RADIOLOGICAL TECHNICAL SPECIFICATIONS

APPENDIX A

TO

FACILITY OPERATING LICENSE NO. DPR-63

FOR THE

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT 1

DOCKET NO. 50-220

DECEMBER 26, 1974

The abnormal operational transients applicable to operation of the plant have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 1850 MWt. The analyses were based upon plant operation in accordance with the operating map given in Reference 11. In addition, 1850 MWt is the licensed maximum power level, and represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 2.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

a. The Average Power Range Monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses [5, 6, 8, 9, 10, 11, 13] demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

However, in response to expressed beliefs[7] that variation of APRM flux scram with recirculation flow is a prudent measure to assure safe plant operation during the design confirmation phase of plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Figure 2.1.1 when the maximum total peaking factor is greater than the limiting total peaking factor.

b. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. For operation in the startup mode while the reactor is at low pressure, the IRM scram setting is 12% of rated neutron flux. Although the operator will set the IRM scram trip at 12% of rated neutron flux or less, the actual scram setting can be as much as 2.5% of rated neutron flux greater. This includes the margins discussed above. This provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power flow condition. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit.

Procedural controls will assure that the IRM scram is maintained up to 20% flow. This is accomplished by keeping the reactor mode switch in the startup position until 20% flow is exceeded and the APRM's are on scale. Then the reactor mode switch may be switched to the run mode, thereby switching scram protection from the IRM to the APRM system.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining a limit above the SLCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

c: As demonstrated in Appendix E-I* and the Technical Supplement to Petition to Increase Power Level, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure scram, generator load rejection scram, and main steam isolation valve closure scram, for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

d. A reactor water low level scram trip setting -12 inches (53 inches indicator scale) relative to the minimum normal water level (Elevation 302'9") will assure that power production will be terminated with adequate coolant remaining in the core. The analysis of the feedwater pump loss in the Technical Supplement to Petition to Increase Power Level, dated April 1970, has demonstrated that approximately 4 feet of water remains above the core following the low level scram.

The operator will set the low level trip setting no lower than -12 inches relative to the lowest normal operating level. However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

e. A reactor water low-low level signal -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9") will assure that core cooling will continue even if level is dropping. Core spray cooling will adequately cool the core, as discussed in LCO 3.1.4.

The operator will set the low-low level core spray initiation point at no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9"). However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

f. Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the SLCPR. This rod block trip setting, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 110% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the design peaking factor, thus, preserving the APRM rod block safety margin.

g-h. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line isolation on reactor low pressure and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at ≤10% valve closure, there is no increase in neutron flux and peak pressure if the vessel dome is limited to 1141 psig. [8, 9, 10].

The operator will set the pressure trip at greater than or equal to 850 psig and the isolation valve stem position scram setting at less than or equal to 10% of valve stem position from full open. However, the actual pressure set point can be as much as 15.8 psi lower than the indicated 850 psig and the valve position set point can be as much as 2.5% of stem position greater. These allowable deviations are due to instrument error, operator setting error and drift with time.

In addition to the above mentioned Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as a secondary backup to the Limiting Safety System Setting chosen. These are as follows:

High fission product activity released from the core is sensed in the main steam lines by the high radiation main steam line monitors. These monitors provide a backup scram signal and also close the main steam line isolation valves.

The scram dump volume high level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharged from the control rod drive hydraulic system as a result of a reactor scram (Section X-C.2.10)*.

i. The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to the worst case transient of a load rejection and subsequent failure of the bypass. In fact, analysis [9,10] shows that heat flux does not increase from its initial value at all because of the fast action of the load rejection scram; thus, no significant change in MCPR occurs.

j. The turbine stop valve closure scram is provided for the same reasons as discussed in i above. With a scram setting of ≤10% valve closure, the resultant transients are nearly the same as for those described in i above; and, thus, adequate margin exists.

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a. The range of set points for a safety valve actuation is selected in accordance with code requirements. A safety valve capability study presented in the Technical Supplement to Petition to Increase Power Level using the stated Limiting Safety System Setting values has demonstrated the maximum pressures occurring at the bottom of the reactor vessel and the bottom of the recirculation piping are 1303 psig and 1315 psig, respectively, some 72 psig below the 1375 psig safety limit. This analysis has assumed the highly improbable event of reactor isolation occurring without scram, in spite of separate and redundant scram signals such that the power output reached 167 percent of rated (1850 Mwt).

In addition to the safety valves, the solenoid-actuated relief valves are used to prevent safety valve lift during rapid reactor isolation at power coupled with failure of the bypass system. Any five of these valves opening at 1090 psig to 1100 psig will keep the maximum vessel pressure below the lowest safety valve setting, as demonstrated in Appendix E-I.3.11 (p. E-35)*. (The Technical Supplement to Petition to Increase Power Level, and letter from T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972). Subsequently, six valves were provided due to the blowdown requirements, following a small line break. The capacity of a solenoid-actuated relief valve is about the same as a safety valve. Therefore, even without scram any combination of 16 safety valves and solenoid-actuated valves will limit the pressure below the safety limit following the worst isolation situation.

b. The reactor high pressure scram setting is relied upon to terminate rapid pressure transients if other scrams, which would normally occur first, fail to function. As demonstrated in Appendix E-I of the FSAR and the Technical Supplement to Petition to Increase Power Level, Page II-12, the reactor high pressure scram is a backup to the neutron flux scram, generator load rejection scram, and main steam isolation-valve closure scram for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine-generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

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c. As shown in Appendix E-I.3.8 and 3.11*, rapid Station transients due to isolation valve or turbine trip valve closures result in coincident high-flux and high-pressure transients. Therefore, the APRM trip, although primarily intended for core protection, also serves as backup protection for pressure transients.

Although the operator will set the scram setting at less than or equal to that shown in Figure 2.1.1, the actual neutron flux setting can be as much as 2.7 percent of rated neutron flux above the line. This includes the errors discussed above. The flow bias could vary as much as one percent of rated recirculation flow above or below the indicated point.

In addition to the above-mentioned Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as secondary backup to the Limiting Safety System Setting chosen. These are as follows:

The primary containment high-pressure scram serves as backup to high reactor pressure scram in the event of lifting of the safety valves. As discussed in Vol. I, VIII, 2.0.c (p. VIII-9)* a pressure in excess of 3.5 psig due to steam leakage or blowdown to the drywell will trip a scram well before the core is uncovered.

The scram dump volume high-level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharge from the control-rod-drive hydraulic system as a result of a reactor scram (Section X-C.2.10)*.

In the event of main-steam-line isolation valve closure, reactor pressure will increase. A reactor scram is, therefore, provided on main-steam-line isolation valve position and anticipates the high reactor pressure scram trip.

