendment No. 18 to Docket No. 50-206.

(3) Amendment No. 22 to Docket No. 50-206.

(4) Amendment No. 23 to Docket No. 90-206.

(5) Description and Safety Analysis, Proposed Change No. 7, dated October 22, 1971.

(6) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 4, WCAP 8131, May, 1973.

(7) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

(8) An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January, 1975.

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

APPLICABILITY: MODE 1

OBJECTIVE: To provide corrective action in the event that the axial offset monitoring system limits are approached.

SPECIFICATION: The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $$IAO = \frac{2.595/P - 2.10}{0.033} - FCC$$

For negative offsets: $$IAO = \frac{2.595/P - 2.10}{-0.033} + FCC$$

where

IAO = incore axial offset

P = fraction of RATED THERMAL POWER

FCC = The larger of 3.0 or the value in percent of incore axial offset by which the current correlation check differs from the incore-excore correlation.

ACTION:

A. With IAO exceeding the limits defined by the specification, within 1 hour, action shall be taken to restore IAO within specified limits, or to reduce THERMAL POWER until IAO is within specified limits.

B. With one or both excore axial offset channel(s) inoperable, as an alternate, one OPERABLE NIS channel for each inoperable excore axial offset channel, shall be logged every two hours to determine IAO.

C. With no method for determining IAO available, be in MODE 2 within 6 hours.

BASIS: The percent full power axial offset limits are conservatively established considering the core design heat flux hot channel factor (F_Q), the relationship between F_Q and incore axial offset considering a wide range of maneuvers and core conditions, and actual measurements relating IAO to the axial offset monitoring systems.(1) The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed IAO limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the relationship between F_Q and IAO has changed in a manner warranting modification to the existing envelope (1,2), then a change to the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION.

Reducing power until IAO is within the specified limits in cases when limits are exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to place the plant in MODE 2 within 6 hours. However, if axial offset channel(s) are inoperable, hand calculational methods of determining IAO from OPERABLE NIS channels can be employed until OPERABILITY of the axial offset channel(s) is restored.

References:

(1) Letter, Westinghouse (S. A. Pujadas) to SCE (Bernie Carlisle), "Integrated Assessment for Refill Volume Discrepancy," April 24, 1991.

(2) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973

(3) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974

(4) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1 Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

ATTACHMENT 3

SCE-91-533

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

April 24, 1991
NS-OPLS-OPL-I-91-197

Mr. Bernie Carlisle, Supervisor
Nuclear Engineering
Southern California Edison Company
23 Parker Street
Irvine, CA 92718

SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1
INTEGRATED ASSESSMENT FOR REFILL VOLUME DISCREPANCY

Dear Mr. Carlisle:

Westinghouse has completed an integrated assessment of the remaining safety analysis disciplines within our scope in order to determine the overall effect of the refill volume discrepancy. This issue had previously been addressed specifically for large break LOCA in SCE-91-528, with Technical Specification changes defined in SCE-91-530. The results of the large break LOCA assessment are repeated here with additional information regarding the remaining safety analysis disciplines.

If you have any questions or comments, please contact the undersigned.

Very truly yours,

San Bungo /FOR

S. A. Pujadas, Manager
Western Area
U.S. Nuclear Projects I

S. DiTommaso
S. DiTommaso/

cc: J. T. Reilly 1L, 1A
W. Flournoy 1L, 1A
T. Yackle 1L, 1A
A. J. Eckhart 1L, 1A

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1 (SONGS-1)

INTEGRATED SAFETY ASSESSMENT FOR REACTOR VESSEL LOWER PLENUM VOLUME DISCREPANCY

BACKGROUND

On March 26, 1991, an inconsistency was discovered between the SONGS-1 refill volume assumed in the Westinghouse IAC (Interim Acceptance Criteria) large break LOCA analysis and the recent small break LOCA NOTRUMP analysis. As part of the NOTRUMP small break LOCA analysis performed to support increases in safety injection (SI) miniflow, SONGS-1 plant data was collected for the Westinghouse plant component data base (IMP). This data base automatically generates the small break analysis noding model. Generation of the NOTRUMP deck provided a lower plenum volume of 611 ft^3 to the bottom of the active fuel inside the core barrel. An additional 61 ft^3 represents the downcomer volume from the bottom of the core barrel to the bottom of the active fuel elevation. Together, this 672 ft^3 volume comprises what can be termed "refill volume" with respect to the IAC large break evaluation model (EM).