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3.1.0 FUEL CLADDING

A) GENERAL APPLICABILITY

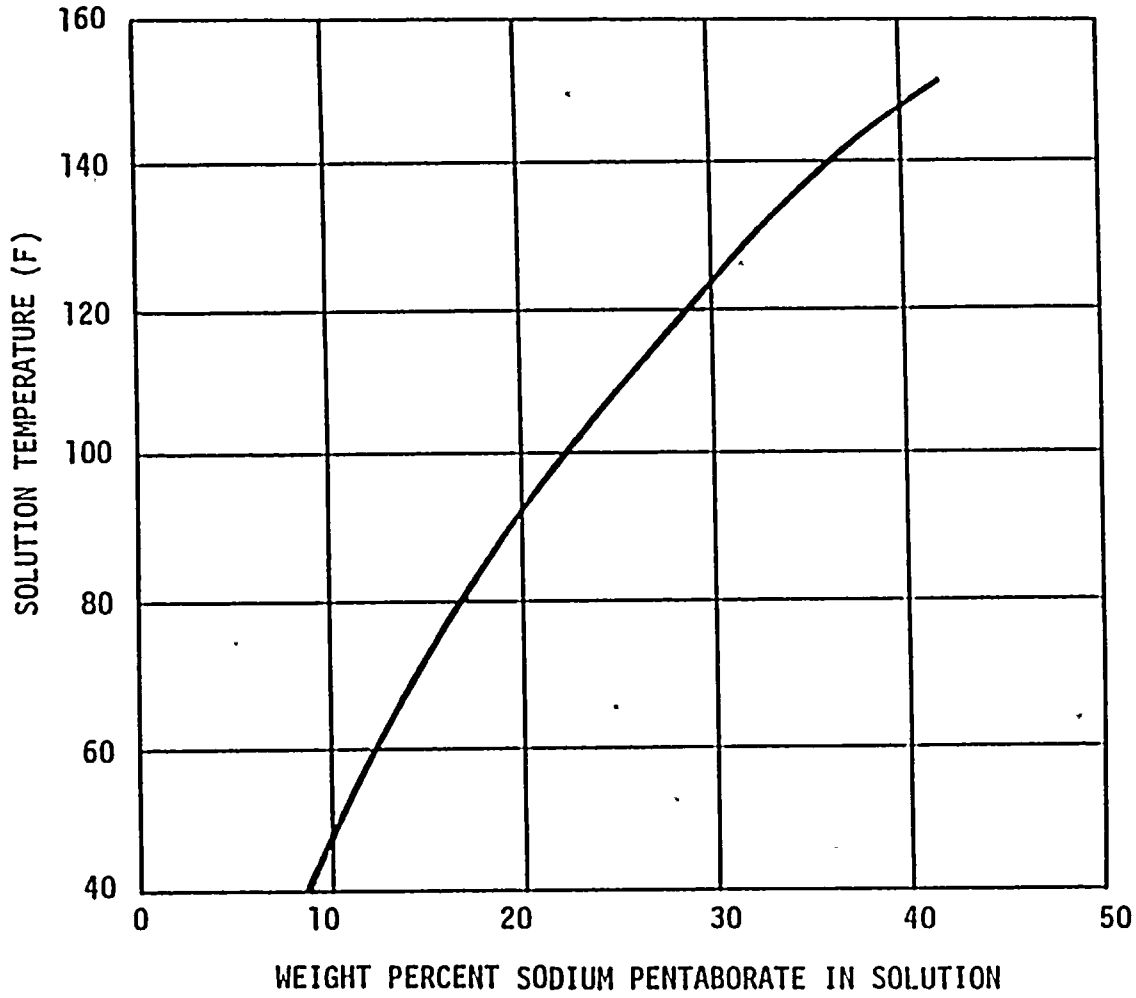
Applies to the power level regulation, control rod system, liquid poison system, emergency cooling system, and core spray system. LCOs for the minimum allowable circuits corresponding to the Limiting Safety System Setting are included in the Reactor Protection System LCO (3.6.2).

B) GENERAL OBJECTIVE

LIMITING CONDITIONS FOR OPERATION - To define the lowest functional capability or performance level of the systems and associated components which will assure the integrity of the fuel cladding as a barrier against the release of radioactivity.

SURVEILLANCE REQUIREMENTS - To define the tests or inspections required to assure the functional capability or performance level of the required systems or components.

Figure 3.1.2b
MINIMUM ALLOWABLE SOLUTION TEMPERATURE



Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature and the peak local cladding oxidation following the postulated design basis loss-of-coolant accident will not exceed the limits specified in 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local 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 within the 10CFR50, Appendix K limit. The limiting value for APLHGR is provided in the Core Operating Limits Report. The APLHGR curves in the Core Operating Limits Report are based on calculations using the models described in References 13, 15 and 16.

The Reference 13 and 15 LOCA analyses are sensitive to minimum critical power ratio (MCPR). In the Reference 15 analysis, an MCPR value of 1.30 was assumed. If future transient analyses should yield a MCPR limit below this value, the Reference 15 LOCA analysis MCPR value would become limiting. The current MCPR limit is provided in the Core Operating Limits Report. For fuel bundles analyzed with the Reference 13 LOCA methodology, assume MCPR values of 1.30 and 1.36 for five recirculation loop and less than five loop operation respectively.

Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated (Reference 12). The LHGR shall be checked daily during reactor operation at ≥25% power to determine if fuel burnup or control rod movement has caused changes in power distribution.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at a minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal-hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial startup testing of the plant, an MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

MCPR limits during operation at other than rated conditions are provided in the Core Operating Limits Report. For the case of automatic flow control, the K_f factor is determined such that any automatic increase in power (due to flow control) will always result in arriving at the nominal required MCPR at 100% power. For manual flow control, the K_f is determined such that an inadvertent increase in core flow (i.e., operator error or recirculation pump speed controller failure) would result in arriving at the 99.9% limit MCPR when core flow reaches the maximum possible core flow corresponding to a particular setting of the recirculation pump MG set scoop tube maximum speed control limiting set screws. These screws are to be calibrated and set to a particular value and whenever the plant is operating in manual flow control, the K_f defined by that setting of the screws is to be used in the determination of required MCPR. This will assure that the reduction in MCPR associated with an inadvertent flow increase always satisfies the 99.9% requirement. Irrespective of the scoop tube setting, the required MCPR is never allowed to be less than the nominal MCPR (i.e., K_f is never less than unity).

Power/Flow Relationship

The power/flow curve is the locus of critical power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis [7, 8, 12, 14] justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

During reactor startup with periods of low reactor water feed demand, one feedwater train is operated with a blocking valve closed downstream of the main flow control valve when core power is less than or equal to 25% of rated thermal power. This allows the low flow control valve to control the reactor water flow during the startup period when feedwater flow demand is low. Use of the low flow control valve provides more uniform feedwater flow which reduces thermal cycling at the reactor pressure vessel feedwater nozzles and in the feedwater piping as well as eliminating a severe service condition in the main flow control valves during reactor startup. Under low feedwater flow conditions, the main flow control valves also experience high pressure drops and fluid velocities which shorten the valve's life and can cause plant transients due to control valve failure. Reactor startup with one HPCI train available is acceptable since LOCA makeup requirements are reduced during startup because of lower reactor pressure, less decay heat, and lower reactor power than assumed in LOCA analyses performed to Appendix K 10 CFR 50 requirements. The other feedwater train (other HPCI loop) with its blocking valve open would remain capable of supplying 3,420 gpm of feedwater upon automatic HPCI initiation at all reactor pressure.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be 35 psig which would rapidly reduce to 22 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 22 psig is calculated to be about 10 seconds. Following the pipe break, the suppression chamber pressure rises to 22 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay.[1]

The design pressures of the drywell and suppression chamber are 62 psig and 35 psig, respectively.[2] As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated primary containment pressure response discussed above and the suppression chamber design pressure; primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.9%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95 percent for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 6.0 rem and the maximum total thyroid dose is about 150 rem at the site boundary considering fumigation conditions over an exposure duration of two hours. The resultant doses would occur for the duration of the accident at the low population distance of 4 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency (Specification 4.4.4) are conservative and provide margin between expected offsite doses and 10CFR100 guideline limits.

The maximum allowable leakage rate (L_a) is 1.5%/day at a pressure of 35 psig (P_a). This value for the test condition was derived from the maximum allowable accident leak rate of about 1.9%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air depleted of oxygen whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing.[3]

LIMITING CONDITION FOR OPERATION

3.4.1 LEAKAGE RATE

Applicability:

Applies to the leakage rate of the secondary containment.

Objective:

To specify the requirements necessary to limit exfiltration of fission products released to the secondary containment as a result of an accident.

Specification:

Whenever the reactor is in the refueling or power operating condition, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 2000 cfm. If this cannot be met after a routine surveillance check, then the actions listed below shall be taken:

a. Suspend immediately irradiated fuel handling, fuel pool and reactor cavity activities, and irradiated fuel cask handling operations in the reactor building.

b. Restore the reactor building leakage rates to within specified limits within 4 hours or initiate normal orderly shutdown and be in a cold shutdown condition within 10 hours.

SURVEILLANCE REQUIREMENT

4.4.1 LEAKAGE RATE

Applicability:

Applies to the periodic testing requirements of the secondary containment leakage rate.

Objective:

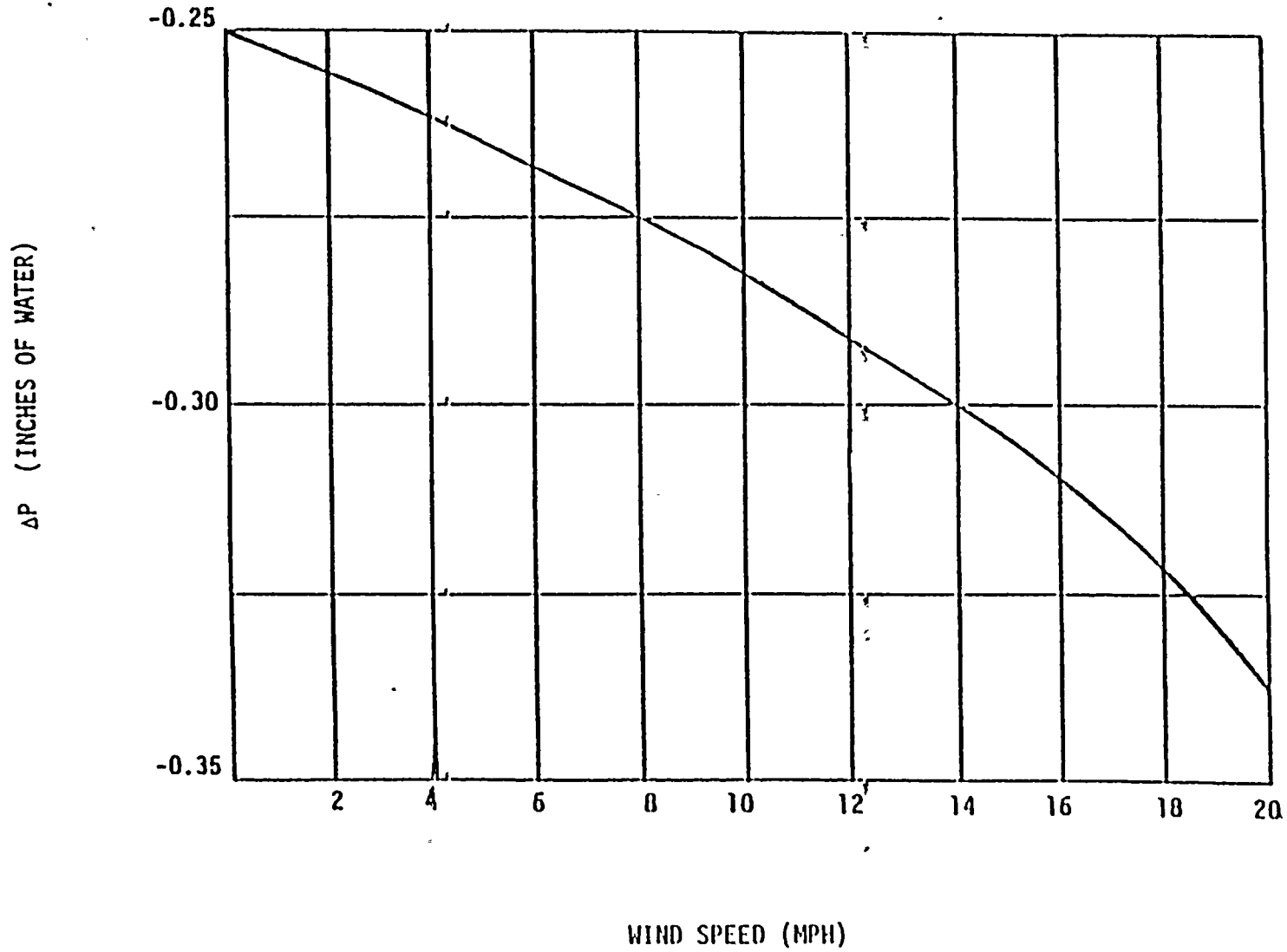
To assure the capability of the secondary containment to maintain leakage within allowable limits.