Review of the SONGS-1 vessel diagrams indicates the refill volume used in the large break LOCA analysis is underestimated using the current analysis practices. This results in an increase in calculated Peak Cladding Temperature (PCT). The following discussion provides an assessment of the effect of this refill volume discrepancy on the large break LOCA analysis as well as the remaining safety analyses performed by Westinghouse for SONGS-1.

SAFETY ANALYSIS ASSESSMENTS

Large Break LOCA Assessment

The current licensing basis large break LOCA analysis assumes a refill volume of 490 ft^3, which is underestimated by 182 ft^3 using the current input calculation methods. This value is used in the large break analysis to calculate the time to bottom of core (BOC) recovery following the RCS blowdown. From the end of blowdown (EOB) to BOC, an adiabatic clad heat up rate is assumed. Once BOC is reached, the clad temperature transient soon reverses and clad temperatures begin to decrease.

For SONGS-1, this adiabatic heat up rate is 12.1°F. The current SI flow rate assumed in the latest evaluation for increased SI miniflow is 655 lbm/sec. At an assumed density of 62.3 lb/ft^3, the delay in BOC is:

ΔT_{BOC} = (182 ft^3)(62.3 lb/ft^3)/(655 lbm/sec) = 17.31 sec

At a 12.1°F/sec adiabatic heat up rate, the PCT penalty is:

ΔPCT = (17.31 sec)(12.1°F/sec) = 209.5°F

The current large break peak clad temperature has been evaluated at 2278.5°F for increased miniflow. Therefore, to maintain compliance with the 2300°F IAC PCT limit, at least 188.0°F of margin is needed to accommodate the larger refill volume.

Reduced Axial Offset (Peaking Factors)

SONGS-1 is analyzed at a total peaking factor (F_Q) of 2.78 (13.2 kw/ft). Reanalysis of the current cycle has demonstrated that reducing axial offset limits results in an overall peaking factor of only 2.38 (11.3 kw/ft). Past sensitivities with the IAC model have shown a PCT sensitivity of 50°F for 0.4 kw/ft. Based on this sensitivity, the expected PCT benefit from this change is:

ΔPCT_{FQ} = [(11.3 - 13.2)kw/ft](50°F/0.4 kw/ft)
= -237.5°F

which compensates for the penalty associated with the increased refill volume.

92% Power Operation

Further margin is available in the large break analysis by crediting power level. SONGS-1 is physically limited to 92% power operation. The current analyses assume 100% power. Power sensitivities with the IAC large break EM have shown PCT benefits of 8°F for a 1% decrease in power level. Therefore:

ΔPCT_{power} = (92% - 100%)(8°F/1%) = -64°F

Reduced Safety Injection Miniflow

As previously stated, a large break LOCA evaluation has been performed to allow operator action to increase miniflow prior to 30 minutes following a LOCA. This is a small break LOCA concern since the large break LOCA transient is terminated within several minutes of break initiation. The assumption of lower miniflow increases SI from 655 lb/sec to 682 lb/sec. The estimated PCT at this higher flow rate is 2232.5°F, such that:

$\Delta PCT_{miniflow}$ = (2232.5 - 2278.5)°F = -46°F

This could be used to provide additional margin within the SONGS-1 licensing basis.

In conclusion, increasing the $490ft^3$ refill volume assumed in the large break analysis to a value of $672ft^3$ consistent with the current methods, results in a 209.5°F PCT penalty. The following is a portion of the margin available to offset this penalty and maintain compliance with the 2300°F IAC PCT limit:

ΔPCT_{FQ}	= -237.5°F
ΔPCT_{power}	= -64.0°F
$\Delta PCT_{miniflow}$	= -46.0°F
TOTAL AVAILABLE MARGIN	= 347.5°F

Safe operation of SONGS-1 at full Rated Thermal Power (RTP) can be assured by adjusting inward the axial offset operating band width.

Post-LOCA Long Term Subcriticality

The large break LOCA does not take credit for RCCAs to ensure reactor shutdown and core subcriticality. The borated water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods are out. The larger refill volume is conservatively assumed to consist of unborated water. Adding $182ft^3$ of unborated water to the previous mixed mean boron concentration calculation results in a reduction of less than 50 ppm in the sump boron concentration. However, it has been determined that for the current cycle, in excess of 1000 ppm margin exists to the critical boron concentration. Therefore, a reduction in the sump boron concentration of less than 50 ppm will not violate the above criteria. Note that post-LOCA boron subcriticality is confirmed for each reload.