Specification:

Once during each operating cycle - isolate the reactor building and start emergency ventilation system fan to demonstrate negative pressure in the building relative to external static pressure. The fan flow rate shall be varied so that the building internal differential pressure is at least as negative as that on Figure 3.4.1 for the wind speed at which the test is conducted. The fan flow rate represents the reactor building leakage referenced to zero mph with building internal pressure at least 0.25 inch of water less than atmospheric pressure. The test shall be done at wind speeds less than 20 miles per hour.

FIGURE 3.4.1

REACTOR BUILDING DIFFERENTIAL PRESSURE



In the answers to Questions II-3 and IV-5 of the Second Supplement and also in the Fifth Supplement*, the relationships among wind speed, direction, pressure distribution outside the building, building internal pressure, and reactor building leakage are discussed. The curve of pressure in Figure 3.4.1 represents the wind direction which results in the least building leakage. It is assumed that when the test is performed, the wind direction is that which gives the least leakage.

If the wind direction was not from the direction which gave the least reactor building leakage, building internal pressure would not be as negative as Figure 3.4.1 indicates. Therefore, to reduce pressure, the fan flow rate would have to be increased. This erroneously indicates that reactor building leakage is greater than if wind direction were accounted for. If wind direction were accounted for, another pressure curve could be used which was less negative. This would mean that less fan flow (or measured leakage) would be required to establish building pressure. However, for simplicity it is assumed that the test is conducted during conditions leading to the least leakage while the accident is assumed to occur during conditions leading to the greatest reactor building leakage.

As discussed in the Second Supplement and Fifth Supplement, the pressure for Figure 3.4.1 is independent of the reactor building leakage rate referenced to zero mph wind speed at a negative differential pressure of 0.25 inch of water. Regardless of the leakage rate at these design conditions, the pressure versus wind speed relationship remains unchanged for any given wind direction.

By requiring the reactor building pressure to remain within the limits presented in Figure 3.4.1 and a reactor building leakage rate of less than 2000 cfm, exfiltration would be prevented. This would assure that the leakage from the primary containment is directed through the filter system and discharged from the 350-foot stack.

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BASES FOR 3.4.5 AND 4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

The control room air treatment system is designed to filter the control room atmosphere for intake air. A roughing filter is used for recirculation flow during normal control room air treatment operation. The control room air treatment system is designed to automatically start upon receipt of a high radiation signal from one of the two radiation monitors located on the ventilation intake and to maintain the control room pressure to the design positive pressure (one-sixteenth inch water) so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorber. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filter and charcoal adsorbers are as specified, adequate radiation protection will be provided such that resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the makeup system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 36 hours or refueling operations are terminated.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability. In addition, air intake radiation monitors will be calibrated and functionally tested each operating cycle, not to exceed 24 months, to verify system performance.

The frequency of tests and sample analysis are necessary to show the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 5-1 of ANSI 509-1980. The replacement charcoal for the adsorber tray removed for the test should meet the same adsorbent quality. Any HEPA filters found defective shall be replaced with filters qualified pursuant to ANSI 509-1980.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(8) Off-Gas and Vacuum Pump Isolation - The respective system shall be isolated or the instrument channel shall be considered inoperable and Specification 3.6.1 shall be applied.	
(9) Diesel Generator Initiation - The diesel generator shall be considered inoperable and Specification 3.6.3 shall be applied.	
(10) Emergency Ventilation Initiation - The emergency ventilation system shall be considered inoperable and Specification 3.4.4 shall be applied.	
(11) High Pressure Coolant Injection Initiation - The high pressure coolant injection system shall be considered inoperable and Specification 3.1.8.c shall be applied.	
(12) Control Room Ventilation - The control room ventilation system shall be considered inoperable and Specification 3.4.5 shall be applied.	
b. During operation with a Maximum Total Peaking Factor (MTPF) greater than the design value, either:	

TABLE 3.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1) SRM							
a. Detector not in Startup Position	2	2(a)(e)	---		x	x	
b. Inoperative	2	2(a)	---		x	x	
c. Upscale	2	2(a)	$\leq 10^5$ counts/sec		x	x	
(2) IRM							
a. Detector not in Startup Position	2	3(b)	---		x	x	
b. Inoperative	2	3(b)	---		x	x	

TABLE 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
c. Downscale	2	3(b)	≥ 5 percent of full scale for each scale		x	x	
d. Upscale	2	3(b)	≤ 88 percent of full scale for each scale		x	x	
(3) APRM							
a. Inoperative	2(h)	3(c)	---		x	x	x
b. Upscale (Biased by Recirculation Flow)	2(h)	3(c)	Figure 2.1.1(h)		x	x	x
c. Downscale	2(h)	3(c)	≥ 2 percent of full scale		(d)	(d)	x

Other than the Station turbine generator, the Station is supplied by four independent sources of a-c power; two 115 kv transmission lines, and two diesel-generators. Any one of the required power sources will provide the power required for the worst loss-of-coolant accident. The required loads of 2500 kva and 2750 kva for the loss-of-coolant are calculated in detail in the First Supplement to the FSAR. This loading is greater than that required during a Station shutdown condition. The monthly test run paralleled with the system is based on the manufacturer's recommendation for these units in this type of service. The testing during operating cycle will simulate the accident conditions under which operation of the diesel-generators is required. A detailed tabulation of the equipment comprising the maximum diesel-generator load is given in the answer to Question V-10 of the First Supplement to the FSAR.

As mentioned above, a single diesel-generator is capable of providing the required power to equipment following a major accident. Two fuel oil storage tanks are provided with piping interties to permit supplying either diesel-generator. A two-day supply will provide adequate time to arrange for fuel makeup if needed. The full capacity of both tanks will hold a four-day supply.

It has been demonstrated in Appendix E-I.3.21* that even with complete d-c loss the reactor can be safely isolated and the emergency cooling system will be operative with makeup water to the emergency cooling system shells maintained manually. Having at least one d-c battery available will permit: automatic makeup to the shells rather than manual, closing of the d-c actuated isolation valve on all lines from the primary system and the suppression chamber, maintenance of electrical switching functions in the Station and providing emergency lighting and communications power.

A battery system shall have a minimum of 106 volts at the battery terminals to be considered operable.

*FSAR

TABLE 3.6.13-1

REMOTE SHUTDOWN PANEL MONITORING

Limiting Condition for Operation

INSTRUMENT	MINIMUM NUMBER OF OPERABLE CHANNELS
Reactor Pressure	1
Reactor Water Level	1
Reactor Water Temperature	1
Torus Water Temperature	1
Drywell Pressure	1
Emergency Condenser Water Level	1
Drywell Temperature	1
"All Rods In" Light	1

TABLE 4.6.15-1
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
Surveillance Requirement

Liquid Release Type	Minimum Sampling Frequency	Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (μCi/ml)
A. Batch Waste[b] Tanks	* Each Batch	* Each Batch	Principal Gamma[c] Emitters	5×10^{-7}
			I-131	1×10^{-6}
	* Each Batch[d]	* Each Batch[d]	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	* Each Batch	Monthly Composite[e]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	* Each Batch	Quarterly Composite[e]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Service Water System Effluent	Once/month[f]	Once/month[f]	Principal Gamma[c] Emitters	5×10^{-7}
			I-131	1×10^{-6}
			Dissolved and Entrained Gases	1×10^{-5}
			H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Once/quarter[f]	Once/quarter[f]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

* Completed prior to each release.

TABLE 4.6.15-2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Surveillance Requirements

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (μCi/ml)
A. Containment Purge[b]	Each Purge	Prior to each release	Principal Gamma Emitters[c]	1×10^{-4}
	Grab Sample	Each Purge	Principal Gamma Emitters[c]	1×10^{-4}
			H-3	1×10^{-6}
B. Stack	Once/Month[d]	Once/Month[d]	Principal Gamma Emitters[c]	1×10^{-4}
	Once/Month[h]	Once/Month	H-3	1×10^{-6}
C. Stack	Continuous[e]	Once/Week[f] Charcoal Sample	I-131	1×10^{-12}
	Continuous[e]	Once/Week[f] Particulate Sample	Principal Gamma Emitters[c]	1×10^{-11}
	Continuous[e]	Once/Month Composite Particulate Sample	Gross alpha, Sr-89, Sr-90	1×10^{-11}
	Continuous[e]	Noble gas monitor	Noble Gases, Gross Gamma or Principal Gamma Emitters[c]	1×10^{-6}[g]

NOTES FOR TABLE 4.6.15-2

(a) The LLD is defined in notation (a) of Table 4.6.15-1.

(b) Purge is defined in Section 1.23.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, I-131 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.

(d) Sampling and analysis shall also be performed following shutdown, startup or an increase on the recombiner discharge monitor of greater than 50 percent, factoring out increases due to changes in thermal power level or dilution flow; or when the stack release rate is in excess of 1000 μCi/second and steady-state gaseous release rate increases by 50 percent.

(e) The sample flow rate and the stack flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.6.15.b.(1).(b) and 3.6.15.b.(3).

(f) When the release rate is in excess of 1000 μCi/sec and steady state gaseous release rate increases by 50 percent. The iodine and particulate collection device shall be removed and analyzed to determine the changes in iodine-131 and particulate release rate. The analysis shall be done daily following each change until it is shown that a pattern exists which can be used to predict the release rate; after which it may revert to weekly sampling frequency. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

(g) When RAGEMS is inoperable the LLD for noble gas gross gamma analysis shall be 1×10^{-4}.

(h) Tritium grab samples shall be taken weekly from the station ventilation exhaust (stack) when fuel is offloaded until stable tritium release levels can be demonstrated.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. Main Condenser The gross radioactivity (beta and/or gamma) rate of noble gases measured at the recombiner discharge shall be limited to less than or equal to 500,000 μCi/sec. This limit can be raised to 1 Ci/sec. for a period not to exceed 60 days provided the offgas treatment system is in operation. With the gross radioactivity (beta and/or gamma) rate of noble gases at the recombiner discharge exceeding the above limits, restore the gross radioactivity rate to within its limit within 72 hours or be in at least Hot Shutdown within the next 12 hours.	c. Main Condenser The radioactivity rate of noble gases at the recombiner discharge shall be continuously monitored in accordance with Table 3.6.14-2. The gross radioactivity (beta and/or gamma) rate of noble gases from the recombiner discharge shall be determined to be within the limits of Specification 3.6.15 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner discharge: Monthly: Within 4 hours following an increase on the recombiner discharge monitor of greater than 50%, factoring out increases due to changes in thermal power level and dilution flow changes.
d. Uranium Fuel Cycle The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.	d. Uranium Fuel Cycle Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.6.15.a.(2), 4.6.15.b.(2) and 4.6.16.b.(3) and in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

Liquid Dose

This specification is provided to implement the requirements of Section II.A, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation expressed as quarter and annual limits are set at those values found in Section II.A. of Appendix I, in accordance with Section IV.A. The Limiting Condition for Operation provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to unrestricted areas will be kept "as low as is reasonably achievable." There are no drinking water supplies that can be potentially affected by plant operations. The dose calculation methodology and parameters in the Offsite Dose Calculation Manual implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculation procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

Dose - Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation expressed as quarter and annual limits are set at those values found in Section II.B of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV-A of Appendix I to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Surveillance Requirement implements the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I, "Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977.

The Offsite Dose Calculation Manual equations provided to determine the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

Dose - Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation expressed as quarter and annual limits are set at those values found in Section II.C of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Offsite Dose Calculation Manual calculational methods specified in the Surveillance Requirement implements the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The Offsite Dose Calculation Manual calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man and 4) deposition on the ground with subsequent exposure of man.

Main Condenser

Restricting the gross radioactivity rate of noble gases from the main condenser provides assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a very small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The primary purpose of providing this specification is to limit buildup of fission product activity within the station systems which would result if high fuel leakage were to be permitted over extended periods.

Total Dose - Uranium Fuel Cycle

This specification is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to a member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contribution from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR Part 190.11 and 10CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in Specification 3.6.15.a.(1) and 3.6.15.b.(1). An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

LIMITING CONDITION FOR OPERATION

3.6.18 MARK I CONTAINMENT

Applicability:

Applies to the venting/purging of the Mark I Containment.

Objective:

To assure that the Mark I Containment is vented/purged so that the limits of specifications 3.6.15.b.(1) and 3.6.15.b.(3) are met.

Specification:

The Mark I Containment drywell shall be vented/purged through the Emergency Ventilation System unless Specification 3.6.15.b.(1) and 3.6.15.b.(3) can be met without use of the Emergency Ventilation System.

If these requirements are not satisfied, suspend all venting/purging of the drywell.

SURVEILLANCE REQUIREMENT

4.6.18 MARK I CONTAINMENT

Applicability:

Applies to the surveillance requirement for venting and purging of the Mark I Containment when required to be vented/purged through the Emergency Ventilation System.

Objective:

To verify that the Mark I Containment is vented through the Emergency Ventilation System when required.

Specification:

The containment drywell shall be determined to be aligned for venting/purging through the Emergency Ventilation System within four hours prior to start of and at least once per 12 hours during venting/purging of the drywell.

LIMITING CONDITION FOR OPERATION

3.6.22 LAND USE CENSUS

Applicability:

Applies to the performance of a land use census in the vicinity of the Nine Mile Point Nuclear Facility.

Objective:

To determine the utilization of land within a distance of three miles from the Facility.