Non-LOCA Assessment

The lower plenum volume is modeled in LOFTRAN analyses as part of the LOFTRAN Reactor Vessel Inlet Volume. The LOFTRAN Reactor Vessel Inlet Volume is made up of the following four volumes:

1) Volume of the downcomer between the bottom of the nozzles and the bottom of the fuel.
2) Volume of the lower plenum below the bottom of the fuel.
3) Volume between the barrel and the baffle from the bottom of the fuel to upper core plate.
4) Volume of the downcomer between the bottom and top of the inlet nozzles.

Comparing the latest data available from the IMP data base (creation date 1/11/91), with a sampling of the non-LOCA analyses performed for SONGS-1, it was determined that the reactor vessel inlet plenum used in the non-LOCA analyses was essentially the same as the value from the IMP data base. LOFTRAN results are insensitive to small changes in the assumed Reactor Vessel Inlet Volume and thus the small difference between the IMP value and the value used in the non-LOCA analyses would not significantly affect the current SONGS-1 non-LOCA analyses.

The reduction in the limits for F_Q and axial offset has been reviewed for non-LOCA analyses. The use of these more restrictive values is conservative in all situations. Thus, the existing analyses remain valid.

In conclusion, the lower plenum volume has an insignificant impact on the non-LOCA safety analyses. The F_Q and Axial Offset limits changes are bounded by the assumptions in the existing non-LOCA safety analyses. As such, the existing non-LOCA safety analyses remain valid.

Containment Integrity Assessment

The refill volume discrepancy has been evaluated relative to the current licensing basis LOCA mass and energy release calculations. The RCS volumetric differences between the current licensing basis LOCA mass and energy release model and the small break LOCA NOTRUMP analysis, which is based upon the latest RCS geometric data, were compared. The evaluation was performed with realistic, but still conservative geometric data, and assumptions consistent with the current plant operating conditions. The results of the evaluation demonstrate that the current licensing basis LOCA mass and energy release analysis results remain bounding. All applicable safety criteria for the containment mass and energy release analysis continue to be met.

Technical Specification Assessment

Westinghouse has confirmed that the maximum total peaking factor (Fq*P) for SONGS-1 cycle 11 will be maintained below 2.38 provided the axial offset limit equations in Technical Specification 3.11 are changed to the following:

For Positive Offsets: $$IAO = \frac{2.595/P - 2.10}{0.033} - FCC$$

For Negative Offsets: $$IAO = \frac{2.595/P - 2.10}{-0.033} + FCC$$

The above equations result in an allowable axial offset window of +/- 15% at 100% full power, which is a 5.6% reduction on both sides of the window. The incore/excore correlation uncertainty (FCC) is not included in the calculation of the +/- 15%, such that the actual axial offset allowed at the plant will be more on the order of +/- 12% at 100% power, or +/- 19% at 92% power.

The values for F_Q and specific power in the Bases to Technical Specification 3.5.2 must also be revised. The F_Q value is reduced to 2.38 based upon a change in specific power to 11.3 kw/ft. The rod insertion limits defined in Technical Specification 3.5.2 remain applicable.

Implementation of these Technical Specification changes will assure that the PCT limit of the IAC large break LOCA analysis is met. In addition, the peak rod F-delta-H, peak to average power in the hot assembly and the fuel rod census will not change. Therefore, the DNBR design bases and fuel temperature limits associated with local power peaking will continue to be met. Based on this information, no other analyses or design conditions are affected by the refill volume discrepancy and no other Technical Specifications changes are required.

Radiological Consequences

The radiological consequences of a large break LOCA are determined based upon a conservative source term assumption which is independent of core response and does not model primary to secondary flow. For transients other than large break LOCA, a primary to secondary leakage rate is conservatively assumed to be equivalent to the allowable Technical Specification value. The vessel refill volume does not have an adverse effect on these assumptions and therefore the dose consequences will not be increased.

Disciplines Not Affected by Large Break LOCA Refill Volume

Included in the Westinghouse scope for reviewing safety issues are disciplines for mechanical component integrity, fluid systems performance, electrical instrumentation and control systems and Emergency Operating Procedure setpoints. These disciplines have been reviewed and it has been determined that they are not adversely affected by the underestimated refill volume.

The steam generator tube rupture analysis was not performed by Westinghouse. For a typical steam generator tube rupture transient, results would not be sensitive to the volume of the lower plenum (refill volume).