Specification:

A land use census shall be conducted and shall identify within a distance of three miles the location in each of the 16 meteorological sectors the nearest residence and within a distance of three miles the location in each of the 16 meteorological sectors of all milk animals. In lieu of a garden census, specifications for vegetation sampling in Table 3.6.20-1 shall be followed, including analysis of appropriate controls.

With a land use census identifying a milk animal location(s) that represents a calculated D/Q value greater than the D/Q value currently being used in specification 4.6.15.b.(3), identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENT

4.6.22 LAND USE CENSUS

Applicability:

Applies to assuring that current land use is known.

Objective:

To verify the appropriateness of the environmental surveillance program.

Specification:

The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as conducting a door-to-door survey, aerial survey or consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

LIMITING CONDITION FOR OPERATION

3.7.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN DEMONSTRATIONS

Applicability:

Applies to shutdown margin demonstration in the cold shutdown condition.

Objective:

To assure the capability of the control rod system to control core reactivity.

a. The reactor mode switch may be placed in the startup position to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

   (1) The source range monitors are operable in the noncoincident condition.

   (2) The rod worth minimizer is operable per Specification 3.1.1b(3)(b) and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

SURVEILLANCE REQUIREMENT

4.7.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN DEMONSTRATIONS

Applicability:

Applies to periodic inspections required to perform shutdown margin demonstrations in the cold shutdown condition.

Objective:

To specify the inspections required to perform the shutdown margin demonstration in the cold shutdown condition.

a. Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that:

   (1) The source range monitors are operable per Specification 3.5.1.

   (2) The rod worth minimizer is operable with the required program per Specification 3.1.1b(3)(b) or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown margin demonstration procedure.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) The continuous rod withdrawal control shall not be used during out-of-sequence movement of the control rods.	(3) No core alterations are in progress.
(4) No core alterations are in progress.	
b. With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the shutdown or refuel position.	

ATTACHMENT 2

f. Linear Heat Generation Rate (LHGR)

The heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the fuel length.

g. Average Planar Linear Heat Generation Rate (APLHGR)

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 fuel rods in the specified bundle at the specified height, divided by the number of fuel rods in the fuel bundle at that height.

h. Critical Power

That assembly power which causes some point in the assembly to experience transition boiling.

i. Critical Power Ratio (CPR)

The ratio of critical power to the bundle power at the reactor condition of interest.

j. Minimum Critical Power Ratio (MCPR)

The minimum in-core critical power ratio.

k. Fraction of Limiting Power Density (FLPD)

The linear heat generation rate (LHGR) existing at a given location divided by the specified LHGR limit for that bundle type.

l. Core Maximum Fraction of Limiting Power Density (CMFLPD)

The highest value of the fraction of limiting power density which exists in the core.

SAFETY LIMIT

2.1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits on the important thermal-hydraulic variables to assure the integrity of the fuel cladding.

Specification:

a. When the reactor pressure is greater than 800 psia and the core flow is greater than 10%, the existence of a Minimum Critical Power Ratio (MCPR) less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12) shall constitute violation of the fuel cladding integrity safety limit.

b. When the reactor pressure is less than or equal to 800 psia or core flow is less than 10% of rated, the core power shall not exceed 25% of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

2.1.2 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings on automatic protective devices related to variables on which the fuel loading safety limits have been placed.

Objective:

To provide automatic corrective action to prevent exceeding the fuel cladding safety limits.

Specification:

Fuel cladding limiting safety system settings shall be as follows:

a. The flow biased APRM scram and rod block trip setting shall be established according to the following relationships:

$S \leq (0.55W + 65\%)T$ with a maximum value of 120%

$S_{RB} \leq (0.55W + 55\%)T$ with a maximum value of 110%

WHERE:

S or S_{RB} = The respective scram or rod block setpoint

W = Loop Recirculation Flow as a percentage of the loop recirculation flow which produces a rated core flow of 67.5 MLB/HR

SAFETY LIMIT

c. The neutron flux shall not exceed its scram setting for longer than 1.5 seconds as indicated by the process computer. When the process computer is out of service, a safety limit violation shall be assumed if the neutron flux exceeds the scram setting and control rod scram does not occur.

To ensure that the Safety Limit established in Specifications 2.1.1a and 2.1.1b is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

d. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be more than 6 feet, 3 inches (-10 inches indicator scale) below minimum normal water level (Elevation 302'9") except as specified in "e" below.

e. For the purpose of performing major maintenance (not to exceed 12 weeks in duration) on the reactor vessel; the reactor water level may be lowered 9' below the minimum normal water level (Elevation 302'9"). Whenever the reactor water level is to be lowered below the low-low-low level setpoint redundant instrumentation will be provided to monitor the reactor water level.

LIMITING SAFETY SYSTEM SETTING

T = FRTP/CMFLPD (T is applied only if less than or equal to 1.0)
FRTP = Fraction of Rated Thermal Power where Rated Thermal Power equals 1850 MW
CMFLPD = Core Maximum Fraction of Limiting Power Density

With CMFLPD greater than the FRTP for a short period of time, rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of rated thermal power and a notice of adjustment is posted on the reactor control panel.

b. The IRM scram trip setting shall not exceed 12% of rated neutron flux.

c. The reactor high pressure scram trip setting shall be ≤ 1080 psig.

d. The reactor water low level scram trip setting shall be no lower than -12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

SAFETY LIMIT

Written procedures will be developed and followed whenever the reactor water level is lowered below the low-low level set point (5 feet below minimum normal water level). The procedures will define the valves that will be used to lower the vessel water level. All other valves that have the potential of lowering the vessel water level will be identified by valve number in the procedures and these valves will be red tagged to preclude their operation during the major maintenance with the water level below the low-low level set point.

In addition to the Facility Staff requirements given in Specification 6.2.2.b, there shall be another control room operator present in the control room with no other duties than to monitor the reactor vessel water level.

LIMITING SAFETY SYSTEM SETTING

f. The reactor low pressure setting for main-steam-line isolation valve closure shall be ≥850 psig when the reactor mode switch is in the run position.

g. The main-steam-line isolation valve closure scram setting shall be ≤10 percent of valve closure (stem position) from full open.

h. The generator load rejection scram shall be initiated by the signal for turbine control valve fast closure due to a loss of oil pressure to the acceleration relay any time the turbine first stage steam pressure is above a value corresponding to 833 Mwt, i.e., 45 percent of 1850 Mwt.

i. The turbine stop valve closure scram shall be initiated at ≤10 percent of valve closure setting (Stem position) from full open whenever the turbine first stage steam pressure is above a value corresponding to 833 Mwt, i.e., 45 percent of 1850 Mwt.

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However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit SLCPR, operation is constrained to ensure that actual fuel operation is maintained within the assumptions of the fuel rod thermal-mechanical design and the safety analysis basis. At full power, this limit is the linear heat generation rate limit with overpower transients constrained by the unadjusted APRM scram and rod block. During steady-state operation at lower power levels, where the fraction of rated thermal power is less than the core maximum fraction of limiting power density, the APRM flow biased scram and rod block settings are adjusted by the equation in Specification 2.1.2a.

At pressure equal to or below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all core flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi.

Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28×10^3 lb/hr is approximately 3.35 MWt. With the design peaking factor, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

However, in response to expressed beliefs[7] that variation of APRM flux scram with recirculation flow is a prudent measure to assure safe plant operation during the design confirmation phase of plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of FRTP and CMFLPD. The scram setting is adjusted in accordance with Specification 2.1.1a when the core maximum fraction of limiting power density exceeds the fraction of rated thermal power.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at a constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the SLCPR. This rod block trip setting, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 110% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the core maximum fraction of limiting power density exceeds the fraction of rated thermal power, thus, preserving the APRM rod block safety margin.

b. Normal operation of the automatic recirculation pump control will be in excess of 30% rated flow; therefore, little operation below 30% flow is anticipated. For operation in the startup mode while the reactor is at low pressure, the IRM scram setting is 12% of rated neutron flux. Although the operator will set the IRM scram trip at 12% of rated neutron flux or less, the actual scram setting can be as much as 2.5% of rated neutron flux greater. This includes the margins discussed above. This provides adequate margin between the setpoint and the safety limit at 25% of rated power. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power flow condition. Effects of increasing pressure at zero or low void content are

minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit.

Procedural controls will assure that the IRM scram is maintained up to 20% flow. This is accomplished by keeping the reactor mode switch in the startup position until 20% flow is exceeded and the APRM's are on scale. Then the reactor mode switch may be switched to the run mode, thereby switching scram protection from the IRM to the APRM system.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to 1% of rated power, thus maintaining a limit above the SLCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

c. As demonstrated in UFSAR Section XV-A and B, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure scram, generator load rejection scram, and main steam isolation valve closure scram, for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

d. A reactor water low level scram trip setting -12 inches (53 inches indicator scale) relative to the minimum normal water level (Elevation 302-9") will assure that power production will be terminated with adequate coolant remaining in the core. The analysis of the feedwater pump loss in UFSAR Section XV-B.3.13 has demonstrated that approximately 4 feet of water remains above the core following the low level scram.

The operator will set the low level trip setting no lower than -12 inches relative to the lowest normal operating level. However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

e. A reactor water low-low level signal -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9") will assure that core cooling will continue even if level is dropping. Core spray cooling will adequately cool the core, as discussed in LCO 3.1.4.

The operator will set the low-low level core spray initiation point at no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9"). However, the actual set point can be as much as 2.6 inches lower due to the deviations discussed above.

f-g. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line isolation on reactor low pressure and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at ≤10% valve closure, there is no increase in neutron flux and peak pressure if the vessel dome is limited to 1141 psig. [8, 9, 10].

The operator will set the pressure trip at greater than or equal to 850 psig and the isolation valve stem position scram setting at less than or equal to 10% of valve stem position from full open. However, the actual pressure set point can be as much as 15.8 psi lower than the indicated 850 psig and the valve position set point can be as much as 2.5% of stem position greater. These allowable deviations are due to instrument error, operator setting error and drift with time.

In addition to the above mentioned Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as a secondary backup to the Limiting Safety System Setting chosen. These are as follows:

High fission product activity released from the core is sensed in the main steam lines by the high radiation main steam line monitors. These monitors provide a backup scram signal and also close the main steam line isolation valves.

The scram dump volume high level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharged from the control rod drive hydraulic system as a result of a reactor scram (Section X-C.2.10)*.

h. The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to the worst case transient of a load rejection and subsequent failure of the bypass. In fact, analysis [9,10] shows that heat flux does not increase from its initial value at all because of the fast action of the load rejection scram; thus, no significant change in MCPR occurs.

i. The turbine stop valve closure scram is provided for the same reasons as discussed in h above. With a scram setting of ≤10% valve closure, the resultant transients are nearly the same as for those described in h above; and, thus, adequate margin exists.

*UFSAR

REFERENCES FOR BASES 2.1.1 AND 2.1.2 FUEL CLADDING

(1) General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO-10958 and NEDE-10958.

(2) Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, February 1973.

(3) UFSAR Section XV-A and B.

(4) UFSAR Section XV-A and B.

(5) UFSAR Section XV-A and B.

(6) UFSAR Section XV-A and B.

(7) Letters, Peter A. Morris, Director of Reactor Licensing, USAEC, to John E. Logan, Vice-President, Jersey Central Power and Light Company, dated November 22, 1967 and January 9, 1968.

(8) UFSAR Section XV-A and B.

(9) Letter, T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.

(10) Letter, Philip D. Raymond, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, dated October 15, 1973.

(11) Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis, NEDO 24012, May, 1977.

(12) Licensing Topical Report "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

(13) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(14) General Electric SIL 299 "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation."

(15) Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-B, Amendment 10."

SAFETY LIMIT

2.2.1 REACTOR COOLANT SYSTEM

Applicability:

Applies to the limit on reactor coolant system pressure.

Objective:

To define those values of process variables which shall assure the integrity of the reactor coolant system to prevent an uncontrolled release of radioactivity.

Specification:

The reactor vessel or reactor coolant system pressure shall not exceed 1375 psig at any time with fuel in the vessel.

LIMITING SAFETY SYSTEM SETTING

2.2.2 REACTOR COOLANT SYSTEM

a. The settings on the safety valves of the pressure vessel shall be as shown below. The allowable initial set point error on each setting will be ±1 percent.

Set Point (Psig)	Number of Safety Valves
1218	4
1227	3
1236	3
1245	3
1254	3
	16

b. The reactor high-pressure scram trip setting shall be ≤1080 psig.

c. The flow biased APRM scram trip settings shall be in accordance with Specification 2.1.2a.

c. As shown in Sections XV-B.3.1 and 3.5*, rapid Station transients due to isolation valve or turbine trip valve closures result in coincident high-flux and high-pressure transients. Therefore, the APRM trip, although primarily intended for core protection, also serves as backup protection for pressure transients.

Although the operator will set the scram setting at less than or equal to that required by Specification 2.1.2a, the actual neutron flux setting can be as much as 2.7 percent of rated neutron flux above the specified value. This includes the errors discussed above. The flow bias could vary as much as one percent of rated recirculation flow above or below the indicated point.

In addition to the above-mentioned Limiting Safety System Setting, other reactor protection system devices (LCO 3.6.2) serve as secondary backup to the Limiting Safety System Setting chosen. These are as follows:

The primary containment high-pressure scram serves as backup to high reactor pressure scram in the event of lifting of the safety valves. As discussed in Section VIII-A.2.1*, a pressure in excess of 3.5 psig due to steam leakage or blowdown to the drywell will trip a scram well before the core is uncovered.

A low condenser vacuum situation will result in loss of the main reactor heat sink, causing an increase in reactor pressure. The scram feature provided, therefore, anticipates the reactor high-pressure scram. A loss of main condenser vacuum is analyzed in Section XV-B.3.1.8*.

The scram dump volume high-level scram trip assures that scram capability will not be impaired because of insufficient scram dump volume to accommodate the water discharge from the control-rod-drive hydraulic system as a result of a reactor scram (Section X-C.2.10)*.

In the event of main-steam-line isolation valve closure, reactor pressure will increase. A reactor scram is, therefore, provided on main-steam-line isolation valve position and anticipates the high reactor pressure scram trip.

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LIMITING CONDITION FOR OPERATION

3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is

SURVEILLANCE REQUIREMENT

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of axial location and average planar exposure shall be determined daily during reactor operation at ≥25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.	
b. Linear Heat Generation Rate (LHGR) During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the limiting value specified in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded at any location, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR at all locations is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until LHGR at all locations is within the prescribed limits.	b. Linear Heat Generation Rate (LHGR) The LHGR as a function of core height shall be checked daily during reactor operation at ≥25% rated thermal power.
c. Minimum Critical Power Ratio (MCPR) During power operation, the MCPR for all fuel at rated power and flow shall be within the limit provided in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within	c. Minimum Critical Power Ratio (MCPR) MCPR shall be determined daily during reactor power operation at >25% rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit.For core flows other than rated, the MCPR limit shall be the limit identified above times K_f where K_f is provided in the Core Operating Limits Report.	
d. Power Flow Relationship During Operation This power/flow relationship shall not exceed the limiting values shown in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until the power/flow relationship is within the prescribed limits.	d. Power Flow Relationship Compliance with the power flow relationship in Section 3.1.7.d shall be determined daily during reactor operation.
e. Partial Loop Operation During power operation, partial loop operation is permitted provided the following conditions are met.	e. Partial Loop Operation Under partial loop operation, surveillance requirements 4.1.7, a, b, c and d above are applicable.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(2) Primary Coolant and Containment Isolation - Isolation valves shall be closed or the valves shall be considered inoperable and Specifications 3.2.7 and 3.3.4 shall be applied.	b. Each trip system shall be tested each time the respective instrument channel is tested.
(3) Emergency Cooling Initiation or Isolation - The emergency cooling system shall be considered inoperable and Specification 3.1.3 shall be applied.	
(4) Core Spray Initiation - The core spray system shall be considered inoperable and Specification 3.1.4 shall be applied.	
(5) Containment Spray Initiation - The containment spray system shall be considered inoperable and Specification 3.3.7 shall be applied.	
(6) Auto Depressurization Initiation - The auto depressurization system shall be considered inoperable and Specification 3.1.5 shall be applied.	
(7) Control Rod Withdrawal Block - No control rods shall be withdrawn.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(8) Off-Gas and Vacuum Pump Isolation - The respective system shall be isolated or the instrument channel shall be considered inoperable and Specification 3.6.1 shall be applied.	
(9) Diesel Generator Initiation - The diesel generator shall be considered inoperable and Specification 3.6.3 shall be applied.	
(10) Emergency Ventilation Initiation - The emergency ventilation system shall be considered inoperable and Specification 3.4.4 shall be applied.	
(11) High Pressure Coolant Injection Initiation - The high pressure coolant injection system shall be considered inoperable and Specification 3.1.8.c shall be applied.	
(12) Control Room Ventilation - The control room ventilation system shall be considered inoperable and Specification 3.4.5 shall be applied.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. During operation with the Core Maximum Fraction of Limiting Power Density (CMFLPD) greater than the Fraction of Rated Thermal Power (FRTP), either: (1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2a; or (2) The APRM gain shall be adjusted in accordance with Specification 2.1.2a; or (3) The power distribution shall be changed such that the CMFLPD no longer exceeds FRTP.	c. During reactor power operation at ≥25 percent rated thermal power, the Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked daily and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified by Specification 2.1.2a.

TABLE 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(ii) Inoperative	2	3(d)(o)	---		x	x	
(b) APRM							
(i) Upscale	2	3(e)(o)	Specification 2.1.2a		x	x	x
(ii) Inoperative	2	3(e)(o)	---		x	x	x
(iii) Downscale	2	3(e)(o)	≥ 5 percent of full scale		(g)	(g)	(g)
(10) Turbine Stop Valve Closure	2	4(o)	≤ 10% valve closure				(i)
(11) Generator Load Rejection	2	2(o)	(j)				(i)

NOTES FOR TABLES 3.6.2a and 4.6.2a

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed in the refuel and shutdown positions of the reactor mode switch with a keylock switch.

(c) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi, or for the purpose of performing reactor coolant system pressure testing and/or control rod scram time testing with the reactor mode switch in the refuel position.

(d) No more than one of the four IRM inputs to each trip system shall be bypassed.

(e) No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. A Traversing In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing.

(f) Calibrate prior to startup and normal shutdown and thereafter check once per shift and test once per week until no longer required.

(g) IRM's are bypassed when APRM's are onscale. APRM downscale is bypassed when IRM's are onscale.

(h) Each of the four isolation valves has two limit switches. Each limit switch provides input to one of two instrument channels in a single trip system.

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

(k) May be bypassed when placing the reactor mode switch in the SHUTDOWN position and all control rods are fully inserted.

(l) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2a, the primary sensor will be calibrated and tested once per operating cycle.

(m) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during reactor operation when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 2.1.2a shall not be included in determining the absolute difference.

TABLE 3.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System (i)	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
c. Downscale	2	3(b)	≥ 5 percent of full scale for each scale		x	x	
d. Upscale	2	3(b)	≤ 88 percent of full scale for each scale		x	x	
(3) APRM							
a. Inoperative	2(h)	3(c)	---		x	x	x
b. Upscale (Biased by Recirculation Flow)	2(h)	3(c)	Specification 2.1.2a (h)		x	x	x
c. Downscale	2(h)	3(c)	≥ 2 percent of full scale		(d)	(d)	x

Changes to the Offsite Dose Calculation Manual (ODCM): Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(s);

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and found acceptable.

f. CORE OPERATING LIMITS REPORT

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

   1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.1.7.a and 3.1.7.e.

   2) The K_f core flow adjustment factor for Specification 3.1.7.c.

   3) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.1.7.c and 3.1.7.e.

   4) The LINEAR HEAT GENERATION RATE for Specification 3.1.7.b.

   5) The Power/Flow relationship for Specification 3.1.7.d and e.

   and shall be documented in the CORE OPERATING LIMITS REPORT.

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.